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Risk-Based Performance Indicators: Results of Phase 1 Development

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H. G. Hamzehee,[1] Principal Author

Contributing Authors:
R. W. Youngblood[2], S. A. Eide[3], R. F. Buell[3],
H. Dezfuli[2], C. Atwood[3], C. Smith[2], R. Bertucio[4],
L. Wolfram[3], J. F. Meyer[2], F. Zikria[2], K. Green[2]

[1]Division of Risk Analysis and Applications
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

[2]ISL, Inc.
11140 Rockville Pike
Rockville, MD 20852

[3]Idaho National Engineering and Environmental Laboratory
2525 N. Fremont Avenue
Idaho Falls, ID 83415

[4]SCIENTECH, Inc.
400 West Gowe Street
Kent, WA 98032

ABSTRACT

This report presents the results of the Phase 1 development of risk-based performance indicators (RBPIs) to potentially enhance the Reactor Oversight Process (ROP). SECY-99-007 recognized that improved performance indicators may be developed as part of the evolution of the ROP. RBPIs reflect changes in licensee performance that are logically related to risk and associated models. To the extent practical, the RBPIs identify declining performance before performance becomes unacceptable, without incorrectly identifying normal variations as degradations (i.e., avoid false-positive indications and false-negative indications). Phase 1 of the RBPI development includes performance indicators that are related to the initiating events cornerstone, mitigating systems cornerstone, and the containment portion of the barrier integrity cornerstone. The potential integration of RBPIs into the ROP would follow the change process described in IMC-0608, "Performance Indicator Program."

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EXECUTIVE SUMMARY

Purpose

The purpose of this report is to present the results of the Phase 1 development of risk-based performance indicators (RBPIs) to potentially enhance the Reactor Oversight Process (ROP). The White Paper entitled "Development of Risk-Based Performance Indicators: Program Overview" described the concepts for the RBPI development. The purpose of the RBPI development is to examine the technical feasibility of providing improved performance indicators for potential implementation in the ROP. Phase 1 of the RBPI development includes indicators that are related to the initiating events cornerstone, the mitigating systems cornerstone, and the containment portion of the barrier integrity cornerstone. In addition, industry-wide trending is provided to support the agency's Strategic Plan Performance Measures, provide input to assessing the ROP's effectiveness, and feedback insights to the inspection program.

This work is part of the development and evolution of performance indicators in the current ROP and is closely coordinated with existing ROP efforts. There are several key implementation issues summarized in this executive summary and Section 6 of the report, including the verification of risk models and data. The potential integration of RBPIs into the ROP would follow the guidelines in IMC-0608, "Performance Indicator Program." This would include a pilot program prior to the full implementation of RBPIs and interaction with stakeholders to resolve implementation issues raised in this report or by external stakeholders during the review of this report.

What Are RBPIs?

RBPIs reflect changes in licensee performance that are logically related to risk and associated models. That is, they provide performance measures whose impact on core damage frequency (CDF) and large early release frequency (LERF) can be established through a risk model or risk logic. In developing RBPIs, "performance" refers to the conduct of activities in design, procurement, construction, operation, and maintenance that support achievement of the objectives of the cornerstones of safety in the ROP.

The RBPIs developed in this report have the following characteristics:

- The RBPIs are compatible with, and complementary to, the risk-informed inspection activities of the oversight process.
- The RBPIs cover all modes of plant operation.
- Within each mode, the RBPIs cover risk-important SSCs to the extent practical.
- The RBPIs are capable of implementation without excessive burdens to licensees or NRC in the areas of data collection and quantification.
- To the extent practical, the RBPIs identify declining performance before performance becomes unacceptable, without incorrectly identifying normal variations as degradations (i.e., avoid false positive indications and false negative indications).
- The RBPIs are amenable to establishment of plant-specific thresholds similar to the ROP.

In addition to plant-specific RBPIs, some risk-significant aspects of performance that cannot be effectively assessed on a plant-specific basis have been identified for industry-wide trending. This task provides an input for measuring the effectiveness of the overall ROP, as well as supporting the agency's Strategic Plan Performance Measures.

Potential Benefits of RBPIs

The ROP uses two methods for monitoring plant performance, cumulative indicators and individual findings from inspections. Both methods provide indications that are evaluated with respect to their risk significance, and are used to determine the level of NRC oversight. The current ROP utilizes performance indicators that measure plant performance and use generic performance thresholds as described in SECY-99-007, "Recommendations for Reactor Oversight Process Improvement." SECY-99-007 recognized that improved performance indicators may be developed as part of the evolution of the ROP.

RBPIs are intended to provide improved indicators for the ROP. However, the decision to use the candidate RBPIs, in whole or in part, in the ROP will be made as part of the established ROP change process.

Subsequent to the closing of the comment period for this report, the agency and industry (through the continuing ROP interactions) have identified several aspects of unreliability and unavailability indicators from the RBPI development that will be piloted in 2002 for potential implementation in the ROP. These involve unreliability and unavailability indicators associated with the six SSUPIs under the mitigating system cornerstone of the current ROP.

In addition to RBPIs, selected performance areas will be trended on an industry-wide basis. The industry-wide trending efforts support the Strategic Plan Performance Measures. Specifically, the industry-wide trending from this program along with trending from other programs, such as the Accident Sequence Precursor (ASP) Program, will be used to assess performance against the Nuclear Reactor Safety measure: "no statistically significant adverse industry trends in safety performance." SECY-01-0111, titled "Development of an Industry Trends Program for Operating Power Reactors," describes the intended approach for using industry trend information in regulatory applications.

The RBPI development potentially provides the following benefits to the ROP:

- More comprehensive coverage of significant contributors to plant risk
  - Unreliability indicators were developed at the component/train/system level.
  - Indicators for shutdown modes were developed. RBPIs for fire and the containment portion of the barrier integrity cornerstone were identified consistent with the state-of-the-art models, data, and methods currently available for these areas.

- More recognition of plant-specific attributes
  - The RBPI threshold values are more plant-specific and reflect risk-significant differences in plant designs.
- Industry-wide trending of plant-specific RBPIs as well as risk-significant performance measures that are impractical to monitor on a plant-specific basis.
  - Trending provides measures of the ROP effectiveness.
  - Trending provides feedback to the ROP to adjust technical emphasis and overall inspection frequencies.
  - Trending provides input to the agency's Strategic Plan Performance Measures.

Risk Perspectives on RBPI Development

The thresholds in the ROP for performance indicators and the Significance Determination Process (SDP) are based on changes in the CDF of approximately 1E-6, 1E-5, and 1E-4 per year. CDF changes associated with the lower thresholds are only a fraction of the total CDF at a plant. Changes in performance corresponding to the red performance band (ΔCDF above 1E-4 per year) are on the same order of magnitude as our current estimates of total CDFs. Thus, the ROP thresholds represent a graded approach that responds to larger increases in risk with greater regulatory response. In addition, our understanding of public risk corresponding to these values for CDF indicates that margin exists between the risk associated with performance changes at the ROP thresholds and both the Quantitative Health Objectives (QHOs) of the Commission's Safety Goal Policy Statement and the existing individual risk of accidental death (approximately a factor of 25 and 2500, respectively).

An inherent implication of monitoring risk attributes is that there is a time delay between the onset of a change in performance and the ability of the indicator to detect that a change has occurred. In this sense, all indicators are "lagging," or at best concurrent with, the performance change being monitored. In the case of RBPIs, this is not a significant issue because each indicator represents one of many elements of risk, for which there is still margin to the agency-stated public health objectives. Thus, the indicators are "lagging" for the parameter monitored, but "leading" indicators for overall risk.

In addition, operating experience does not indicate that the large changes in the reliability of equipment or the frequency of initiators necessary to cause an indicator to go from nominal to unacceptable performance occur often. However, even if large changes occur, the monitoring intervals and thresholds have been set so that the probability of failure to detect that performance has changed over the monitoring period is low and the incremental risk accumulation over that time is small compared to the QHOs and individual accidental death risk.

Summary of Results

The Phase 1 RBPI development identified performance indicators and areas for industry-wide trending for potential use in the ROP. The risk elements were disaggregates, to develop thresholds for the indicators that would consistently reflect the risk impact of performance changes. For the majority of RBPIs, train-level rather than system-level indicators were required, and unreliability and unavailability were treated separately, rather than as a combined failure probability. The differences in the risk implication of performance changes at these levels are inherent in the calculation of risk. While performance can be monitored at other levels, setting thresholds with consistent risk implications between and among indicators is problematic. A total of 21 indicators for pressurized water reactors (PWRs) and 16 indicators for boiling water reactors (BWRs) were identified (including proposed RBPIs with no current data reporting). These RBPIs are listed in Table ES-1and are briefly discussed below.

Initiating Events Cornerstone

Three initiating event frequency indicators for internal events were identified under the initiating events cornerstone of safety: general transients, loss of feedwater, and loss of heat sink. Other risk-significant initiating events did not accumulate data in a timely manner for plant-specific assessment of performance due to their low frequencies, and were therefore included in the industry-wide trending. Fire initiating events and the risk-significant initiating events during the shutdown modes were also included in the industry-wide trending due to their low frequencies.

Mitigating Systems Cornerstone

For power operation, 13 RBPIs for BWRs and 18 RBPIs for PWRs under the mitigating systems cornerstone of safety were identified. These involved unreliability and unavailability indicators at the train-level for risk-significant safety systems and cross-system performance of key components. RBPIs for key components were developed to help assess cross-cutting performance issues that might not be practical to detect by an individual system or train performance indicator.

The thresholds for these indicators are currently based on plant-specific assessment of CDF changes. Some of these systems may also affect LERF, and it is possible for thresholds determined from changes in LERF to be more limiting than thresholds determined from changes in CDF. However, the LERF models and related data needed to determine these thresholds are not currently available.

For shutdown modes of operation, four potential RBPIs under the mitigating systems cornerstone of safety for PWRs and BWRs were proposed. They monitor time spent in risk-significant shutdown configurations. The risk-significant shutdown configurations are combinations of equipment unavailabilities and the reactor states associated with decay heat rates, reactor coolant system (RCS) integrity, and RCS level. The threshold values are generic and reflect CDF changes associated with spending excess time in the more risk-significant shutdown configurations. The generic baseline performance values were based on the past performance data for a number of plants. The generic threshold values were derived using two shutdown risk models (one for a PWR, and one for a BWR) that are representative of risk during shutdown operation. Internal and external stakeholder comments indicated that the approach presented for potential shutdown RBPIs was more appropriate for potential application in the Significance Determination Process (SDP) of the ROP. Consequently, this report documents the technical work associated with shutdown RBPIs, but future use of this effort will concentrate on evaluating the significance of shutdown conditions for the SDP.

Potential RBPIs for fire events under the mitigating systems cornerstone of safety were identified. These RBPIs were related to the unreliability and unavailability of fire detection/suppression systems. However, the models and data currently available are not amenable for use in determining RBPIs.

Containment Portion of Barrier Integrity Cornerstone

Potential RBPIs for the containment systems affecting LERF for selected containment types were identified. These involved the containment isolation function, the drywell spray system, and the hydrogen ignitor function. However, baseline performance values for these potential containment RBPIs could not be determined due to the unavailability of performance data. LERF models for setting thresholds are not available for all containment types. In addition, the available models are not compatible with the Standardized Plant Analysis Risk (SPAR) Revision 3 models for assessing the CDF impacts which are the inputs to the LERF models. Therefore, no containment RBPIs are provided.

Industry-Wide Trending

The industry-wide trending includes the plant-specific RBPIs as well as risk-significant performance areas that cannot be monitored at a plant-specific level. The RBPIs each have the characteristic that there is sufficient data for plant-specific trending so that plant-specific performance changes can be detected in a timely manner. Some performance features that are risk-significant do not occur frequently enough to be trended on a plant-specific basis, but can be trended on an industry basis. For example, loss-of-offsite-power events during power operations occur on average about once every 20 reactor years. Thus, approximately five events are expected to occur each year in the industry. These events can be trended at the industry level, but are not amenable to plant-specific monitoring. The industry trending will consist of trending of each of the RBPIs identified earlier as well as the performance elements noted in Table ES-2. SECY-01-0111 describes the intended approach for using industry trend information in regulatory applications.

Risk Coverage

As part of this RBPI development effort, an evaluation was done to assess the extent of risk coverage by RBPIs and industry-wide trending. Approximately 40% of the risk-significant elements in the SPAR models were covered by RBPIs.

In addition, the dominant accident sequences from the Individual Plant Examination (IPE) database were reviewed. Most of the dominant accident sequences had one or more events covered by RBPIs or industry-wide trending. Tables (4-2a and 4-2b) are provided in the report to show which elements of the dominant accident sequences were covered and which ones were not.

Key Issues Affecting Feasibility of Potential Implementation of RBPIs

Several implementation issues have emerged in the course of the development described in this report. These are discussed briefly below.

Are any additional performance indicators needed in the ROP?

Interactions with stakeholders commenting on the White Paper indicated differing views on this subject. Industry representatives questioned whether NRC needed to have a broader coverage of risk measured in the ROP indicators, especially if the coverage did not result in a corresponding reduction in the inspection program. Other external stakeholder comments favored more indicators as well as additional inspections.

The RBPI development program is focused on demonstrating the technical feasibility of providing additional objective indicators that cover a broader spectrum of risk-significant plant performance. Future work may identify additional candidates. Any potential new performance indicators will be assessed in a pilot program consistent with the change process described in IMC-0608 prior to implementation.

Subsequent to the closing of the comment period for this report, the agency and industry (through the continuing ROP interactions) have identified several aspects of unreliability and unavailability indicators from the RBPI development that will be piloted in 2002 for potential implementation in the ROP. These involve unreliability and unavailability indicators associated with the six SSUPIs under the mitigating system cornerstone of the current ROP.

Is the number of potential new indicators appropriate? Which of the proposed indicators would be most beneficial?

The RBPI Phase 1 development identified 22 potential indicators for PWRs and 17 potential indicators for BWRs. If all of these performance indicators were implemented, they could potentially replace 8 (3 initiating event and 5 mitigating system) of 18 existing indicators in whole or in part, bringing the total number of indicators per plant to about 30. In addition to the issue of the appropriate risk scope of ROP indicators (noted above), it will be necessary to assess whether potentially expanding the total number of indicators to approximately 30 per plant (approximately 25 of them based on currently available data) is reasonable from a logistics/process point of view. For example, the criteria that result in plants entering various columns of the Action Matrix will have to be reconsidered. Section 6.5 discusses results of preliminary work to examine the feasibility of developing indicators at a higher level (system or cornerstone level) by combining results of lower level data and models. In follow-on work, higher level indicators may be investigated further.

Do the data sources for RBPIs exist and have sufficient quality for use in the ROP?

A significant portion of the RBPIs require access to and use of data from the Equipment Performance and Information Exchange (EPIX) system. These data are voluntarily provided by industry in response to the Commission decision to forgo the Reliability Data Rule. Full industry participation, verification, and validation of the existing EPIX and the development of guidelines for consistent reporting are important to the feasibility of many RBPIs as potential improvements to the ROP.

Performance data are not readily available from EPIX for several of the proposed indicators. The NRC is working with industry groups to expand the reliability data collection in this voluntary system to include data that will support evaluation of performance in these areas.

Data accuracy and licensee burden in this area are recognized as important implementation issues, which will be further investigated during the implementation phase using the change process in IMC-0608.

Will SPAR Revision 3i models be used for setting plant-specific thresholds for all plants?

Approximately 50 Standardized Plant Accident Risk (SPAR) Revision 3i models are currently available. Completion of all 70 SPAR Revision 3i models is scheduled for the end of calendar year 2002. As more models are made available for use in the RBPI development program, it will be possible to determine if plants can be grouped so that a few models can be used to set thresholds for all plants or individual models will be needed for each. The RBPI development program will continue to use the SPAR Revision 3i models as they are developed. External stakeholder comments on the White Paper indicated that peer review by licensees should be included in the development of these models. Two additional implementation issues are whether licensees or NRC will calculate the thresholds and indicators and whether licensee models (meeting as-yet-to-be-developed NRC specifications) could be used instead of the SPAR models.

It is yet to be determined whether a plant-specific model will be required to set performance thresholds for each plant or whether a representative model is sufficient for a group of plants. Furthermore, it has not been determined whether the calculation for thresholds and indicators will be routinely performed by NRC staff using SPAR Revision 3i models, licensees using SPAR Revision 3i models, or licensees using their own risk models that meet specifications agreed upon and reviewed by the NRC. These are potential options that will be dealt with through IMC-0608.

Will LERF models be used for setting thresholds for mitigating and containment systems?

There are a limited number of large early release frequency (LERF) models available to set thresholds for performance of systems that impact the integrity of the containment barrier. In addition, currently available data are inadequate for establishing performance measures for the containment systems. Also, for some systems under the mitigating systems cornerstone, the thresholds associated with changes in core damage frequency (CDF) due to performance degradations may not be limiting compared to changes in LERF. To assess that condition, LERF models that reflect the impact of potential CDF changes are needed. The current plan for developing LERF models over the next several years will support limited capability for identifying RBPIs or setting plant-specific LERF thresholds.

Table ES-1 Summary of Phase 1 Risk-Based Performance Indicators

Safety Cornerstone	Existing PIs	Proposed RBPIs			
Initiating Event	- Unplanned scram - LONHR - Unplanned reactor power changes	- General transient - LOFW - LOHS			
		PWR at Power	**BWR at Power**	**Shutdown**	**Fire**
Mitigating System	- EPS (UA) - RHR (UA) - PWR AFW (UA) HPI (UA) - BWR HPCS/HPCI (UA) RCIC/IC (UA) - Safety system functional failures	- EPS (UR&UA) - AFW-MDP (UR&UA) - AFW-TDP (UR&UA) - HPI (UR&UA) - PORV (UR) - RHR (UR&UA) - SWS (UR&UA) - CCW (UR&UA) - AOV (UR) - MOV (UR) - MDP (UR)	- EPS (UR&UA) - HPCS/HPCI (UR&UA) - RCIC/IC (UR&UA) - RHR (UR&UA) - SWS (UR&UA) - AOV (UR) - MOV (UR) - MDP (UR)	- *Time in High, medium, low risk-significant, or early reduced inventory (vented) configurations	None
Barriers	- RCS specific activity - RCS identified leak rate	- *CIV (UR&UA)	- *Drywell spray (MarkI)(UR&UA) - *CIV (Mark III) (UR&UA)	None	None

* Requires data that are not currently reported.

Note: The emergency preparedness, occupational radiation safety, public radiation safety, and physical protection cornerstones of safety are not included in the Phase 1 RBPI scope.

Table ES-2 Summary of Phase 1 Performance Areas Proposed for Industry-Wide Trending

Safety Cornerstone	Industry-Wide Trend
Initiating Event	Full Power – All proposed IE RBPIs listed in Table ES-1 – Internal flooding – Initiators evaluated as ASPs – Loss of instrument/control air (for BWRs and PWRs) – LOOP – Loss of vital AC bus – Loss of vital DC bus – Small LOCA (including very small LOCA) – SGTR – Stuck open safety/relief valves Shutdown – LOOP during shutdown modes – Loss of RHR during shutdown modes – Loss or diversion of RCS inventory during shutdown modes leading to loss of RHR – Loss of RCS level control (during transition to mid-loop) leading to loss of RHR (for PWRs only) Fire – Fire events in risk-significant fire areas
Mitigating System	– All proposed mitigating system RBPIs listed in Table ES-1 – CCF events for AFW pumps – CCF events for diesel generators – Total CCF events
Barriers	None

FOREWORD

The Reactor Oversight Process (ROP) was implemented to improve the NRC's regulatory oversight of licensee operation of commercial nuclear power plants. It is intended to better risk-inform agency actions and bring more objectivity to the regulatory process. The ROP is consistent with the goals of the Commission's PRA Policy Statement and the NRC's Strategic Plan (NUREG-1614), which include increased use of the PRA technology in "regulatory matters to the extent supported by the state-of-the-art in PRA methods and data and in a manner that complements the NRC's deterministic approach and supports the NRC's traditional defense-in-depth philosophy." The development of the potential risk-based performance indicators (RBPIs) described in this report is intended to represent a further improvement to the ROP that would be appropriate as part of this regulatory evolution.

SECY-99-007 and 99-007A described the revised Reactor Oversight Process. The ROP was implemented at all plants in April 2000 following a 6-month pilot program conducted in 1999. The results of this pilot program were described in SECY-00-0049. A fundamental aspect of the ROP is the use of both performance indicators and inspection findings to determine whether the objectives of the ROP's cornerstones of safety are being met on a plant-specific basis.

In addition to these changes at the NRC, the industry is using more performance-based approaches to enhance its operations, including gathering and analyzing both plant-specific and industry-wide data. Furthermore, technological advances such as the Internet have resulted in improved capabilities to gather and share such data. Through such technological developments, both the industry and the NRC have expanded their capabilities to model and assess the risk-significance of plant operations.

In light of these evolving capabilities and the movement toward more risk-informed and performance-based oversight, the Risk-based Performance Indicators were developed to (1) address specific areas in the current ROP that were identified in SECY-00-0049 as possible enhancements and (2) potentially support any future development of performance indicators using improved risk analysis tools. This report discusses the technical feasibility of using currently available risk models and data to enhance the NRC's ability to monitor plant-specific safety performance of reactors in a risk-informed and performance-based manner. This development activity is designed to fit into the ROP concept for indicators, thresholds, and performance monitoring while continuing to move the NRC's programs forward in accordance with the PRA Policy Statement and the goals of the Strategic Plan.

The Strategic Plan also articulates the NRC's efforts to increase public confidence. One of the strategies for achieving that goal is as follows: "We will make public participation in the regulatory process more accessible. We will listen to the public's concerns and involve our stakeholders more fully in the regulatory process." In keeping with this philosophy, the NRC has sought, and continues to seek, input from internal and external stakeholders on the ROP as the program evolves. With respect to the development of potential RBPIs, the first key stakeholder interactions were held to obtain input to the RBPI White Paper (SECY-00-0146), which described the principles for the RBPI development. This report represents the second opportunity for external stakeholder participation in the RBPI development process. There will be additional opportunities for internal and external stakeholder involvement as the process continues to evaluate the feasibility of potential implementation of these (or other) performance indicators in accordance with the ROP change process.

During recent ROP stakeholder discussions, the NRC and the industry have identified several aspects of the unreliability and unavailability indicators from the RBPI development that would potentially enhance the current set of the ROP performance indicators. As a result, the NRC staff is working with the industry to start a pilot program in early 2002. This pilot program will include unreliability indicators for six mitigating systems in the ROP, as well as changes to the current Safety System Unavailability Performance Indicators (SSUPIs). In addition, some follow-on development work will be continued, as summarized below:

- Development of enhanced performance indicators for the containment portion of the barrier integrity cornerstone of safety.
- Development of an approach for determining plant-specific performance thresholds for unavailability and unreliability indicators.
- Development of performance indicators at higher levels, such as at the system or function level.
- Technology transfer of shutdown RBPI results and insights to support the Significance Determination Process (SDP) for shutdown modes.

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ABBREVIATIONS AND ACRONYMS

AC	vital AC buses
ACBU1	other onsite backup 1
ACC	accumulators
ADS	automatic depressurization system
AFW	auxiliary feedwater
AM1	alternate makeup 1
AM2	alternate makeup 2
AOV	air-operated valve
ARI	alternate rod insertion
ASP	accident sequence precursor
ASPC	alternate suppression pool cooling
AUXC1	auxiliary cooling 1
AUXC2	auxiliary cooling 2
BI	borated injection
BWR	boiling water reactor
CCDP	conditional core damage probability
CCDF	conditional core damage frequency
CCF	common cause failure
CCW	component cooling water
CD	core damage
CDF	core damage frequency
CDP	core damage probability
CHPI	normally running makeup (injection)
CHPR	normally running makeup (during recirculation)
CIV	containment isolation valve
CONDA	condenser available
CRDS	control rod drive pumps
CS	core spray
CSR	containment spray recirculation

CTS	condensate pumps
DBI	design basis issue
DDP	diesel-driven pump
DWS	drywell spray
EAC	emergency AC power (usually EDGs)
EDC	battery-backed DC buses
EDG	emergency diesel generator
EPIX	Equipment Performance and Information Exchange System
EPS	emergency power system
ESAS1	engineered safety actuation system 1
ESW	emergency service water
FTLR	Fail to Load and Run
FTR	Fail to Run
FTS	Fail to Start
GT	general transients
HP1	high-pressure 1
HPCI	high-pressure coolant injection
HPCS	high-pressure core spray
HPI	high-pressure injection system
HPR	high head safety injection (during recirculation)
HUM	operator action
HVAC	heating, ventilation, air conditioning
HVAC1	heating, ventilation, air conditioning 1
HVAC2	heating, ventilation, air conditioning 2
HVAC3	heating, ventilation, air conditioning 3
IA	instrument air compressors
IC	isolation condenser
INEEL	Idaho National Engineering and Environmental Laboratory
INPO	Institute for Nuclear Power Operations
IPE	Individual Plant Examination
IPEEE	Individual Plant Examination for External Events

ISLOCA	interfacing systems LOCA
LER	Licensee Event Report
LERF	large early release frequency
LLOCA	large loss-of-coolant accident
LOCA	loss-of-coolant accident
LOFW	loss of feedwater
LOHS	loss of heat sink
LONHR	loss of normal heat removal
LOOP	loss-of-offsite-power event
LOSP	loss of offsite power
LP1	low-pressure 1
LP2	low-pressure 2
LP3	low-pressure 3
LPCI	low-pressure coolant injection
LPCS	low-pressure core spray
LPI	low-pressure injection
LPR	low-pressure recirculation
MDAFW	motor-driven auxiliary feedwater pumps
MDP	motor-driven pump
MFW	main feedwater pumps
MLE	Maximum Likelihood Estimate
MLOCA	medium loss-of-coolant accident
MOR	monthly operating report
MOV	motor-operated valve
MSIV	main steam isolation valve
NISP	non-1E startup pumps
NRC	U.S. Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
OA3	alternate air system 3
PORV	power-operated relief valve
PPORV	pressurizer power-operated relief valves

PRA	probabilistic risk assessment
PSRV	pressurizer safety relief valve
PWR	pressurized water reactor
QHO	Quantitative Health Objective
RADS	Reliability and Availability Database System
RAW	Risk Achievement Worth
RBCLCW	reactor building closed loop cooling water
RBPI	risk-based performance indicator
RCIC	reactor core isolation cooling
RCPS	reactor coolant pump seal
RCS	reactor coolant system
RECIRC	recirculation pumps
RHR	residual heat removal
ROP	Reactor Oversight Process
RPS	reactor protection system
RWST	refueling water storage tank
SBO	station blackout
SCSS	sequence coding and search system
SDC	shutdown cooling
SDP	Significance Determination Process
SG	steam generator
SGTR	steam generator tube rupture
S1	medium Loss-of-Coolant Accident
S2	small loss-of-coolant accident
S3	small-small loss-of-coolant accident
SDAFW	steam-driven auxiliary feedwater pumps
SGA	steam generator atmospheric dump valves
SGS	steam generator safety valves
SI	safety injection
SLC	standby liquid control
SLOCA	small loss-of-coolant accident

SPAR	Standardized Plant Analysis Risk
SPC	suppression pool cooling
SRV	safety relief valve
SRVS	safety relief valves steam
SSCs	systems, structures, and components
SSW	standby service water
SSUPI	safety system unavailability performance indicator
SW2	alternate service water 2
SW3	alternate service water 3
SWS	service water system
T&M	test & maintenance
T-AC	transient - initiated by loss of vital AC buses
T-ATWS	transient - anticipated transient without scram
T-AUXC2	transient - initiated by loss of auxiliary cooling 2
T-CCW	transient - initiated by loss of component cooling water
T-DC	transient - initiated by loss of DC buses
T-ESW	transient - initiated by loss of essential service water pumps
T-EXFW	transient - excessive feedwater addition
T-HVAC1	transient - initiated by loss of heating, ventilation, and air conditioning 1
T-HVAC2	transient - initiated by loss of heating, ventilation, and air conditioning 2
T-IA	transient - initiated by loss of instrument air compressors
T-IFL	transient - internal flood
T-IORV	transient - inadvertent open relief valve
T-IORV/ SORV	transient - inadvertent or stuck open relief valve
T-LMFW	transient - loss of main feedwater
T-LOOP	transient - loss of offsite power
T-MSIV	transient - initiated by failure of main steam isolation valve
T-NSW	transient - initiated by loss of normal service water pumps
T-RX	transient - reactor trip
T-SGTR	transient - steam generator tube rupture

T-SLBIC	transient - steam line break inside containment
T-SLBOC	transient - steam line break outside containment
T-SW2	transient - initiated by loss of alternate service water 2
T-TBCLCW	transient - initiated by loss of turbine building closed loop cooling water
T-TT	transient - turbine trip
T-UHS	transient - loss of ultimate heat sink
T-VAC	transient - initiated by loss of vital instrument AC
TB	turbine bypass valves
TDP	turbine-driven pump
UA	unavailability
UR	unreliability
V	interfacing system loss-of-coolant accident
V&V	validation and verification
V-AR1	interfacing system loss-of-coolant accident in alternate recirculation 1
V-CCW	interfacing system loss-of-coolant accident in component cooling water
V-CHPI	interfacing system loss-of-coolant accident in normally running makeup (injection)
V-HPI	interfacing system loss-of-coolant accident in high head safety injection
V-LPI	interfacing system loss-of-coolant accident in low-pressure injection
V-RHR	interfacing system loss-of-coolant accident in residual heat removal
VAC	vital instrument AC
VENT	venting system

1. INTRODUCTION

1.1 Purpose of Report

The purpose of this report is to present the results of the Phase 1 development of risk-based performance indicators (RBPIs) to potentially enhance the Reactor Oversight Process (ROP). The development process was previously described in the White Paper entitled "Development of Risk-Based Performance Indicators: Program Overview" (Ref. 1, provided here as Appendix G).

This work is part of the development and evolution of performance indicators in the current ROP and is closely coordinated with existing ROP efforts. Changes to the existing ROP indicators are not imminent. There are several key implementation issues summarized in the executive summary and Section 6 of the report, including the verification of risk models and data. The potential integration of RBPIs into the ROP would follow the guidelines in IMC-0608, "Performance Indicator Program." This would include a pilot program prior to the full implementation of RBPIs and interaction with stakeholders to resolve implementation issues raised in this report or by external stakeholders during the review of this report.

The current results presented include:

- Plant-specific RBPIs and their thresholds for 44 plant models;
- Assessment of risk coverage provided by potential RBPIs; and
- Results of validation and verification.

In addition, potential candidates for industry-wide trending were identified.

In addition to the Phase 1 results, this report describes the process for RBPI development. This process is intended to lead to a set of RBPIs having the characteristics discussed in Section 1.2 of this report.

The Phase 1 RBPI development includes indicators that are related to the initiating events cornerstone, the mitigating systems cornerstone, and the containment portion of the barrier integrity cornerstone. This includes assessment of feasibility and development of potential indicators for:

- Initiating events;
- Unreliability and unavailability performance under the mitigating systems cornerstone;
- Containment barrier performance;
- Performance areas involved in scenarios initiated by fire;
- Performance areas involved in scenarios initiated during shutdown operation.

Areas that are not amenable to RBPI treatment are assessed for potential for trending at the industry level. SECY-01-0111, titled "Development of an Industry Trends Program for Operating Power Reactors" (Ref. 2) describes the intended approach for using industry trend information in regulatory applications.

Potential RBPIs were identified to address fire, shutdown, and containment. However, models and information adequate to support a complete assessment of these potential RBPIs were not available to this project during this phase. Intermediate results in these areas are presented here in order to support ongoing discussions of these areas.

1.2 Characteristics of RBPIs

As noted in the White Paper, "performance" refers to the conduct of those activities in design, procurement, construction, operation, and maintenance that support achievement of the objectives of the cornerstones of safety in the Reactor Oversight Process.

SECY 99-007, "Recommendations for Reactor Oversight Process Improvements" (Ref. 3), Attachment 2, "Technical Framework for Licensee Performance Assessment," lists the key attributes of performance within each cornerstone. RBPIs are performance measures that are logically related to the risk-significant elements of these key attributes. In this development, RBPIs are logically related to elements of risk models.

The RBPIs developed in this report collectively have the following characteristics:

- The RBPIs are compatible with, and complementary to, the risk-informed inspection activities of the oversight process.
- The RBPIs cover all modes of plant operation.
- Within each mode, the RBPIs cover risk-important systems, structures, and components (SSCs) to the extent practical.
- The RBPIs are capable of implementation without excessive burdens to licensees or NRC in the areas of data collection and quantification.
- To the extent practical, the RBPIs identify declining performance before performance becomes unacceptable, without incorrectly identifying normal variations as degradations (i.e., the RBPIs avoid false positive indications and false negative indications).
- The RBPIs are amenable to establishment of plant-specific thresholds consistent with the Reactor Oversight Process (ROP).

1.3 Organization of Report

This section is the introduction to the report. Section 2 discusses the RBPI development process in accordance with the development steps from the White Paper. Section 3 presents results of the process steps discussed in Section 2. The results are organized by internal events at power, shutdown events, and external events, because the indicators and thresholds in these areas use similar risk models and insights for each cornerstone. Within each area, each safety cornerstone is discussed. Section 4 analyzes the extent of risk coverage by the RBPIs. Section 5 discusses three aspects of validation and verification of the RBPIs. Section 6 addresses key issues affecting RBPI development and implementation.

2. PROCESS FOR IDENTIFICATION OF RBPIs

2.1 Systematic Process for RBPI Development

The steps in RBPI development are the following:

1. Assess the potential risk impact of degraded performance.
2. Obtain performance data for risk-significant, equipment-related elements.
3. Identify indicators capable of detecting performance changes in a timely manner.
4. Identify performance thresholds consistent with the graded approach to performance evaluation in SECY 99-007.

Figure 2.1 shows this RBPI development process. The following discusses each step of the flowchart in Figure 2.1.

Step 1: Assess the potential risk impact of degraded performance

The processing in this step is shown on Sheet 1 of Figure 2.1.

A performance attribute is suitable for RBPI consideration if the risk significance of changes in performance can be determined using a risk model or risk logic. An example of a performance attribute under the mitigating systems cornerstone that is typically modeled is equipment performance. Reliability and availability of mitigating systems are typically modeled, and the risk impact of performance changes in these areas can be quantified. An example of a performance attribute that is not typically modeled is "procedure quality" under the initiating events cornerstone. Some PRA models reflect procedure quality as performance shaping factors influencing human error probabilities that affect risk, but this kind of modeling is not typical in most probabilistic risk assessments (PRAs) or Individual Plant Examinations (IPEs) even for mitigating systems.

The test of risk significance of a performance attribute is whether degraded performance can cause changes in mean core damage frequency (CDF) or mean large early release frequency (LERF) that exceed 1E-6 or 1E-7, respectively. Development of RBPIs and thresholds under the initiating events cornerstone and the mitigating systems cornerstone has been carried out using CDF as the measure of risk significance. Some performance areas under these cornerstones could affect LERF, and this could affect determination of associated RBPI thresholds. Assessment of this will be completed when integrated CDF/LERF models become available to this project. Development of RBPIs and thresholds for the containment barrier under the barrier integrity cornerstone has been initiated based on assessment of published results, using LERF as the measure of risk significance. Completion of this development also requires integrated CDF/LERF models.

Figure 2.1 RBPI Development Process

Sheet 1

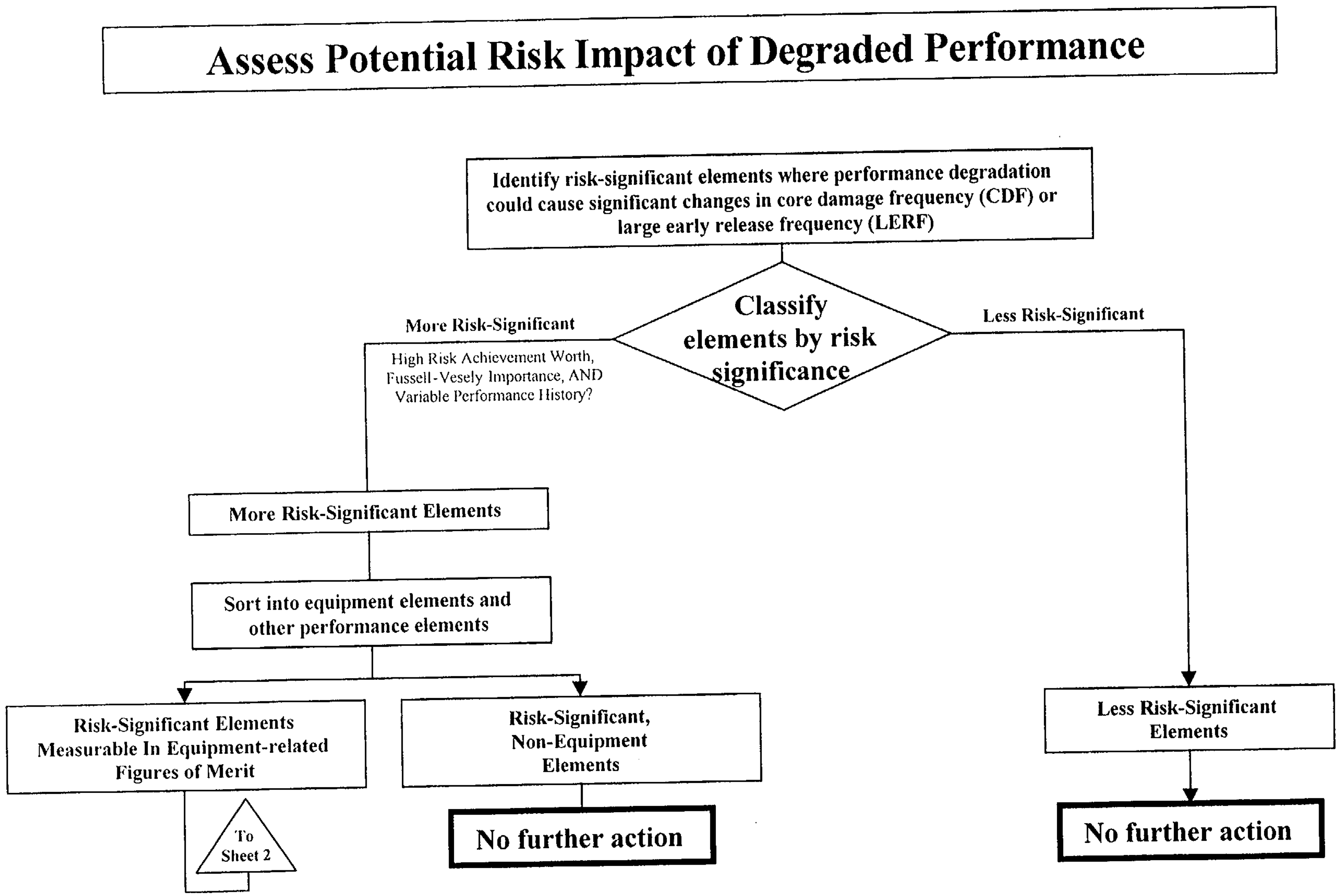

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Figure 2.1 RBPI Development Process

Sheet 2

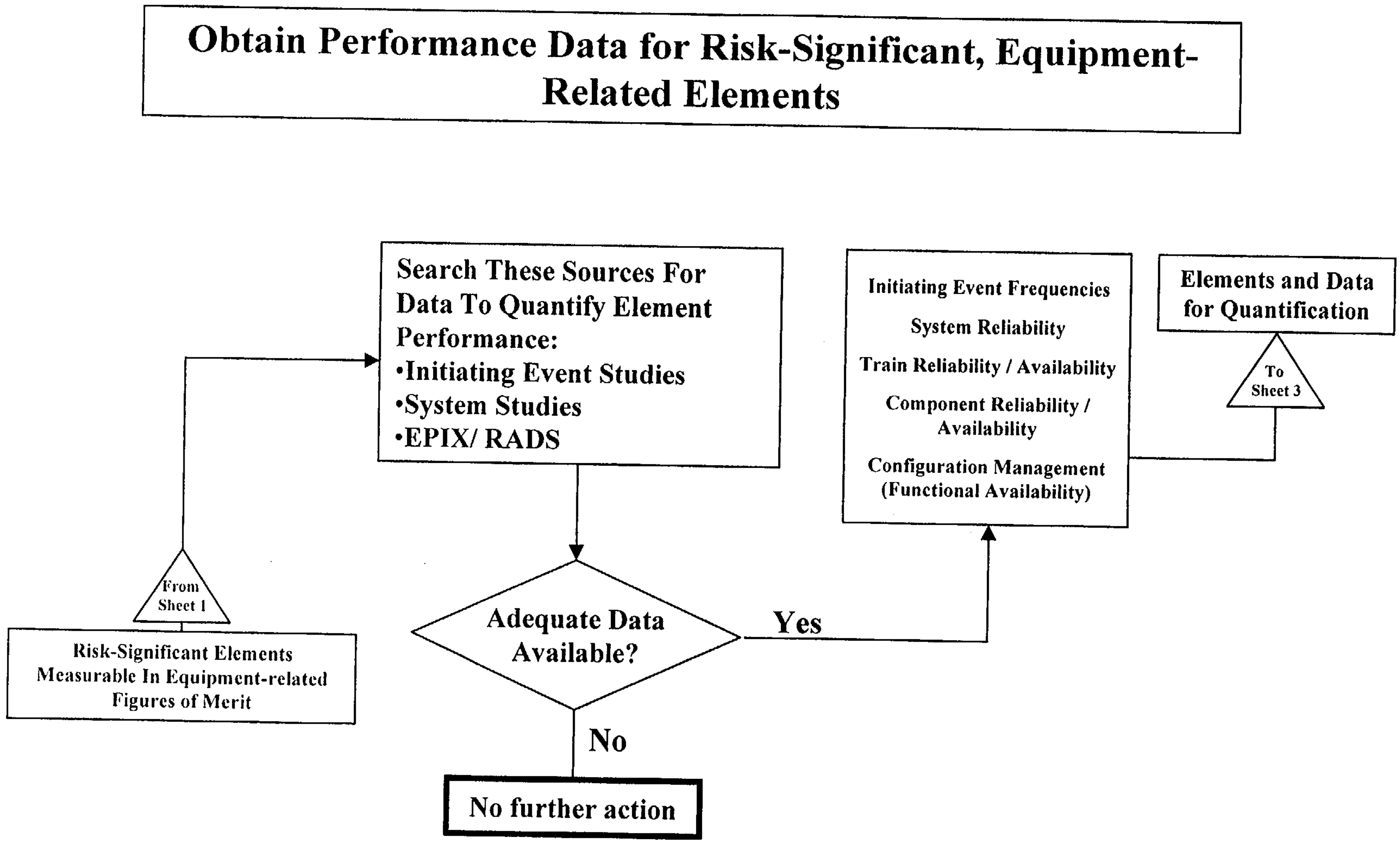


Figure 2.1 RBPI Development Process

Sheet 3

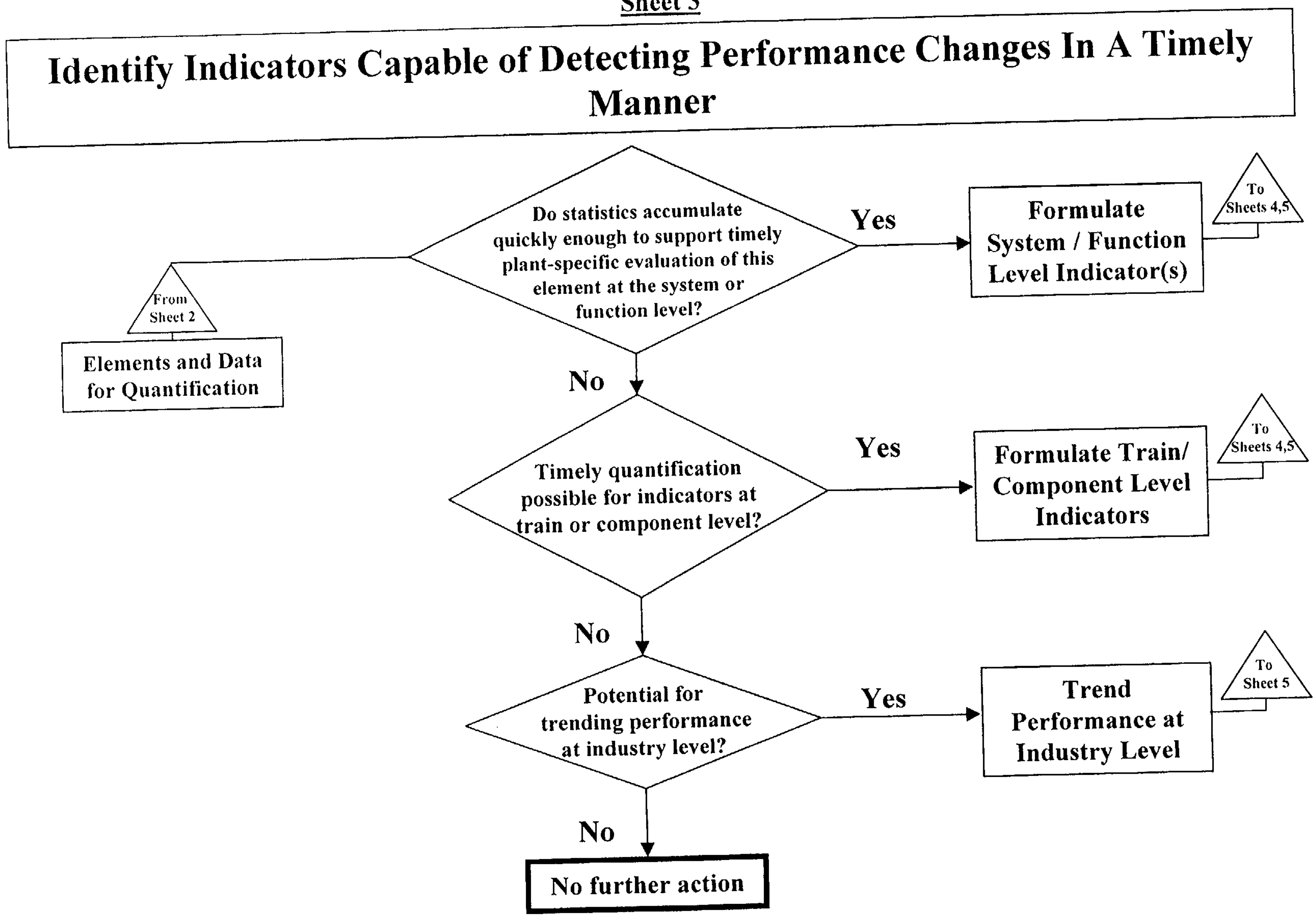


Figure 2.1 RBPI Development Process

Sheet 4

Identify Performance Thresholds Consistent With A Graded Approach To Performance Evaluation from SECY 99-007

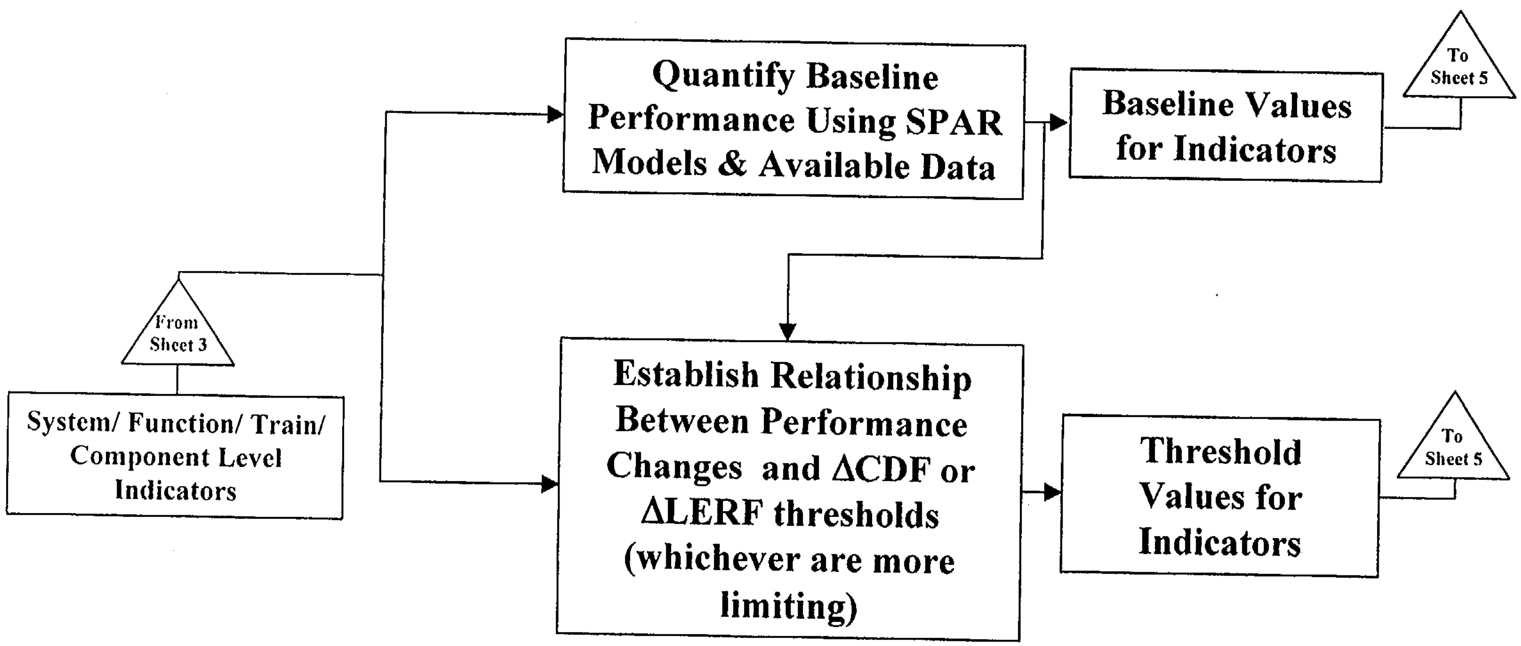
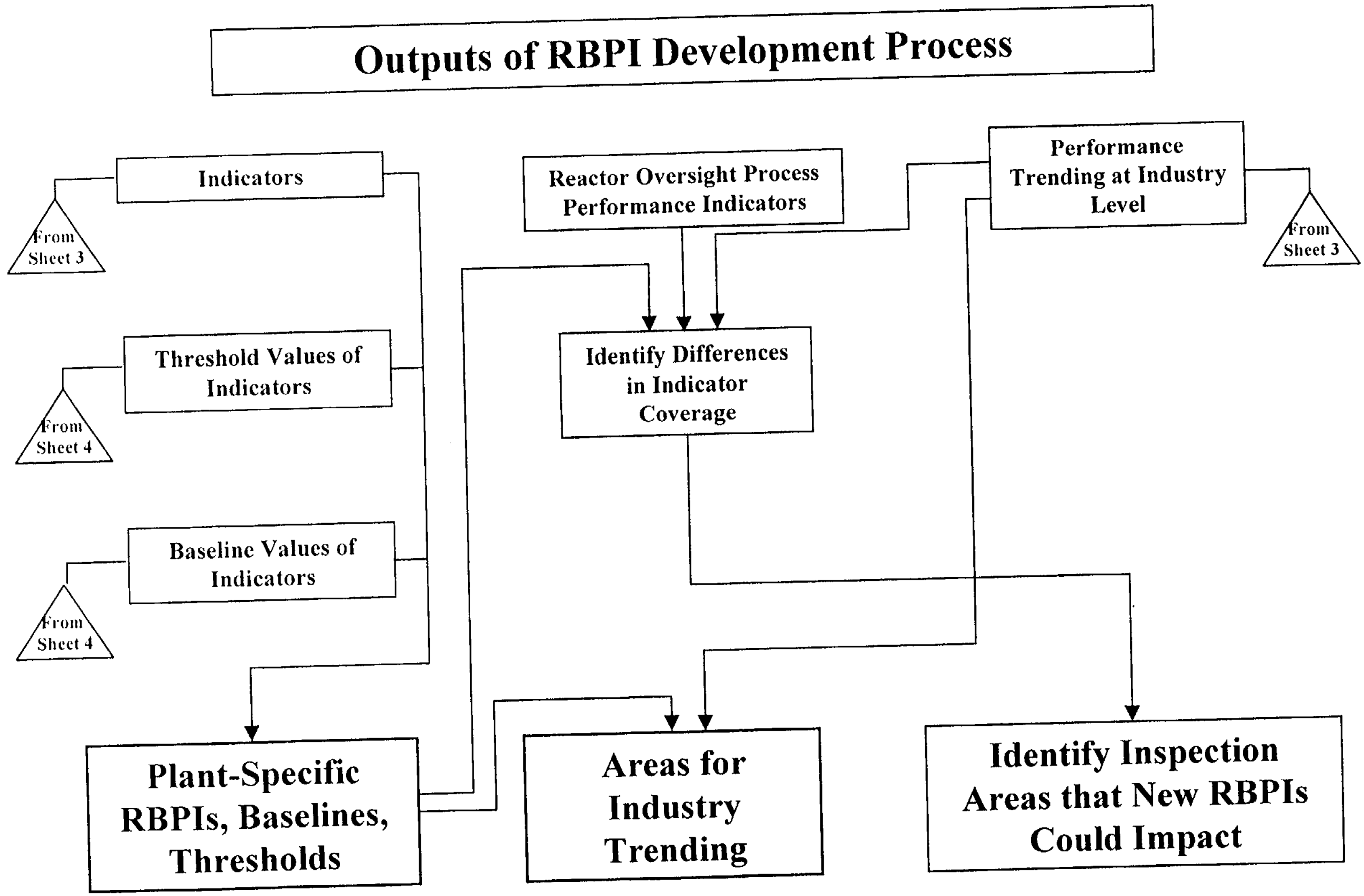


Figure 2.1 RBPI Development Process

Sheet 5



Risk significance of performance attributes can be assessed on the basis of "importance measures" such as Risk Achievement Worth (RAW), Fussell-Vesely (F-V), and Birnbaum. These are measures of the risk significance of an element. The RAW of an element indicates how risk might increase if performance degrades, while F-V indicates how much the baseline performance contributes to risk. This is discussed further in Appendix A.

Performance attributes that are not equipment-related are not within the scope of the RBPI development. Some human errors, including post-accident response, are examples of this. However, other aspects of human performance, such as the conduct of maintenance, affect equipment-related figures of merit such as unreliability and unavailability, which are within the scope of RBPI development.

The output from Step 1 is:

- Risk-significant, modeled, and equipment-related elements for which there is potential to develop RBPIs

<u>Step 2: Obtain performance data for risk-significant, equipment-related elements</u>

This step is illustrated on Sheet 2 of Figure 2.1.

The output of this step is a set of industry-wide data supporting quantification of the baseline performance of each element. For some elements, this will be unreliability/unavailability/failure frequency information, including both the number of adverse events (the numerator) and the total number of opportunities for the adverse events (operating hours, number of demands, etc.). For some elements, it may be the time spent in particular plant configurations that are important to risk.

The output from Step 2 is:

- Risk-significant, equipment-related elements and data for quantification

<u>Step 3: Identify indicators capable of detecting performance changes in a timely manner</u>

This step is shown on Sheet 3 of Figure 2.1.

This step identifies potential RBPIs to determine whether a high-level indicator (e.g., system/function performance) in each area is capable of detecting significant performance changes in a timely manner, and if not, whether a lower-level indicator (e.g., train/component performance) in that area is capable of detecting performance changes in a timely manner.

In this report, timely detection means that it is unlikely that performance degradation would not be detected using data over the most recent 3-year period. It should be noted that the data collection interval can vary depending on the type of RBPI, as explained in Section 5.3 and Appendix F of this report.

Even if a high-level indicator can detect performance changes in a timely manner, it will be necessary to disaggregate the high-level indicator into lower-level indicators if it is not possible to define an appropriate threshold value for the high-level indicator. This is discussed below under Step 4, and discussed more fully in Appendix A, Section 2.1.2.

Risk-significant areas for which it is not practical to detect plant-specific performance changes in a timely manner are considered for industry-wide trending. If performance data accumulate at the industry level quickly enough to allow trends to be identified in a given area, the area is identified for potential industry-wide trending.

The outputs of this step are the following:

- Potential RBPIs at the system/function level
- Potential RBPIs at the train/component level
- Potential industry-wide trending

Step 4: Identify performance thresholds consistent with the graded approach to performance evaluation in SECY 99-007

This step is illustrated on Sheet 4 of Figure 2.1.

The purpose of this step is to determine RBPI baseline values and the changes in each RBPI value that correspond to changes in CDF or LERF for the performance bands.

It is not possible to define an appropriate risk threshold for a high-level indicator if the risk significance of its lower level constituents differs significantly. A given net change in performance at the higher level can be caused by different sets of performance changes at lower levels having different risk impacts. This situation arises when different trains of a given system depend on different support systems, and therefore play different roles in different accident sequences. It also occurs due to the different impact of CCF on sequence quantification and the different impact of potential recovery actions on unavailability and unreliability. For these reasons, identification of thresholds for potential RBPIs above the train level needs special care. If an appropriate threshold cannot be defined, the potential RBPI must be disaggregated into lower level elements for which appropriate thresholds can be defined. This is discussed further in Appendix A, Section 2.1.2.

Some elements under the initiating events cornerstone and mitigating systems cornerstone affect LERF as well as CDF. Performance thresholds for the corresponding RBPIs need to be determined in light of both kinds of impacts. In this report, thresholds for RBPIs under the initiating events and mitigating systems cornerstones reflect only the CDF impact of performance changes. Refinement of the RBPI threshold development based on consideration of LERF as well as CDF will be undertaken in ongoing work. Complete characterization of the risk significance of such elements requires integrated models that are still being developed.

After the performance thresholds have been identified for an RBPI, the potential for false positive indications (false indications of declining performance) and false negative indications (failures to identify declining performance) is evaluated as a function of monitoring time interval. An RBPI parameter model and an associated monitoring time interval that adequately minimizes the probabilities of false indications are determined.

The outputs of this step are:

- A parameter definition for each RBPI
- A set of plant-specific threshold values for each RBPI

Outputs of RBPI Development Process

Outputs of the RBPI development process are summarized on Sheet 5 of Figure 2.1.

The content of the inspection program is related to the coverage provided by the performance indicators. This process develops some RBPIs that are different from the PIs used in the Reactor Oversight Process. Therefore, the differences are identified and summarized, and this information is evaluated with respect to its implications for the inspection program.

The following outputs of the RBPI development process are obtained:

- Plant-specific RBPI parameter definitions, baseline values, and threshold values
- Performance areas for industry-wide trending
- Inspection areas that new RBPIs could impact

2.2 Risk Perspectives Associated With RBPI Development

The RBPIs are potential improvements to the Reactor Oversight Program (ROP). They are intended to allow the NRC (and licensees) to determine when plant-specific performance in areas relating to cornerstone objectives is degrading in order to take timely corrective actions. The graded approach to regulatory response to changes in licensee performance relies on the principle that agency response is linked to the severity of the changes in performance from a risk perspective. The following discusses that principle and its relationship to the RBPI development.

There are numerous studies estimating the public risks associated with operation of nuclear power plants. These vary in scope from Level 1 estimates of core damage frequency (CDF) for internal initiators during power operations to Level 3 evaluations of an offsite dose from internal and external initiators during both power and shutdown operations. Some useful perspectives relating to public risk can be gleaned from this body of work.

General Risk Insights

Mean estimates of CDF from Individual Plant Examinations range from low E-6 to mid E-4 per year with an average in the mid E-5 range (NUREG-1560, Ref. 4). NUREG-1150 (Ref. 5) produced similar results. In addition, NUREG-1150 evaluated the probabilities of early fatalities and latent cancer fatalities for the five plants modeled. Other risk studies have done similar analyses. Using this information, it is possible to make general comparisons with the Quantitative Health Objectives of the Commission's Safety Goal Policy.

The thresholds in the ROP for Performance Indicators and the Significance Determination Process are based on changes in the CDF of approximately E-6, E-5, and E-4 per year. The lower thresholds generally represent a fraction of the currently estimated total CDF at a plant. The threshold between the yellow and red performance bands is on the same order of magnitude as our current estimates of total CDF. Thus, the ROP thresholds represent a graded approach that treats larger increases in risk with greater regulatory response.

The relationship between total CDF and the probability of early fatalities (the more limiting of the QHOs) is a function of the particular containment design and operation as well as the distribution of population around the plant and the effectiveness of emergency response capability. Using worst-case characteristics from the NUREG-1150 (Ref. 5) analysis and assuming that the baseline mean total CDF is 1E-4 per reactor year, the mean frequency of an individual early fatality is 2E-8 per year. The QHO for early fatalities is approximately 5E-7 per year (a factor of about 25 higher). This QHO is based on the Safety Goal Policy objective that early fatalities should be less than 0.1% of (three orders of magnitude, or a factor of 1000, lower than) the existing individual accidental death risk. The individual accidental death rate is approximately 5E-4 per year. Figure 2.2 displays the CDF values related to early fatality on a logarithmic scale. There are two important implications of this perspective on the development of RBPIs.

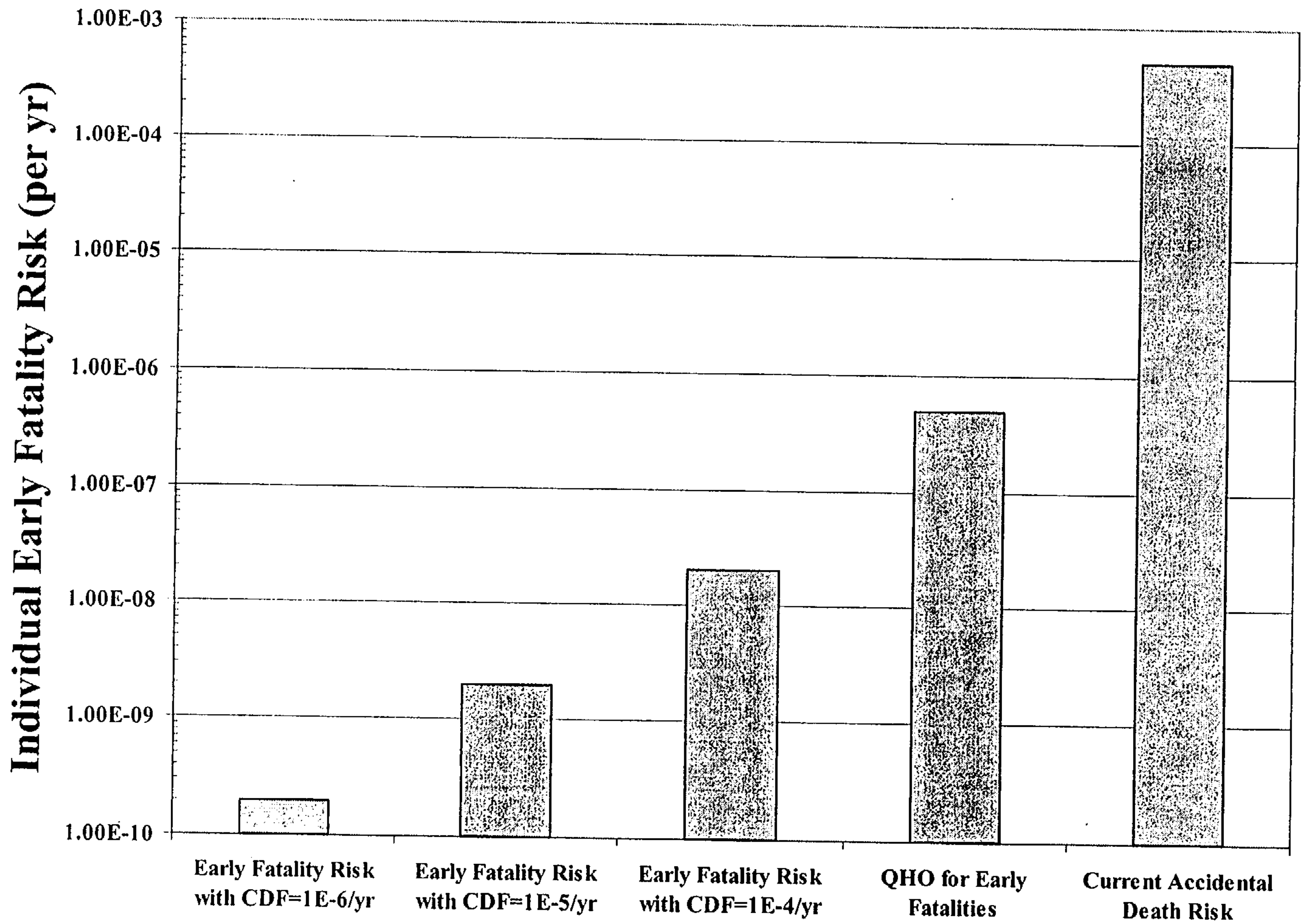
Specific Implications of General Risk Insights on RBPI Development

The first important implication is that a large margin exists between the risks associated with performance changes at the ROP thresholds and both the QHOs and the existing public risk of accidental death. This determines the precision needed to monitor performance parameters and quantify the thresholds. Errors in data, models, and/or calculations would have to be large to result in approaching either the QHO or the existing individual accidental death risk.

The second important implication deals with the ability of the RBPIs to detect potential degradations in a timely manner so that regulatory actions can be taken before the associated risk becomes too large. The ROP red performance band is the "unacceptable" performance area. It is approximately equivalent to an increase in CDF of greater than 1E-4 per year. This would increase the risk of a plant from its baseline to twice its baseline value (assuming the baseline was 1E-4). This would still be substantially below the QHO (a factor of 12 instead of 25), assuming the worst-case NUREG-1150 (Ref. 5) assumptions and still far below the existing individual accidental death risk.

An inherent implication of monitoring risk attributes is that there is a time delay between the onset of a change in performance and the ability of the indication to detect that a change has occurred. In this sense, all indicators are "lagging," or at best concurrent with, the performance change being monitored. However, in the case of RBPIs, this is not significant, because each indicator represents one of many elements of risk, for which there is still a large margin to the

Figure 2.2
Margin Between Core Damage Early Fatality Risk, QHOs, and Current Overall Accidental Death Risk



agency-stated public health objective. The issue can be addressed by the question, "If performance were to degrade instantaneously to be in the red performance band, can this change in performance be detected in time to take corrective actions before the accumulation of risk becomes unacceptable?" In answering the question, it is important to note two things.

First, the operating experience does not indicate that the unreliability of equipment or the frequency of challenges to the safety equipment is likely to change in that manner. For example, in order for the initiating events cornerstone indicator for trips with loss of heat sink to be in the red performance band, the frequency of events would have to change from about 1 every 5-10 years to more than 15 per year. There is no evidence of plants having that kind of performance. Under mitigating systems, an emergency diesel generator train unreliability change from a nominal performance of 0.04 to about 0.15 per demand is needed to be in the red performance band. NUREG/CR-5500, Vol. 5 (Ref. 6), which evaluated EDG train unreliability and trended industry unreliability performance based on actual demands (losses of normal power to buses) does not indicate any plants with a mean unreliability estimate worse than 0.066. Thus, the failure probability would have to more than double to be in the red performance band, and no plants exhibit that kind of performance degradation.

Second, even if these scenarios occurred, the RBPIs have been formulated so that the probability of the indication remaining nominal (green zone) while the performance becomes unacceptable (red zone) is low. For the two examples given above, the probability of not detecting the performance change was essentially zero for loss-of-heat-sink initiators and was less than one chance in 300 for EDG unreliability. (Plant-specific probabilities for these conditions are contained in the appendices.)

Assuming that the degradation of a single performance indicator occurred immediately following the update of the indicator for the annual performance review and was not evaluated again for a year (unlikely since the data updates are expected quarterly), there could be a maximum undetected risk addition of 1E-4 to the total CDF for 1 year. It is extremely unlikely that such performance would remain undetected beyond that time.

Performance degradations corresponding to the yellow and white zones constitute a rate of risk accumulation that is 10 to 100 times lower than this example. Thus, for these cases, the risk accumulation from "lagging" indication would be proportionately less.

In summary, the potential degradations in plant performance monitored by the ROP represent a small portion of the existing individual accidental death risk and have a substantial margin to the agency's QHOs. For the events, conditions, and equipment proposed for monitoring in the RBPIs, the likelihood of failing to detect significant degradations in performance before they pose a significant risk relative to the QHOs or the existing individual accidental death risk is small.

3. RESULTS

The results are organized by internal events at power, shutdown events, and external events, because the indicators and thresholds in each of these areas use similar risk models and insights for each cornerstone. Within each area, each safety cornerstone is discussed.

3.1 Results for Internal Events at Full Power

Risk-based Performance Indicators are chosen to reflect changes in licensee performance that are logically related to risk and associated models. They provide performance measures whose impact on core damage frequency (CDF) and large early release frequency (LERF) can be established through a risk model or risk logic. In developing RBPIs, "performance" refers to activities in design, procurement, construction, operation, and maintenance that support achievement of the objectives of the cornerstones of safety in the ROP. This section summarizes the selection and application of RBPIs at 23 plants.

3.1.1 Initiating Events Cornerstone

3.1.1.1 RBPIs

RBPIs for initiating events were determined through evaluation of the Individual Plant Examination (IPE) submittals and the associated IPE Database (Ref. 7). From this database, initiators with a conditional core damage probability (CCDP) $\geq$ 1E-6 and a contribution to industry-wide CDF $\geq$ 1% were identified as risk-significant. In accordance with the data analysis performed in NUREG/CR-5750 (Ref. 8), three schemes for grouping initiating events were considered: industry-wide, pressurized water reactors (PWRs), and boiling water reactors (BWRs). The complete list of initiators, their industry CDF contributions, and the plant group to which they belong are given in Appendix A.

The analysis of initiating event data and calculation of initiating event frequencies also relied on several data sources. The three data sources used in the selection, and their contribution to the analysis, of initiating event RBPIs are described below:

- NUREG/CR-5750, "Rates of Initiating Events at U.S. Nuclear Power Plants: 1987–1995" (Ref. 8), provided initiating event frequencies calculated for various initiators as well as the definitions of initiators and related functional impact groupings. These initiating event frequencies were incorporated into SPAR models (Ref. 9) as part of the process of establishing plant-specific baseline core damage frequencies.
- The Sequence Coding and Search System (SCSS) (Ref. 10) is a database maintained at Oak Ridge National Laboratory that provides access to electronic copies of Licensee Event Reports (LERs). This database was the source of initiating event data for NUREG/CR-5750 and will be used to identify trips and scrams used in future calculations of initiating event frequencies and RBPI thresholds.
- Monthly operating report (MOR) data as tabulated in INEEL database MORP1 (Ref. 11) provides a source of critical-operating-hour data used in the calculation of initiating event

frequencies reported in NUREG/CR-5750 and subsequently incorporated into the baseline SPAR models (Ref. 9). This database will be used to identify critical hours used in future calculations of initiating event frequencies and corresponding RBPI thresholds.

In addition to being risk-significant, initiating event performance indicators must be capable of detecting performance changes in a timely manner. The associated monitoring period must be long enough to reduce the probabilities of false negatives and false positives to acceptable levels, but no longer. Statistical analyses were performed with the results, including monitoring periods, documented in Appendices E and F. Finally, the impacts of changes in frequencies of candidate initiating events must be readily quantifiable. Plant-specific baseline models were developed by incorporating generic industry data (through 1996) into plant-specific SPAR (Revision 3i) (Ref. 9) models.

Three initiator/initiator groups (general transient, loss of feedwater, loss of heat sink) meeting the criteria were selected as candidate initiating event RBPIs. The candidate initiating event RBPIs along with representative thresholds are shown in Table 3.1.1-1. Threshold values were calculated using SPAR Revision 3i logic models. There are two thresholds indicated at the green/white interface. The 95% value represents the 95th percentile of the industry baseline values. The other is based on a ΔCDF equal to 1E-6. This value is more consistent with the current Significance Determination Process (SDP). Appendix A provides details of these calculations. This report recommends the use of the ΔCDF method for determination of the green/white interface, the rationale for which is contained in Appendix A. There are three RBPIs for each plant for the initiating events cornerstone. Detailed plant-specific threshold information for all 23 plants evaluated in this phase is contained in Appendix A. Definitions, data, and calculational procedures are provided in Appendix H.

Table 3.1.1-1 Initiating Event RBPIs

RBPIs & Example Thresholds for BWR 3/4 Plant 18					
Initiator RBPI	Baseline Frequency (NUREG/CR-5750)	Green/White 95th%ile	Green/White ΔCDF=1E-6/yr[a]	White/Yellow ΔCDF=1E-5/yr[a]	Yellow/Red ΔCDF=1E-4/yr[a]
General transient (GT)	1.2 / year[a]	2.2 / year	1.9 / year[a]	7.8 / year[a]	67 / year[a]
Loss of feedwater (LOFW)	6.8E-2 / year[a]	2.0E-1 / year	3.0E-1 / year[a]	2.5 / year[a]	24 / year[a]
Loss of heat sink (LOHS)	2.3E-1 / year[a]	3.1E-1 / year	4.1E-1 / year[a]	3.4 / year[a]	33 / year[a]
RBPIs & Example Thresholds for WE 4-Lp Plant 22					
Initiator RBPI	Baseline Frequency (NUREG/CR-5750)	Green/White 95th%ile	Green/White ΔCDF=1E-6/yr[a]	White/Yellow ΔCDF=1E-5/yr[a]	Yellow/Red ΔCDF=1E-4/yr[a]
General transient (GT)	9.6E-1 / year[a]	1.8 / year	1.8 / year[a]	8.8 / year[a]	78 / year[a]
Loss of feedwater (LOFW)	6.8E-2 / year[a]	2.0E-1 / year	8.0E-1 / year[a]	7.2 / year[a]	74 / year[a]
Loss of heat sink (LOHS)	9.6E-2 / year[a]	2.6E-1 / year	2.4E-1 / year[a]	1.5 / year[a]	15 / year[a]

a. Year refers to a calendar year assumed to include 7000 critical hours.

3.1.1.2 Industry-Wide Trending

The RBPI development program also provides industry-wide trending of the initiating events that are RBPIs as well as initiating events that are not possible to trend on a plant-specific basis. Since more data are available at the industry level, trends emerging at the industry level may be apparent before plant-specific changes can be determined. To be selected for trending, the candidate initiators must be risk-significant (i.e., contribute >1% to industry-wide CDF) and have at least one occurrence since 1987 as recorded in NUREG/CR-5750.

The loss-of-offsite-power (LOOP) initiator is an example of a performance element that is difficult to trend at a plant-specific level yet can be trended at the industry level. The IPE results indicate that LOOP is a dominant contributor to risk at U.S. nuclear power plants; however, plant-specific performance indicators are not practical because of the excessive period required to detect changes in this initiator.

Thirteen initiating event types/groups meet these conditions and are identified as candidates for industry-wide trending. These initiating event types/groups and their respective NUREG/CR-5750 category are listed below:

1. General transients (Q)
2. Loss of feedwater initiators (P1)
3. Loss of heat sink initiators (L)
4. Loss-of-offsite-power events (B1)
5. Steam generator tube rupture (F1)
6. Small/very small LOCA (G1, G3)
7. Stuck-open safety/relief valve - BWR (G2)
8. Loss of vital AC bus (C1, C2)
9. Loss of vital DC bus (C3)
10. Loss of instrument/control air - BWR (D1)
11. Loss of instrument/control air - PWR (D1)
12. Internal flood (J1)
13. Initiators evaluated as accident sequence precursors (ASP)

The process and rationale for the selection of these initiator types/groups is outlined in more detail in Appendix A. An example plot of LOOP initiating events during power operation is presented below in Figure 3.1.1-1. General transients, loss of feedwater, and loss of heat sink are trended in Table 5.3-6 of this report. Trends associated with the other initiating events are shown in Appendix A.

3.1.1.3 Inspection Areas Covered by New RBPIs

The RBPIs developed in this report for the initiating events cornerstone were compared with the performance indicators in the ROP to identify those RBPIs that are not currently in the ROP. The inspection areas that could be impacted by the new initiating event RBPIs were then determined. The results are summarized in Table 3.1.1-2.

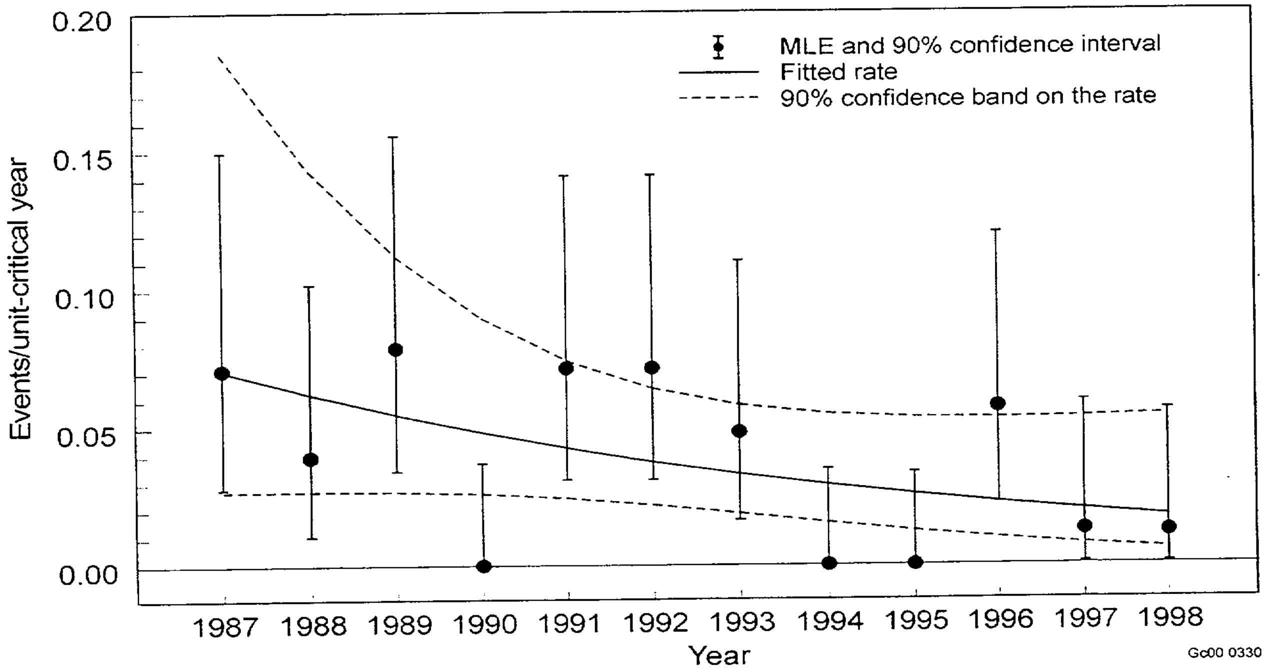


Figure 3.1.1-1 Time-Dependent Trending of Loss of Offsite Power Initiating Events

Table 3.1.1-2 Summary of Inspection Areas Impacted by New RBPIs for Initiating Events Cornerstone

RBPI	Attribute	Inspection Area
- General transient - LOFW - LOHS	- Equipment performance	71111.12, Maintenance Rule Implementation 71111.08, Inservice Inspection Activities 71152, Identification and Resolution of Problems
	- Human performance	71111.14, Personnel Performance During Nonroutine Evolutions

3.1.2 Mitigating Systems Cornerstone

This section discusses development of RBPIs that address the mitigating systems cornerstone for full-power, internal events. External events and nonpower modes are addressed in other sections.

3.1.2.1 RBPIs

The risk significance of mitigating systems was determined through analysis of Revision 3i SPAR models supplemented by quantification results found in the Individual Plant Examination (IPE) submittals and the associated IPE Database (Ref. 7). Specific equipment (i.e., mitigating systems and component classes) was identified as risk-significant based on combinations of importance measure values calculated from these sources. Plants were grouped so that a given set of RBPIs apply to the entire group based on common sets of risk-significant systems. Due to the limited number of plants for which SPAR Revision 3i models exist, two distinct plant groups were used (BWR and PWR). Additional plant groups may be developed as more SPAR Revision 3i models become available.

In addition to being risk-significant, candidate mitigating system RBPIs must be capable of detecting performance changes in a timely manner. The associated monitoring period must be long enough to reduce the probabilities of false negatives and false positives to acceptable levels, but no longer. Appendices E and F document the statistical analyses and results, including monitoring periods, of these analyses. Finally, the impacts of changes in mitigating system performance must be readily quantifiable. Plant-specific SPAR (Revision 3i) models, baselined to 1996 performance, were used to quantify the impact of these changes and to calculate corresponding threshold values.

Once risk-significant mitigating systems were identified, elements within those systems amenable to performance monitoring were selected. Two distinct elements of equipment performance, unreliability and unavailability, were selected to be monitored as RBPIs. In the RBPI development, the term "unavailability" is defined as the ratio of time when the component, train, or system was incapable of meeting its risk-significant safety function divided by the total time that ability to perform the risk-significant function could be needed. The term "unreliability" is defined as the probability that the component, train, or system would fail to perform its risk- significant safety function (fail to start or fail to run/operate) given that it was available to do so. These elements are compatible with divisions identified in SECY 99-007 and the Maintenance Rule (10 CFR 50.65 (Ref. 12) and Regulatory Guide 1.160 (Ref. 13)). These elements can be applied at any of several levels (i.e., system, train, component). The train level was determined to be the best choice and the rationale for selecting this level of monitoring is detailed in Appendix A.

The evaluation of risk-significance also identified several component classes that were important. These were chosen because they can provide plant-wide performance attributes that would potentially reflect performance changes due to "cross-cutting" issues before individual system or train indicators. Unreliability was selected to be the RBPI for each of these component classes. Component class unreliability indicators are defined in terms of failures corresponding to demands to perform the component's risk-significant safety function. Failures associated with design basis functions that do not impact the ability to perform the risk-significant function are not included in this calculation. Failures not associated with demands to perform the risk-significant function are treated through unavailability measures for the affected safety system trains.

Mitigating systems and component classes meeting the criteria were selected as candidate mitigating system RBPIs. Thirteen mitigating system/component class RBPIs were identified at each BWR plant (five in current ROP). For PWR plants, 18 mitigating system/component class RBPIs were identified (5 in the current ROP). The candidate mitigating systems and component classes are identified in Table 3.1.2-1. Examples of plant-specific thresholds are identified for two plants in Tables 3.1.2-2 and 3.1.2-3. Detailed plant-specific threshold information for all 44 plants evaluated in this phase is contained in Appendix A. Definitions, data, and calculational procedures are provided in Appendix H.

The analysis of mitigating system performance also relies on several data sources. The primary data sources used in the selection of, and their contribution to, the analysis of mitigating system RBPIs are described below:

- **System Reliability Studies** (Refs. 14-19) contain failure data for several risk-significant mitigating systems. The generic data from these studies were incorporated into the SPAR models as part of the process of establishing plant-specific 'baseline' models and associated core damage frequencies. The data currently reflected in SPAR models were derived from the original system studies; these are currently being updated. In the statistical analysis, false-positive/false-negative evaluations did not consider model uncertainty associated with the SPAR models. The model uncertainty will be addressed as part of the SPAR model verification.
- **The Reliability and Availability Database System (RADS)** (Ref. 20) will provide unreliability and unavailability data and parameter estimation capability for use in periodic evaluations of mitigating system performance. It imports data from the Institute of Nuclear Power Operations' (INPO) EPIX database (Ref. 21) as well as other established sources such as LERs and MORs.

Table 3.1.2-1 Candidate Mitigating System RBPIs

BWR RBPI SYSTEMS	RBPI Parameter and Level
Emergency AC power (EPS)	*Unreliability* and *unavailability* at the train level.
High-pressure coolant injection systems • High-pressure coolant injection (HPCI) • High-pressure core spray (HPCS)	*Unreliability* and *unavailability* at the train level.
High-pressure heat removal systems • Isolation condenser (IC) • Reactor core isolation cooling (RCIC)	*Unreliability* and *unavailability* at the train level.
Residual heat removal (SPC, RHR)	*Unreliability* and *unavailability* at the train level.
Service Water (SWS)	*Unreliability* and *unavailability* at the train level.
PWR RBPI SYSTEMS	
Auxiliary feedwater (AFW/EFW) • Motor-driven pump train • Turbine-driven pump train	*Unreliability* and *unavailability* at the train level. *Unreliability* and *unavailability* at the train level.
Component cooling water (CCW)	*Unreliability* and *unavailability* at the train level.
Emergency AC power (EPS)	*Unreliability* and *unavailability* at the train level.
High-pressure injection (HPI)	*Unreliability* and *unavailability* at the train level.
Power-operated relief valve (PORV)	*Unreliability* at the system level.
Residual/decay heat removal (RHR)	*Unreliability* and *unavailability* at the train level.
Service Water (SWS)	*Unreliability* and *unavailability* at the train level.
COMPONENT CLASSES (all plants)	
Air-operated valves (AOVs)	*Unreliability* at the component level.
Motor-operated valves (MOVs)	*Unreliability* at the component level.
Motor-driven pumps (MDPs)	*Unreliability* at the component level.

Table 3.1.2-2 BWR Mitigating System RBPIs

RBPIs & Example Thresholds for BWR 3/4 Plant 18					
Mitigating System	Baseline Train Unavailability or Unreliability[1]	Green/White 95th %ile	Green/White ΔCDF =1E-6/yr	White/Yellow ΔCDF =1E-5/yr	Yellow/Red ΔCDF =1E-4/yr
Emergency AC Power	(Unreliability) 4.0E-2	9.9E-2	4.2E-2	5.8E-2	1.5E-1
	(Unavailability) 9.7E-3	1.9E-2	1.4E-2	4.9E-2	3.9E-1
Reactor Core Isolation Cooling	(Unreliability) 7.9E-2	1.7E-1	9.1E-2	2.0E-1	Not Reached
	(Unavailability) 1.3E-2	4.0E-2	2.8E-2	1.7E-1	Not Reached
Essential Service Water	(Unreliability) 2.5E-2	8.0E-2	2.7E-2	4.2E-2	1.3E-1
	(Standby Train Unavail.) 1.9E-2	5.4E-2	2.2E-2	5.6E-2	3.9E-1
HPCI	(Unreliability) 2.4E-1	4.3E-1	2.6E-1	4.6E-1	Not Reached
	(Unavailability) 9.7E-3	3.8E-2	8.2E-2	7.3E-1	Not Reached
Residual Heat Removal	(Unreliability) 8.8E-3	2.3E-2	2.0E-2	6.8E-2	2.2E-1
	(Unavailability) 1.0E-2	2.5E-2	1.4E-1	Not Reached[2]	Not Reached[2]
AOVs	Component Class Unreliability	N/A	Increase 2.2X	Increase 13X	Increase 83X
MOVs	Component Class Unreliability	N/A	Increase 1.7X	Increase 7.0X	Increase 28X
MDPs	Component Class Unreliability	N/A	Increase 1.2X	Increase 5.1X	Increase 28X

1. Train unreliability evaluated using the plant-specific SPAR Rev. 3i system fault tree (at the train level).
2. This threshold can be reached if the test and maintenance (T&M) outages associated with this system are not confined to TS allowablecombinations.

Table 3.1.2-3 PWR Mitigating System RBPIs

RBPIs & Example Thresholds for WE 4-Lp Plant 22					
Mitigating System	Baseline Train Unavailability or Unreliability[1]	Green/White 95th %ile	Green/White ΔCDF =1E-6/yr	White/Yellow ΔCDF =1E-5/yr	Yellow/Red ΔCDF =1E-4/yr
Auxiliary Feedwater	(MDP Train Unreliability) 8.7E-3	2.1E-2	9.8E-3	1.8E-2	5.4E-2
	(TDP Train Unreliability)1.9E-1	3.4E-1	2.0E-1	2.9E-1	Not Reached
	(MDP Train Unavailability) 1.1E-3	2.5E-3	3.7E-3	2.8E-2	2.5E-1
	(TDP Train Unavailability) 4.6E-3	1.8E-2	2.1E-2	1.7E-1	Not Reached
Component Cooling Water	(Unreliability) 1.6E-2	4.7E-2	2.0E-1	6.5E-1	Not Reached
	(Standby Train Unav) 1.1E-2	4.4E-2	7.8E-1	Not Reached	Not Reached
Emergency AC Power	(Unreliability) 4.2E-2	1.0E-1	4.3E-2	5.5E-2	1.3E-1
	(Unavailability) 9.7E-3	1.9E-2	1.3E-2	3.9E-2	3.0E-1
High-Pressure Injection (Includes CVC trains)	(SI Unreliability) 9.7E-3	2.1E-2	8.8E-1	Not Reached	Not Reached
	(SI Unavailability) 4.2E-3	1.6E-2	Not Reached[2]	Not Reached[2]	Not Reached[2]
	(CVC Unreliability) 5.9E-2	1.9E-1	4.3E-1	Not Reached	Not Reached
	(CVC Standby Train Unav) 5.4E-2	1.7E-1	Not Reached[2]	Not Reached[2]	Not Reached[2]
Power-Operated Relief Valves	(System Unreliability) 3.2E-2	6.8E-2	5.7E-2	2.6E-1	Not Reached
Residual/Decay Heat Removal	(Unreliability) 1.7E-2	3.8E-2	3.8E-2	1.3E-1	4.7E-1
	(Unavailability) 7.3E-3	2.4E-2	9.3E-2	8.8E-1	Not Reached[2]
Service Water	(Unreliability) 3.2E-2	9.4E-2	1.3E-1	2.1E-1	3.2E-1
	(Standby Train Unav) 2.7E-2	9.0E-2	Not Reached[2]	Not Reached[2]	Not Reached[2]
AOVs	Component Class Unreliability	N/A	Increase 2.2X	Increase 13X	Increase 106X

Table 3.1.2-3 (Continued)

RBPIs & Example Thresholds for WE 4-Lp Plant 22					
Mitigating System	Baseline Train Unavailability or Unreliability	Green/White 95th %ile	Green/White ΔCDF =1E-6/yr	White/Yellow ΔCDF =1E-5/yr	Yellow/Red ΔCDF =1E-4/yr
MOVs	Component Class Unreliability	N/A	Increase 2.4X	Increase 11X	Increase 39X
MDPs	Component Class Unreliability	N/A	Increase 1.2X	Increase 3.2X	Increase 16X

1. Train unreliability evaluated using the plant-specific SPAR Rev. 3i system fault tree (at the train level).
2. This threshold can be reached if the T&M outages associated with this system are not confined to TS allowable combinations.

The risk significance of specific performance degradations varies from plant to plant as a result of factors such as variation in functional redundancy from plant to plant. As a result, some thresholds are not reached at a specific plant because those systems, trains, or components are less risk-significant at that plant, even though they may be more risk-significant at other plants.

3.1.2.2 Industry-Wide Trending

In addition to providing plant-specific information, the RBPI development program provides industry-wide trending, including trending on risk-significant performance elements that are not possible to trend on a plant-specific basis. Since more data are available at the industry level, trends emerging at the industry level may be apparent before plant-specific changes can be determined.

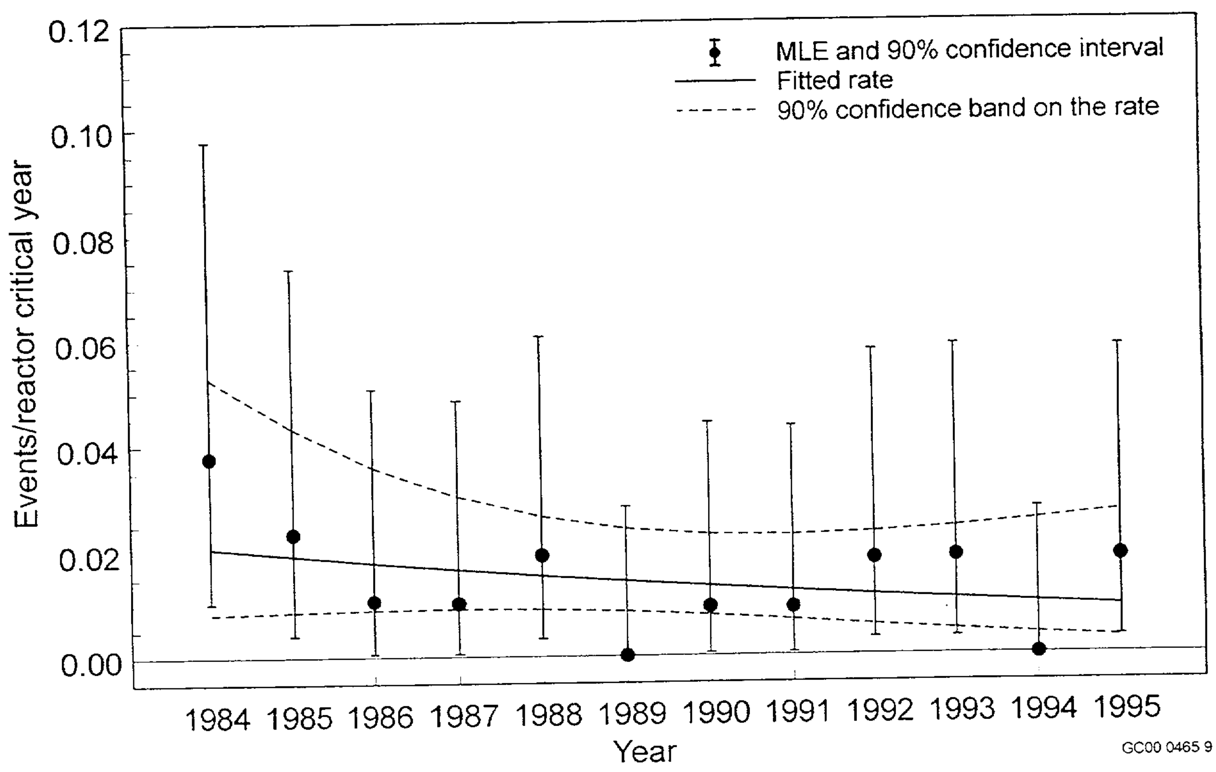


Figure 3.1.2-1 Time-Dependent Trending of Emergency Diesel Generator Common Cause Failure Events

Candidates for industry-wide trending must be risk-significant and have at least one occurrence since 1987. In addition to the RBPIs identified in Table 3.1.2-1, three mitigating systems or performance elements meet these conditions and are identified as candidates for industry-wide performance trending. Mitigating systems to be trended are:

- RBPIs from Table 3.1.2-1
- Common cause failure events for auxiliary feedwater pumps
- Common cause failure events for emergency diesel generators
- Common cause failure events for all safety-related systems

The process and rationale for the selection of the specific mitigating systems and performance elements are outlined in more detail in Appendix A.

3.1.2.3 Inspection Areas Potentially Affected by RBPIs

The RBPIs developed in this report for the mitigating systems cornerstone were compared with the performance indicators in the ROP to identify those RBPIs that are not currently in the ROP.. The inspection areas that could be impacted by the new mitigating system RBPIs were then determined. The results are summarized in Table 3.1.2-4.

Table 3.1.2-4 Summary of Inspection Areas Potentially Affected by RBPIs for Mitigating Systems Cornerstone

RBPI	Attribute	Inspection Area
Full Power: Mitigating Systems (UR)	Equipment Performance	71111.04, Equipment Alignment 71111.12, Maintenance Rule Implementation 71111.15, Operability Evaluations 71111.22, Surveillance Testing 71152, Identification and Resolution of Problems
Mitigating Systems (UA)	Equipment Performance	71111.12, Maintenance Rule Implementation
	Human Performance (Pre-Event)	71111.14, Personnel Performance During Nonroutine Evolutions 71152, Identification and Resolution of Problems
	Configuration Control	71111.04, Equipment Alignment 71111.12, Maintenance Rule Implementation 71111.13, Maintenance Risk Assessments and Emergent Work Evaluation 71111.23, Temporary Plant Modifications 71152, Identification and Resolution of Problems

3.1.3 Barrier Integrity Cornerstone: Containment Performance

This section presents RBPI development results that address the containment integrity portion of the barrier integrity cornerstone for internal events at full power. The scope of the structures, systems, and components related to the containment barrier includes the primary and secondary containment buildings (including personnel airlocks and equipment hatches), primary containment penetrations and associated isolation systems, and risk-significant systems and components necessary for containment heat removal, pressure control, and degraded core hydrogen control.

RBPI development for containment integrity uses large early release frequency (LERF) as the metric for determining the risk significance of changes in containment performance, conditional on CDF performance at the baseline value. Development of the SDP has led to classification of kinds of performance elements according to whether they affect both LERF and CDF (Type A), only LERF (Type B), or only CDF. In the terminology of the SDP, the present work on containment has examined Type B findings. Some containment-related features may affect CDF, but assessment of such Type A performance areas is not currently practical because integrated CDF/LERF models are not currently available.

For certain containment types, it has been found that the following factors influence early failure of the containment barrier. However, most of these factors affecting LERF involve mechanistic phenomena that are not amenable to RBPI development.

- Containment isolation performance
- Direct impingement of core debris on important containment elements
- Overpressure due to excessive heat loads from ATWS sequences
- RCS pressure at vessel failure
- Penetration seal integrity
- Suppression pool bypass
- Ice condenser performance
- Hydrogen ignitor performance
- Drywell spray performance

Many containment barrier mitigation systems affect late containment failure. Treatment of non-LERF risk scenarios is a topic for future discussion with stakeholders (Section 6). The following factors influence late failure of the containment barrier (Ref. 4):

- Overpressurization due to loss of containment heat removal (sprays, heat exchangers, etc.)
- Overpressurization due to core-concrete interactions
- Venting

The following potential containment RBPIs have been identified. Each potential indicator is applicable to specific containment designs:

- Unreliability/unavailability of drywell spray (Mark I BWRs, Mark II BWRs, Mark III BWRs)
- Unreliability/unavailability of large containment isolation valves (PWRs, Mark III BWRs) (valves isolating paths that connect the containment atmosphere directly to the outside atmosphere)
- Unreliability/unavailability of hydrogen ignitors (Ice condenser PWRs, Mark III BWRs)

However, for these potential RBPIs, models and data are not available for formulating baseline values and quantifying thresholds. LERF models for setting thresholds are not available for all containment types. In addition, the available models are not compatible with the SPAR Revision 3 models for assessing CDF impacts which are the inputs to the LERF models. Therefore, no containment RBPIs are provided.

Moreover, drywell spray is closely identified with Type A functionality (low-pressure injection and suppression pool cooling), so the RBPI and associated thresholds need to be defined within an integrated CDF/LERF perspective (see Section 6.1). Although containment heat removal is not generally an important contributor to LERF, in some PWRs it has a role in core damage prevention and in prevention of large early releases. This, too, is a Type A function, and needs to be examined within an integrated CDF/LERF perspective.

When better models and data are obtained, RBPI development will be completed for these potential RBPIs. In addition, RBPIs previously analyzed under the initiating events and mitigating systems cornerstones will also be reexamined to determine whether LERF considerations alter the findings of Sections 3.1.1 and 3.1.2.

These RBPIs are not among the performance indicators in the ROP. The inspection areas that could be impacted by these RBPIs were determined. The results are summarized in Table 3.1.3-1.

Table 3.1.3-1 Summary of Inspection Areas Impacted by Potential RBPIs for Containment Portion of Barrier Integrity Cornerstone

RBPI	Attribute	Inspection Area
CIV (UR&UA), Drywell Spray (UR&UA), and Hydrogen Ignitors (UR&UA)	Design Control	71111.02, Evaluation of Changes, Tests, or Experiments 71111.17, Permanent Plant Modifications 71111.23, Temporary Plant Modifications 71152, Identification and Resolution of Problems
	Barrier Performance	71111.12, Maintenance Rule Implementation 71111.15, Operability Evaluations 71111.20, Refueling and Outage Activities 71111.22, Surveillance Testing

3.2 Results for Shutdown

The results of the RBPI development process are qualitatively different from full-power results for the following reasons.

- Shutdown occupies a much smaller fraction of the year than does full-power operation, so shutdown-specific unreliability, unavailability, and frequency metrics accumulate failure data much more slowly than do comparable metrics for full power.

- Configuration management is a more significant factor in shutdown risk than in full-power risk.

- Relatively few models for shutdown CDF and LERF are available relative to full power. Therefore, the results presented below are based on risk insights from the representative models available (Refs. 22-24).

3.2.1 Initiating Events Cornerstone

No initiating events accumulate statistics quickly enough to support timely detection of declining performance. Therefore, there are no plant-specific initiating event RBPIs for shutdown operations.

However, industry trending of the following events is warranted based on existing shutdown risk studies:

- Loss of offsite power during shutdown
- Loss of operating train of RHR due either to local fault or loss of support systems
- Loss or diversion of inventory leading to loss of RHR
- Loss of level control when entering mid-loop operation leading to loss of RHR (PWR only)

3.2.2 Mitigating Systems Cornerstone

Most licensees manage shutdown risk in accordance with NUMARC 91-06, "Guidelines for Industry Actions to Assess Shutdown Management" (Ref. 26). They manage defense in depth, through configuration control, for key safety functions (decay heat removal, inventory control, electrical power availability, reactivity control, and containment). Based on available models and data, this subsection develops and illustrates RBPIs that could directly measure licensee performance in configuration control by measuring the time the plant spent in risk-significant configurations (combinations of equipment unavailabilities and plant conditions with respect to decay heat and RCS inventory).

Because the NUMARC guidance promotes avoidance of risk-significant configurations, compliance with the NUMARC guidance will promote good performance as measured by the proposed RBPIs. The NUMARC guidance addresses the following two aspects (among others): (1) hardware defense in depth, and (2) crew readiness to respond to initiating events within available time. Aspect 1 is generally captured in available models, and is reflected in the assessment of the risk significance of specific configurations. Aspect 2 is not explicitly treated in available models, which do not model a distinction between standard practice and additional compensatory operator actions resulting from adherence to 91-06.

The RBPIs reflect excess time spent in risk-significant configurations during the observation period. Four categories of configurations are defined: low, medium, early reduced-inventory (vented), and high. These are defined in terms of conditional core damage frequency (CCDF) and, in the case of the early reduced-inventory category, operational conditions. The baseline for each category (the typical time spent in configurations associated with that category) has been determined by examination of representative outage profiles, as discussed in Appendix B. Spending time over and above the baseline duration in configurations having relatively high CCDF results in core damage probability above the baseline value. The RBPI thresholds follow from the relationship:

Threshold $\Delta t = \Delta CDP$ threshold/configuration CCDF, where the ΔCDP thresholds are the standard G/W, W/Y, and Y/R thresholds (1E-6, 1E-5, and 1E-4), and the configuration CCDF corresponds to the configuration's risk category. As explained in Appendix B, all realizable configurations are classified into configuration categories, corresponding to CCDF ~ 1E-6/day (low), CCDF ~ 1E-5/day (medium), and CCDF ~ 1E-4/day (high and early reduced-inventory (vented)). Then, for example, since the medium risk configurations are associated with a CCDF of approximately 1E-5 per day, the G/W, W/Y, and Y/R thresholds for medium are, respectively, .1 day above baseline, 1 day above baseline, and 10 days above baseline. The baselines and thresholds for all three categories are shown in Tables 3.2.2-1 and 3.2.2-2, rounded in some cases to an even number of days or hours for simplicity.

Table 3.2.2-1 Baseline and Thresholds for Time in Risk-Significant Configurations Indicators - PWRs

Configuration Category	Baseline	G/W Threshold	W/Y Threshold	Y/R Threshold
Low	20 days	21 days	30 days	120 days
Medium	2 days	2 days + .08 day (2 hrs)	3 days	12 days
Early Reduced-Inventory (vented)[a]	1 day	1 day	1.08 days (1 day + 2 hrs)	2 days
High	0	0	.08 day (2 hrs)	1 day

a. This configuration category assumes that measures are taken to compensate for the risk associated with early reduced-inventory operations, as explained in Appendix B. If compensatory measures are not taken, these configurations are assigned to the high configuration category.

Table 3.2.2-2 Baseline and Thresholds for Time in Risk-Significant Configurations Indicators - BWRs

Configuration Category	Baseline	G/W Threshold	W/Y Threshold	Y/R Threshold
Low	2 days	3 days	12 days	102 days
Medium	0.20 day (5 hrs)	0.29 day (7 hrs)	1 day	10 days
High	0	0	.08 day (2 hrs)	1 day

The configurations are classified in Tables 3.2.2-3 (PWRs) and 3.2.2-4 (BWRs). As explained in Appendix B, the risk associated with these configurations has been assessed based on risk insights from representative models. Illustrative results are provided for a representative PWR (Ref. 23) and a representative BWR (Ref. 25). Risk-significant configurations are characterized by reactor coolant system (RCS) conditions, time after shutdown, and a given set of systems or trains being unavailable, either for maintenance or as a result of equipment failure. The RBPI for each configuration category is the total time spent in configurations assigned to that category during the 1-year observation period. A blank entry in a cell means that the indicated configuration in that plant operating state (POS) has a minimal conditional core damage frequency (CCDF) and time spent in that configuration need not be counted. Shaded cells indicate combinations of POS and configuration that are not analyzed, either because the configuration violates the POS definition or the systems involved play no role in the POS. The intent is that each credible plant configuration

Table 3.2.2-3 PWR Shutdown Configurations Risk Classification (Based on a Generic Westinghouse 4-Loop Shutdown PRA Model)

POS				Configuration Risk Classification															
				No Maintenance Unavailability	Backup RHR Train Unavailable	Electrical Power Trains Unavailable				Support Cooling Trains Unavailable		Secondary Cooling Trains Unavailable	Emergency Injection Trains Unavailable				Other Trains Unavailable		
Group	Mode	RCS Boundary	Days After Shutdown		RHR	One EDG	Two EDG	One Safety-Related AC Bus	One Safety-Related DC Bus	One Train of ESW	One Train of CCW	(All SGs)	RWST	Two SI[a]	Both Sumps	Two PORV	All SG and PORV	All SG and RWST	All SG Both Sumps
Low Inventory Configurations Occurring Very Early (within the first 5 days) in an Outage																			
Depressurized RHR Cooling with Reduced Inventory	Mode 5 Cold shutdown	Intact or isolatable	2	Low	Med	Low	Low	Low	Low	Low	Med	High	Low	Low	Med	Low	High	High	High
Depressurized RHR Cooling with Reduced Inventory	Mode 5 Cold shutdown	vented	< 5	ERI-V[b]	ERI-V[b]								ERI-V[b]						
Representative Configurations Occurring in a Typical Outage																			
Pressurized Cooldown	Mode 4 Hot shutdown	Intact	4			Low	Med	Low		Low			Med		Low				
Depressurized RHR Cooldown with Normal Inventory	Mode 5 Cold shutdown	Intact	8				Low	Low				Low	Low		Low	Low	High	High	High
Depressurized RHR Cooling with Reduced Inventory	Mode 5 Cold shutdown	Intact or isolatable	12		Low		Low	Low		Low	Low	Med	Low		Med	Low	High	High	High
Depressurized RHR Cooling with Reduced Inventory	Mode 5 Cold shutdown	vented	7	Med	Med	Med	Med	High	Med	Med	Med		High	Med	Med				
Depressurized RHR Cooling with Reduced Inventory	Mode 5 Cold shutdown	vented	13	Med	Med	Med	Med	High	Med	Med	Med		High	Med	Med				
Refueling Cavity Filled	Mode 6	vented	14												Med				
Low Inventory Configurations Occurring Late in a Typical Outage																			
Depressurized RHR Cooling with Reduced Inventory	Mode 5 Cold shutdown	Vented	24	Low	Med	Low	Med	Low	Low	Low	Med		High	Low	Med				

Notes: Shaded cells indicate combinations of POS and configuration that are not analyzed, either because the configuration violates the POS definition or the systems involved play no role in the POS.

Blank cells represent configurations whose CCDF < 1.0E-6 per day.

a. In this configuration it is assumed that a makeup pump is available.

b. This configuration category assumes that measures are taken to compensate for the risk associated with early reduced-inventory operations, as explained in Appendix B. If compensatory measures are not taken, these configurations are assigned to the high configuration category.

Key:

Low	Low risk configuration
Med	Medium risk configuration
High	High risk configuration
AC	Alternating current power division
CCW	Component cooling water
DC	Direct current power division
EDG	Emergency diesel generator
ERI-V	Early reduced-inventory (vented)
ESW	Emergency service water
PORV	Power-operated relief valve
RHR	Residual heat removal
RWST	Refueling water storage tank
SG	Steam generator
SI	Safety injection

Table 3.2.2-4 BWR Shutdown Configurations Risk Classification (Based on NUREG/CR-6166 Results)

POS			Configuration Risk Classification																
			No Maintenance Unavailability	Emergency AC/DC Trains Unavailable					Support Cooling Trains Unavailable			Emergency Cooling Trains Unavailable				Other Trains Unavailable			
Group	Mode	RCS Boundary		EDG I or II	EDG I and II	EDG I and III	One BAT division	Two BAT divisions	SSW A	SSW C	SSW A and C	HPCS	LPCS and HPCS	SP empty	SRVs all	SSW A and HPCS	SSW A and CDS	RHR A and all SRVs	SDC A and SP
POS 4	Hot shutdown	Intact		Low	Med	Low		High		Low	Med	Low	Low	Med	Med	Med		High	Med
POS 5	Cold shutdown	Vessel head on		Low	Med	Low	Low	High	Low	Low	Med	Low	Low	High	High	Med	Low	High	High
POS 6	Refueling	Vessel head off (level raised to steam line)												Med					Med
POS 7	Refueling	Upper pool filled									Low			Low		Low			Low

Note: Blank cells indicate combinations of POS and configuration that are not analyzed, either because the configuration violates the POS definition or the systems involved play no role in the POS.

Key:

Low	Low Risk Configuration
Med	Medium Risk Configuration
High	High Risk Configuration
EDG	Emergency diesel generator
BAT	Battery
SSW	Standby service water
HPCS	High-pressure core spray
LPCS	Low-pressure core spray
SP	Suppression pool
SRV	Safety relief valve
CDS	Condensate system
SDC	Shutdown cooling

correspond uniquely to one cell of that plant type's table, and that conditional core damage frequency (and configuration category) be implied by that cell's characteristics. Appendix H provides more detailed calculational guidance.

A significant fraction of PWR shutdown risk is associated with certain reduced-inventory operations. Because of high decay heat, early reduced-inventory operations (reduced-inventory operations conducted less than 5 days after shutdown with the RCS vented) have a CCDF that is comparable to the high CCDF configurations unless compensatory measures are taken. They are the only configurations potentially having high CCDF for which a nonzero baseline is assigned. The RBPIs allow credit for a baseline of 1 day in ERI-V conditions provided that the compensatory measures of NUMARC 91-06 are in place. Reduced-inventory operations conducted later in the shutdown may have medium CCDF, even if standby systems are nominally available. The baseline for PWRs reflects the need for PWRs to spend some time in reduced-inventory operations (including some time early in the shutdown). The balance of nominal risk from shutdown operation in PWRs derives from lower risk configurations. The threshold assignments follow directly from the calculated CCDF associated with the indicated configurations. BWR shutdown CDF is generally lower than PWR shutdown CDF; therefore, the baseline values are different.

The proposed RBPIs somewhat resemble condition assessments in that they are capable of manifesting a significant change in risk over a relatively short time. Changes in plant configuration at shutdown induce potentially large changes in conditional CDF that persist over short times (hours, days, or weeks), rather than moderate changes persisting over longer times (greater than 1 year) as is typical of full-power PIs. The development of the shutdown RBPIs has been carried out in such a way that the risk impact of shutdown performance changes is mapped into performance bands consistently with the full-power development.

Internal and external stakeholder comments indicated that the approach presented above for potential shutdown RBPIs is more appropriate for potential application in the SDP process of the ROP. Consequently, future use of this effort will concentrate on evaluating the significance of shutdown conditions for the SDP.

The inspection areas that could be impacted by the new RBPIs were determined. The results are summarized below in Table 3.2.2-5.

Table 3.2.2-5 Summary of Inspection Areas Impacted by Potential Shutdown RBPIs for Mitigating Systems Cornerstone

RBPI	Attribute	Inspection Area
Time in High/Medium/Low/ERI-V Configurations	Configuration Control	71111.04, Equipment Alignment 71111.13, Maintenance Risk Assessments and Emergent Work Evaluation 71111.20, Refueling and Outage Activities 71111.23, Temporary Plant Modifications

3.2.3 Barrier Integrity Cornerstone: Containment Integrity at Shutdown

Containment performance at shutdown is affected by one issue that does not pertain to full-power RBPIs, namely, that containment may be open during shutdown and must be reclosed expeditiously under certain conditions.

PWRs

The analysis documented in NUREG-1449 (Ref. 27) shows that timely closure of PWR containment prevents large early release in core damage scenarios initiated at shutdown.

BWRs With Mark I and Mark II Containments

The analysis in NUREG-1449 shows that BWR secondary containment alone is not expected to prevent large early release in core damage scenarios. This means that a change in BWR Mark I and II shutdown CDF equates to a change in LERF if primary containment is open. This circumstance is offset by generally lower shutdown CDFs for BWRs.

BWRs With Mark III Containments

The analysis documented in NUREG/CR-6143 (Ref. 24) shows that timely closure of these BWR containments prevents large early release in core damage scenarios initiated at shutdown.

These results suggest possible containment RBPIs analogous to the possible time-in-risk-significant-configurations RBPIs defined above in 3.2.2. These would be defined for the risk-significant configuration categories introduced for the RBPIs defined for mitigating systems.

Potential RBPI for PWRs and Mark III BWRs

Time spent in risk-significant configurations with containment not closed and preparations for timely closure not complete (with "timely" defined as before boiling if the RCS is vented).

Potential RBPI for Mark I and Mark II BWRs

Time spent in risk-significant configurations with primary containment not intact and not capable of timely closure.

An increase in time spent in a particular configuration with containment not capable of timely closure implies an increase in LERF that is equal to the increase in CDF associated with that configuration. Configurations with negligible conditional CDF are therefore associated with negligible changes in LERF, but risk-significant configurations contribute directly and significantly to LERF if containment is open and timely closure is not provided for.

Configurations in which only a short time is available to respond to initiating events are also generally those in which only a short time is available to effect containment closure.

NUMARC 91-06 emphasizes (p. 32) the need to maintain capability for closure of containment in a time commensurate with plant conditions, with due regard for the availability of power (needed for closure) and the potential for adverse environmental conditions. It states (p. 19) that "containment hatches ... should either be closed or capable of being closed prior to core boiling following a loss of DHR ...." Core boiling is not the same as core damage, but since personnel are expected to evacuate when environmental conditions become adverse (p. 32), it is possible that operator actions in containment will become problematic as boiling occurs.

Based on the above, time spent in risk-significant configurations with containment not capable of timely closure would be an indicator that comports with NUMARC 91-06 guidance. However, its detailed formulation would need to reflect the availability of needed auxiliaries, the absence of physical obstacles to containment closure, and the readiness and availability of personnel to effect closure when it is warranted. Data and models are not presently available to quantify these indicators. Therefore, neither baselines nor thresholds can be quantified. Quantification of these indicators would require, in addition to the time spent in risk-significant configurations, the time spent with containment in the indicated state during those risk-significant configurations.

3.3 Results for External Events (Fire)

This section provides preliminary RBPI results for fire. Other external events, such as seismic and flood, are not included in the scope of Phase 1 RBPI development.

The results from the Individual Plant Examinations for External Events (IPEEEs) were used to assess the risk-significant performance attributes in accordance with the RBPI development process flowchart shown in Figure 2.1. In addition, the Fire Protection Risk Significance Screening Methodology, used in the current fire significance determination process (Ref. 29), was reviewed to supplement the insights of the IPEEE information. The IPEEE results are not collated in as comprehensive a way as was done for the IPE program, although draft NUREG-1742 (Ref. 30) does provide a comprehensive summary of the perspectives gleaned from the technical reviews of the IPEEE submittals. These studies indicate that fire CDF varies significantly among plants. However, fire CDF is generally high enough that some elements of fire scenarios are risk-significant compared to risk associated with shutdown or full-power internal events. Specifically, NUREG-1742 states that "... the CDFs from accidents initiated by fires are of the same order of magnitude as those from other random internal events for the industry taken as a whole."

The elements of fire-initiated core damage sequences include the following:

- Occurrence of fire in specific fire area
- Failure of detection/suppression (automatic and/or manual) systems
- Fire damage to plant systems/cables in the fire area
- Fire barrier/separation effectiveness

- Failure of post-fire safe shutdown systems (typically normal mitigation systems that are not affected by the fire scenario, covered in Section 3.1.2).

Fire occurrence, including conditions leading up to the fire, is within the scope of the initiating events cornerstone. The remaining elements are within the scope of the mitigating systems cornerstone.

3.3.1 Initiating Events Cornerstone

No RBPIs are identified under the initiating events cornerstone for fire because the occurrence of fire events is too infrequent to support timely quantification of changes in plant-specific fire frequency. Based on an NRC study of fire events from 1986 to 1994 (Ref. 28), the fire initiating event frequencies for these areas range from 6.9E-2/year to 8.5E-4/year. These frequencies (once every 14 years or longer on a plant-specific basis) do not allow for timely quantification of changes in the fire frequencies. The risk-significant fire areas vary from plant to plant. However, the following fire areas are the most common among the list of risk-significant fire areas based on the accident sequences identified in the IPEEE for each plant:

- Switchgear room
- Control room
- Cable spreading room
- Auxiliary building (PWR)/reactor building (BWR)
- Turbine building
- Battery room
- Cable vault/tunnel/chase zones
- Diesel generator rooms

However, the occurrence rate of fire events in these areas is sufficient for industry-wide trending. The frequencies of occurrence of fire events in the most commonly risk-significant fire areas listed above will be used for industry-wide trending.

3.3.2 Mitigating Systems Cornerstone

The RBPI development identified fire suppression system unreliability and unavailability as potential RBPIs. The risk significance of fire suppression is highly plant-specific and area-specific, but at many plants, the risk significance of fire suppression is such that performance degradation in fire suppression could cause changes in CDF that are significant compared to the performance thresholds. Monitoring of suppression system unreliability and unavailability could provide feasible plant-specific RBPIs. However, although automatic suppression system unreliability and unavailability may in fact be risk-significant, the models and data currently available are not amenable for use in determining thresholds. Credit for compensatory actions would strongly affect RBPI thresholds for fire detection and suppression systems. Unfortunately, compensatory actions are not modeled well enough in available models to enable their use for threshold determination in the RBPI program.

3.3.3 Barrier Integrity Cornerstone: Containment Performance

The IPEEEs typically only provide a qualitative analysis of barrier integrity, with the general conclusion that the results of the IPE analysis are unchanged as a result of the fire scenarios. Consideration of fire does not lead to any risk-significant LERF scenarios whose containment barrier attributes are not already being addressed under the internal events treatment of the containment barrier.

4. ASSESSMENT OF RISK COVERAGE BY RBPIs

The purpose of this section is to show the extent of risk coverage by RBPIs associated with core damage sequences, to show which risk-significant contributors are not covered by RBPIs, and to indicate briefly why these contributors are not covered by RBPIs.

How Coverage Is Assessed

Two approaches to assessment of the extent of RBPI coverage of core damage frequency have been applied.

One approach is based on risk achievement worth (RAW), which measures how quickly CDF increases if element performance degrades. Given the baseline CDF and the RAW associated with a given element, the magnitude of the CDF increment that could be caused by degradation of the element can be determined. This is done for all basic events appearing in the SPAR model, and the extent of RBPI coverage is then assessed for each basic event whose failure could cause a CDF change greater than 1.0E-6. This assessment is closely related to the method for selecting candidate RBPIs in the first place (Section 3).

In addition, an assessment of RBPI coverage of dominant accident sequences (sequences whose frequency contributes most to overall CDF) was performed. Dominant accident sequences are examined to determine which contributors to risk are covered by an RBPI. This is similar to a Fussell-Vesely importance evaluation.

Results of Coverage Assessment

Table 4-1 shows results for two plants, designated BWR 3/4 Plant 18 and Westinghouse four-loop Plant 22 (WE 4-Lp) for the RAW importance-based assessment of coverage, derived from SPAR models for these plants. For those events whose failure could lead to an increase in CDF > 1.0E-6/y, approximately 40% of the events in the SPAR models are part of the RBPIs. The types of elements in the other 60% are indicated in Table 4-1.

Table 4-1 Coverage of Risk-Significant Core Damage Elements from SPAR Models

Category	BWR 3/4 Plant 18	WE 4-Lp Plant 22
Total number of SPAR model elements whose failure can result in $\Delta CDF \geq 1E\text{-}6/y$	178	203
- Initiating events	14	14
- Mitigating system elements	164	189
Elements covered by RBPIs		
- Initiating events	3/14 (21%)	3/14 (21%)
- Initiating events covered by trending	3/14 (21%)	4/14 (29%)
- Mitigating system elements	70/164 (43%)	72/189 (38%)

Table 4-1 (Continued)

Category	BWR 3/4 Plant 18	WE 4-Lp Plant 22
Types of elements not explicitly covered by RBPIs	Batteries Check valves Electrical buses Heat exchangers Post-event human errors Reactor protection system Strainers Tanks	Batteries Check valves Electrical buses Heat exchangers Post-event human errors Reactor protection system Strainers Fans

Tables 4-2a and b show RBPI coverage of dominant accident sequences at the initiating event/system level for the same two plants. The tables are derived from the IPE data base results for these plants. Almost all sequences are covered in part by multiple RBPIs. Most of the elements that are not covered either are not amenable to RBPI treatment or appear in sequences that contribute a relatively small fraction of core damage frequency. Some normally operating systems are credited for plant-specific reasons and do not appear in enough plant PRAs to have justified generically applicable RBPIs.

Figures 4-1a and b show RBPI coverage of initiating events for BWR 3/4 Plant 18 and Westinghouse four-loop Plant 22, based on relative contribution to core damage frequency (full-power internal events), derived from the IPE data base for these plants. Similar results for other plants are provided in Appendix D.

Many initiating events occur too infrequently to permit timely quantification of declining performance, and RBPIs based on frequency of occurrence of individual initiating events in this category are therefore not defined. However, as discussed in Section 3.1.1, initiating events contributing more than 1% on average to industry-wide CDF and which includes one or more occurrences (industry-wide) since 1987 are included in the industry-wide trending.

Elements Not Covered by RBPIs

Five initiating events from the IPE data base information in Tables 4-2a and b were not covered by either RBPIs (indicators of event frequency) or trended initiators. Tables 4-2a and b, prepared using the IPE data base format, display ATWS events as if ATWS were an initiator. ATWS as such is not covered by an RBPI, but initiating events potentially leading to ATWS are covered as shown. Medium and large LOCA initiators are not covered because of their very low frequencies. Certain support systems whose loss is an initiating event are monitored under the mitigating systems cornerstone (service water and component cooling water in PWRs). Although no RBPI directly monitors the frequency of total loss of these systems, the corresponding initiating events are implicitly monitored at a lower level (the train level rather than the system level).

Table 4-2a RBPI Coverage of Dominant Full-Power Internal Event Core Damage Sequences - BWR 3/4 Plant 18 (IPE Data Base Results)

IE RBPI

Industry-Wide Trending

System RBPI

SEQ	CDF	INITIATOR	ACCIDENT SEQUENCE FAILURES				
1	5.28E-07	T-LOOP	AC	EAC			
2	1.60E-07	S1	HUM				
3	2.70E-08	T-LOOP	HP1	HUM	AC		
4	2.21E-08	T-LOOP	AC	EAC			
5	2.05E-08	T-ATWS	RPS	CONDA	HUM		
6	1.80E-08	T-LOOP	HPCI(HPCS)	RCIC	AC	EAC	
7	1.34E-08	T-LOOP	HP1	HUM	AC		
8	1.16E-08	T-RX	ADS	DC			
9	1.10E-08	T-LOOP	HPCI(HPCS)	RCIC	HP1	HUM	AC
10	8.96E-09	T-LOOP	HP1	LPCI	SPC	AC	
11	8.12E-09	T-RX	DC				
12	7.76E-09	T-ATWS	RPS	LPCI	CS	CONDA	HUM
13	7.59E-09	T-LOOP	SPC	HUM	AC		
14	7.00E-09	T-LOOP	HP1	SPC	HUM	AC	
15	6.90E-09	T-LOOP	HP1	SPC	HUM	AC	
16	6.72E-09	T-LOOP	HP1	HUM	AC		
17	6.13E-09	T-ATWS	RPS	CONDA	HUM		
18	5.83E-09	T-ATWS	RPS	CONDA	HUM		
19	5.77E-09	T-LOOP	HPCI(HPCS)	RCIC	HP1	HUM	AC
20	5.66E-09	A	LPCI	CS			
21	5.53E-09	T-LOOP	HPCI(HPCS)	RCIC	HUM	AC	
22	5.43E-09	T-LOOP	HPCI(HPCS)	RCIC	HP1	HUM	AC
23	5.10E-09	T-RX	HPCI(HPCS)	RCIC	HP1	HUM	
24	5.02E-09	S2	HPCI(HPCS)	HUM			
25	4.60E-09	A	SPC	AC			
26	4.46E-09	T-LOOP	HP1	LPCI	SPC	AC	
27	4.44E-09	T-LOOP	LPCI	SPC	HUM	AC	
28	3.88E-09	T-ATWS	RPS	HP1	CONDA	HUM	
29	3.83E-09	S1	HPCI(HPCS)	HUM			
30	3.78E-09	T-LOOP	SPC	HUM	AC		
31	3.62E-09	T-ATWS	RPS	HPCI(HPCS)	CONDA	HUM	
32	3.46E-09	T-LOOP	HP1	HUM	AC		
33	3.42E-09	T-LOOP	SPC	HUM	AC		
34	3.38E-09	T-RX	HPCI(HPCS)	RCIC	MFW	HP1	HUM

Table 4-2a (Continued)

IE RBPI

Industry-Wide Trending

System RBPI

SEQ	CDF	INITIATOR	ACCIDENT SEQUENCE FAILURES				
35	3.33E-09	T-LOOP	LPCI	SPC	HUM	AC	
36	3.33E-09	T-LOOP	HP1	HUM	AC		
37	2.86E-09	T-LOOP	HPCI(HPCS)	RCIC	HP1	HUM	AC
38	2.77E-09	T-LOOP	LPCI	SPC	HUM	AC	
39	2.63E-09	T-LOOP	HPCI(HPCS)	RCIC	HUM	AC	
40	2.57E-09	T-RX	HPCI(HPCS)	RCIC	HUM		
41	2.57E-09	A	SPC				
42	2.42E-09	T-LOOP	HP1	LPCI	SPC	HUM	AC
43	2.40E-09	T-LOOP	HUM	AC			
44	2.26E-09	T-LOOP	HP1	HUM	AC		
45	2.21E-09	T-ATWS	RPS	CONDA	HUM		
46	2.16E-09	S2	HPCI(HPCS)	MFW	HUM		
47	2.15E-09	T-LOOP	HPCI(HPCS)	HP1	AC	EAC	
48	2.10E-09	A	HUM				
49	2.08E-09	T-RX	HPCI(HPCS)	RCIC	MFW	HP1	HUM
50	2.05E-09	T-LOOP	HP1	HUM	AC		
51	1.97E-09	T-LOOP	HP1	LPCI	SPC	AC	
52	1.96E-09	T-LOOP	HP1	LPCI	SPC	AC	
53	1.90E-09	T-LOOP	HUM	AC			
54	1.89E-09	T-LOOP	HP1	SPC	HUM	AC	
55	1.82E-09	T-ATWS	RPS	SLC	CONDA		
56	1.79E-09	T-LOOP	HP1	SPC	AC		
57	1.74E-09	T-ATWS	RPS	MFW	CONDA	HUM	
58	1.72E-09	T-LOOP	HP1	SPC	HUM	AC	
59	1.70E-09	T-RX	HPCI(HPCS)	RCIC	MFW	HUM	
60	1.66E-09	T-LOOP	HP1	LPCI	SPC	HUM	AC
61	1.62E-09	T-ATWS	RPS	RECIRC	CONDA		
62	1.50E-09	T-LOOP	HP1	SPC	HUM	AC	
63	1.43E-09	T-ATWS	RPS	MFW	HUM		
64	1.39E-09	A	HUM				
65	1.38E-09	T-RX	HPCI(HPCS)	RCIC	MFW	HP1	HUM
66	1.33E-09	T-LOOP	HP1	HUM	AC		
67	1.19E-09	T-ATWS	RPS	HUM			
68	1.15E-09	T-LOOP	HP1	LPCI	SPC	VENT	AC
69	1.14E-09	T-LOOP	HUM	AC			

Table 4-2a (Continued)

IE RBPI

Industry-Wide Trending

System RBPI

SEQ	CDF	INITIATOR	ACCIDENT SEQUENCE FAILURES				
70	1.13E-09	T-LOOP	HPCI(HPCS)	RCIC	HP1	HUM	AC
71	1.13E-09	A	LPCI	CS	DC		
72	1.13E-09	T-LOOP	HP1	HUM	AC		
73	1.12E-09	T-LOOP	HPCI(HPCS)	RCIC	HP1	HUM	AC
74	1.10E-09	T-LOOP	HP1	SPC	HUM	AC	
75	1.10E-09	T-ATWS	RPS	MFW	HP1	CONDA	HUM
76	1.09E-09	T-LOOP	HP1	HUM	AC		
77	1.05E-09	T-RX	HPCI(HPCS)	RCIC	MFW	HUM	
78	1.03E-09	T-LOOP	HP1	SPC	HUM	AC	
79	1.03E-09	T-ATWS	RPS	HPCI(HPCS)	MFW	CONDA	HUM
80	1.03E-09	T-LOOP	HP1	HUM	AC	NSW	
81	1.02E-09	T-LOOP	HP1	HUM	AC		
82	1.01E-09	T-LOOP	HP1	AC			
83	9.90E-10	T-LOOP	HP1	LPCI	SPC	AC	
84	9.80E-10	T-LOOP	HP1	SPC	AC		
85	9.75E-10	T-LOOP	HP1	LPCI	SPC	AC	
86	9.53E-10	S2	HPCI(HPCS)	MFW	HUM		
87	9.41E-10	T-LOOP	HP1	SPC	DWS	HUM	AC
88	9.41E-10	T-LOOP	HP1	HUM	AC		
89	9.18E-10	T-LOOP	HPCI(HPCS)	RCIC	HP1	HUM	AC
90	9.15E-10	A	SPC	AC			
91	9.03E-10	T-LOOP	HUM	AC			
92	8.85E-10	T-LOOP	HP1	SPC	DWS	AC	
93	8.62E-10	T-ATWS	RPS	CONDA	HUM		
94	8.50E-10	T-LOOP	HP1	SPC	HUM	AC	
95	8.16E-10	T-LOOP	HP1	LPCI	CS	AC	
96	8.00E-10	T-LOOP	AC	EAC			
97	7.93E-10	T-LOOP	LPCI	CS	HUM	AC	
98	7.88E-10	T-LOOP	HPCI(HPCS)	RCIC	HP1	HUM	AC
99	7.55E-10	A	SPC				
100	7.28E-10	T-LOOP	HP1	HUM	AC		
101	1.52E-07	REMAINDER					
102		T-IFL					

Table 4-2b RBPI Coverage of Dominant Full-Power Internal Event Core Damage Sequences WE 4-Lp Plant 22 (IPE Data Base Results)

IE RBPI

Industry-Wide Trending

System RBPI

SEQ	CDF	INITIATOR	ACCIDENT SEQUENCE FAILURES		
1	2.14E-05	T-CCW	HUM	CCW	
2	1.27E-05	S2	HUM		
3	5.99E-06	T-CCW	HUM	CCW	
4	3.98E-06	T-AC	SDAFW	HVAC1	
5	3.26E-06	S2	HUM		
6	2.88E-06	T-SGTR	SGS	HUM	
7	2.56E-06	T-CCW	HUM		
8	2.38E-06	T-AC	ESW		
9	2.12E-06	T-CCW	HUM	CCW	
10	1.90E-06	T-AC	HUM	HVAC1	
11	1.80E-06	T-AC	ESW		
12	1.77E-06	T-AC	HUM	CCW	
13	1.69E-06	T-CCW	HUM	CCW	
14	1.30E-06	S1	HUM		
15	1.29E-06	T-CCW	HUM	CCW	
16	1.22E-06	T-DC	MDAFW	SDAFW	HUM
17	1.16E-06	T-AC	AC	EAC	
18	1.14E-06	T-CCW	HUM	CCW	
19	1.07E-06	T-IFL	ESW		
20	1.06E-06	T-IFL	ESW		
21	9.84E-07	T-CCW	HUM	CCW	
22	9.59E-07	T-LOOP	AC	ESW	
23	9.51E-07	T-ESW	ESW		
24	8.94E-07	T-AC	AC	EAC	
25	8.61E-07	T-RX	ESW		
26	8.50E-07	S2			
27	8.46E-07	S2			
28	7.78E-07	T-TT	ESW		
29	7.70E-07	S2	HUM		
30	7.37E-07	T-DC	MDAFW	SDAFW	HUM
31	7.19E-07	T-CCW	HUM	CCW	
32	5.96E-07	T-AC	HVAC1		
33	5.95E-07	T-CCW	HUM	CCW	
34	5.93E-07	T-LMFW	ESW		

Table 4-2b (Continued)

IE RBPI

Industry-Wide Trending

System RBPI

SEQ	CDF	INITIATOR	ACCIDENT SEQUENCE FAILURES			
35	5.56E-07	T-AC	CCW			
36	5.42E-07	T-AC	ESW			
37	5.39E-07	T-LOOP	AC	EAC		
38	5.34E-07	T-AC	HUM	CCW		
39	5.13E-07	T-LOOP	AC	EAC		
40	5.10E-07	A	ACC			
41	4.99E-07	T-LOOP	SDAFW	HVAC1		
42	4.85E-07	T-SGTR	LPR	HUM		
43	4.84E-07	T-TT	RPS	PPORV	MDAFW	SDAFW	HUM
44	4.77E-07	T-IFL	HVAC1			
45	4.75E-07	T-CCW	HUM	CCW		
46	4.75E-07	T-CCW	HUM	CCW		
47	4.73E-07	T-CCW	HUM	CCW		
48	4.52E-07	T-IFL	CCW			
49	4.32E-07	S2				
50	4.27E-07	T-RX	HVAC1			
51	4.25E-07	T-LOOP	AC	EAC		
52	4.05E-07	A				
53	3.86E-07	T-TT	CCW			
54	3.66E-07	S1	HUM			
55	3.64E-07	T-LOOP	SDAFW	HVAC1		
56	3.62E-07	T-CCW	HUM	CCW		
57	3.58E-07	T-IFL	CCW			
58	3.53E-07	T-MSIV	SDAFW	HVAC1		
59	3.47E-07	T-AC	HUM			
60	3.44E-07	T-RX	HUM	HVAC1		
61	3.42E-07	T-RX	HUM	HVAC1		
62	3.41E-07	T-SGTR	LPR	HUM		
63	3.39E-07	T-CCW	HUM	CCW		
64	3.23E-07	T-SGTR	LPR	HUM		
65	3.21E-07	T-IFL	SDAFW	HVAC1		
66	3.14E-07	T-SGTR	HUM			
67	3.13E-07	T-RX	CCW			
68	3.12E-07	T-LMFW	RPS	PPORV	HUM	
69	3.11E-07	T-TT	HUM	HVAC1		

Table 4-2b (Continued)

IE RBPI

Industry-Wide Trending

System RBPI

SEQ	CDF	INITIATOR	ACCIDENT SEQUENCE FAILURES		
70	3.09E-07	T-TT	HUM	HVAC1	
71	3.08E-07	T-TT	HUM	CCW	
72	3.06E-07	T-CCW	HUM	CCW	
73	2.94E-07	T-LMFW	ESW		
74	2.85E-07	T-CCW	HUM	CCW	
75	2.83E-07	T-TT	ESW		
76	2.79E-07	T-TT	HUM	CCW	
77	2.76E-07	T-CCW	HUM	CCW	
78	2.73E-07	T-LOOP	ESW		
79	2.68E-07	T-CCW	HUM	CCW	
80	2.63E-07	T-CCW	HUM	CCW	
81	2.63E-07	T-CCW	HUM	CCW	
82	2.56E-07	T-VAC	MDAFW	HUM	
83	2.52E-07	T-DC	MDAFW	SDAFW	HUM
84	2.40E-07	T-MSIV	HUM	HVAC1	
85	2.39E-07	T-AC	AC	EAC	
86	2.37E-07	T-LMFW	RPS	PPORV	
87	2.37E-07	T-LMFW	HUM	HVAC1	
88	2.35E-07	T-LMFW	HUM	HVAC1	
89	2.35E-07	T-CCW	HUM	CCW	
90	2.33E-07	T-SGTR	HUM		
91	2.31E-07	S2	HUM		
92	2.31E-07	S2	HUM		
93	2.31E-07	T-CCW	HUM	CCW	
94	2.31E-07	T-CCW	HUM	CCW	
95	2.28E-07	T-TT	RPS	PPORV	HUM
96	2.27E-07	T-LOOP	ESW		
97	2.25E-07	T-LOOP	ESW		
98	2.24E-07	S2	HUM		
99	2.24E-07	T-CCW	HUM	CCW	
100	2.23E-07	S2	HUM		
102	6.08E-05	REMAINDER			
101	3.06E-06	T-IFL			

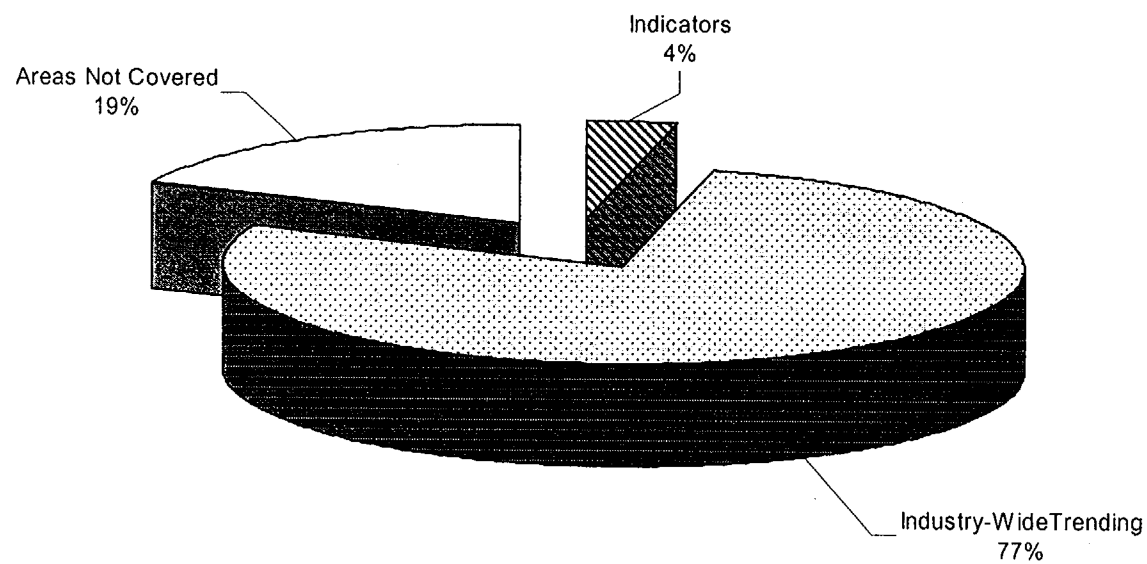


Figure 4-1a RBPI Coverage of Dominant Full-Power Internal Event Core Damage Sequences by Initiating Events for BWR 3/4 Plant 18

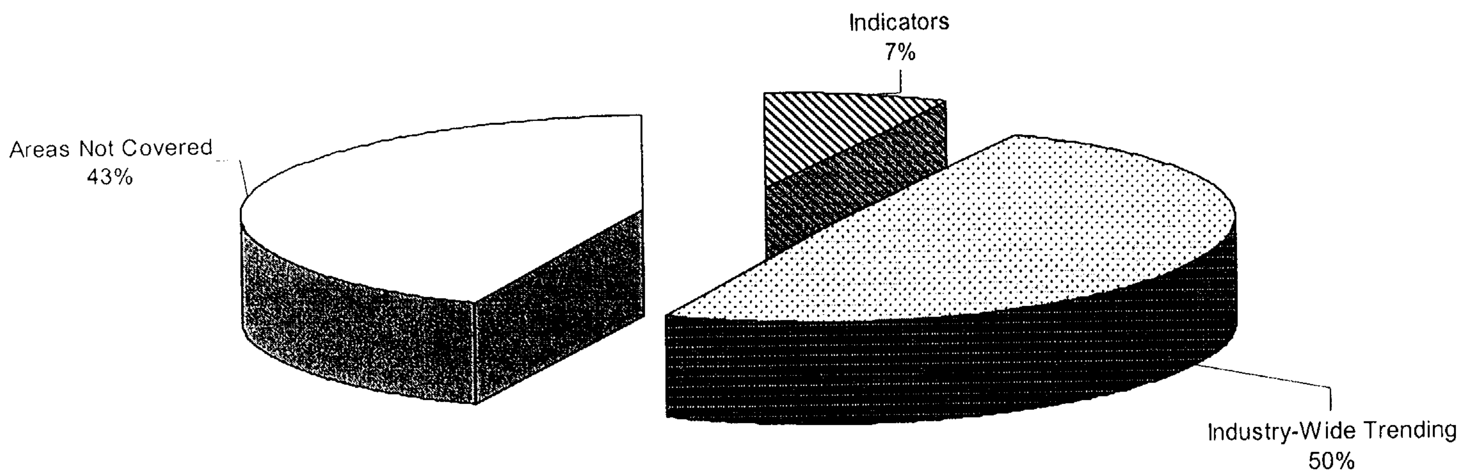


Figure 4-1b RBPI Coverage of Dominant Full-Power Internal Event Core Damage Sequences by Initiating Events for WE 4-Lp Plant 22

Table 4-3 lists mitigating system elements in Tables 4-2a and b that are not covered by RBPIs, with an explanation.

Table 4-3 Mitigating System Elements That Appear in Dominant Core Damage Sequences but Are Not Covered by RBPIs

WE 4-Lp Plant 22	
Element	**Reason for No RBPI**
Post-Accident Human Action	Not amenable to PI treatment (timely quantification directly from performance data not practical)
Steam Generator Safety Valves	Not amenable to PI treatment (timely quantification directly from performance data not practical)
Non-Safety AC Power System	Not amenable to PI treatment (timely quantification directly from performance data not practical)
Heating/Ventilation/Air Conditioning	Loss of HVAC with support systems available is not risk-significant at most plants
Reactor Protection System	Not amenable to PI treatment (timely quantification directly from performance data not practical)
Safety Injection System Accumulators	Not amenable to PI treatment (timely quantification directly from performance data not practical)
BWR 3/4 Plant 18	
Element	**Reason for No RBPI**
Post-Accident Human Action	Not amenable to PI treatment (timely quantification directly from performance data not practical)
Reactor Protection System	Not amenable to PI treatment (timely quantification directly from performance data not practical)
Non-Safety AC Power System	Not amenable to PI treatment (timely quantification directly from performance data not practical)
Automatic Depressurization	Risk-significant performance degradation of ADS valves is unlikely
Safe Shutdown Makeup Pump	Not generically important
Low-Pressure Coolant Injection	Most hardware shared with Suppression Pool Cooling, which is covered by an RBPI
Main Feedwater	This area is covered by an RBPI under the IE cornerstone but appears here as a system mitigating a reactor trip initiator. For that specific function the data would not accumulate quickly enough to support RBPI quantification.
Non-Safety DC	Not amenable to PI treatment (timely quantification directly from performance data not practical)
Drywell Spray	Most hardware shared with Suppression Pool Cooling, which is covered by an RBPI
Venting	Not amenable to PI treatment (timely quantification directly from performance data not practical)

5. VALIDATION AND VERIFICATION

The White Paper discusses two steps of validation and verification (V&V): step 1 activities undertaken as part of the development and testing of RBPIs, and step 2 activities that are an ongoing and integral part of the reactor oversight inspection process. The step 1 V&V presented in this report covers the following:

- Process for RBPI identification
- RBPI characteristics
- Testing of RBPIs

5.1 Development of a Systematic Process for RBPI Identification

Sections 2 and 3 of this report describe the process and results for identifying RBPIs. The process for identifying RBPIs is both risk-based and systematic, as indicated by the flowchart in Section 2 of this report. Potential RBPIs are identified and then compared with various selection criteria to determine whether the RBPIs can be developed. Results for full-power internal events from this systematic process are presented in Appendix A for the 30 sites (44 plants) used in the V&V testing activity.

5.2 Assurance That RBPIs Satisfy Specific Characteristics

Section 1.2 of this report lists six characteristics that RBPIs should have. Each of those characteristics is discussed below:

- RBPIs should be compatible with, and complementary to, the risk-informed inspection activities of the Reactor Oversight Process (ROP).

  The RBPI identification process (flowchart in Section 2 of this report) ensures that RBPIs are both compatible with and complementary to inspection activities. Potential RBPIs are identified using a process similar to that used for the ROP. RBPIs are compared to existing ROP indicators and the potentially affected inspection areas are identified.

- RBPIs should cover all modes of plant operation.

  The RBPIs developed in this report cover both full-power and shutdown modes of plant operation.

- RBPIs should cover risk-important SSCs to the extent practical.

  Risk coverage is discussed in Section 4 of this report. The RBPI development process ensures that as much of the risk as feasible is covered by the RBPIs.

- RBPIs should be capable of implementation without excessive burdens on licensees or NRC in the areas of data collection and quantification.

  Most of the RBPIs identified in this report can be quantified using existing databases as indicated in Section 5.3. This report identifies potential RBPIs that would require additional data collection effort, for example, the time spent in risk-significant configurations during shutdown operations and the unreliability and unavailability of containment barrier systems and fire suppression systems. Quantification of RBPI values for the 44 plants covered in the V&V testing activity and comparison with plant-specific thresholds to determine plant performance (Section 5.3 of this report) requires NRC resources, but this process is expected to be automated to the extent possible as the RBPI development effort continues.

- To the extent practical, RBPIs should identify declining performance before performance becomes unacceptable, without incorrectly identifying normal variations as degradations (i.e., avoid false-positive indications and false-negative indications).

  The suggested misclassification probability criteria are discussed in Appendix F. In general, the RBPIs selected have acceptable false-negative probabilities (less than 5% chance of obtaining a green RBPI indication when performance is actually in the red performance band). Most of the RBPIs also have acceptable false-positive probabilities (less than 20% chance of obtaining a white RBPI indication when performance is actually at the RBPI's baseline level). However, many of the unreliability RBPIs have a significant chance of obtaining a white RBPI indication when performance is actually at the RBPI's baseline level. Therefore, for all unreliability RBPIs, when white band performance is indicated, the probability of performance actually being at the RBPI's baseline value will also be presented. More details can be found in Appendices E and F.

- The RBPIs should be amenable to establishment of plant-specific thresholds consistent with the ROP.

  For the RBPIs presented in this report, plant-specific thresholds were developed using the SPAR Revision 3i core damage frequency models. Results are described in Appendix A of this report.

5.3 Testing of the RBPIs for Practicality of Calculation and Credibility of Results

The RBPIs for internal events while plants are at power were tested by evaluating plant-specific data from 44 plants over the period 1997–1999. Baseline SPAR models including industry-average values reflecting 1996 performance were used. The data collection effort to test the RBPIs at 44 plants was accomplished using INPO's EPIX database for unreliability (with RADS as the search and quantification software package), ROP data for unavailability, and NUREG/CR-5750 for initiating events. The overall data collection process was straightforward, although there are areas where data are not presently available (indicated in the tables).

Definitions, data, and calculational procedures are provided in Appendix H. The constrained, noninformative prior and the recommended data collection intervals were used (1 year for the general transient (GT) initiating event and mitigating system unavailabilities, and 3 years for the loss of heat sink (LOHS) and loss of feedwater (LOFW) initiators, mitigating system unreliabilities, and component class unreliabilities). This prior and the data collection intervals were identified from the statistical analyses (Appendix F) as most appropriate for the RBPIs being tested. Results are presented in Tables 5.3-1 through 5.3-4 for the initiating event, mitigating system unavailability, mitigating system unreliability, and component class RBPIs. For the RBPIs with available data from 1997–1999, approximately 94% of the RBPIs indicate green plant performance, with the other 6% indicating white or yellow performance.

Table 5.3-1 Plant Performance Bands for Initiating Event RBPIs (1999)[a, e]

	1999		
Plant	GT[b]	LOHS[c]	LOFW[c,d]
BWRs			
BWR 123 Plant 1	3.2E-1/y (G)	9.0E-2/y (G)	5.2E-2/y (G)
BWR 123 Plant 2	9.6E-1/y (G)	4.7E-1/y (W)	5.3E-2/y (G)
BWR 3/4 Plant 1	2.2E+0/y (G)	2.6E-1/y (G)	5.1E-2/y (G)
BWR 3/4 Plant 2	3.0E-1/y (G)	8.7E-2/y (G)	5.2E-2/y (G)
BWR 3/4 Plant 3	1.5E+0/y (G)	8.7E-2/y (G)	5.1E-2/y (G)
BWR 3/4 Plant 4	2.3E+0/y (W)	8.9E-2/y (G)	5.2E-2/y (G)
BWR 3/4 Plant 5	3.0E-1/y (G)	9.2E-2/y (G)	5.3E-2/y (G)
BWR 3/4 Plant 6	3.4E-1/y (G)	9.1E-2/y (G)	5.2E-2/y (G)
BWR 3/4 Plant 8	1.6E+0/y (G)	9.0E-2/y (G)	5.2E-2/y (G)
BWR 3/4 Plant 11	3.3E-1/y (G)	9.2E-2/y (G)	5.2E-2/y (G)
BWR 3/4 Plant 12	9.1E-1/y (G)	2.6E-1/y (W)	5.2E-2/y (W)
BWR 3/4 Plant 13	9.7E-1/y (G)	8.8E-2/y (G)	5.1E-2/y (W)
BWR 3/4 Plant 15	9.1E-1/y (G)	8.6E-2/y (G)	5.1E-2/y (G)
BWR 3/4 Plant 16	3.2E-1/y (G)	8.8E-2/y (G)	5.2E-2/y (G)
BWR 3/4 Plant 18	9.4E-1/y (G)	9.8E-2/y (G)	5.5E-2/y (G)
BWR 3/4 Plant 19	3.0E-1/y (G)	1.1E-1/y (G)	5.8E-2/y (G)
BWR 5/6 Plant 2	3.5E-1/y (G)	2.7E-1/y (G)	5.1E-2/y (G)
BWR 5/6 Plant 8	3.9E-1/y (G)	1.0E-1/y (G)	5.4E-2/y (W)
PWRs			
B&W Plant 3	2.9E-1/y (G)	5.8E-2/y (G)	5.2E-2/y (G)
B&W Plant 4	1.6E+0/y (W)	6.3E-2/y (G)	5.5E-2/y (G)
B&W Plant 5	2.8E+0/y (Y)	1.8E-1/y (W)	5.3E-2/y (G)
B&W Plant 6	2.8E-1/y (G)	6.0E-2/y (G)	5.4E-2/y (G)
B&W Plant 7	3.0E-1/y (G)	5.8E-2/y (G)	5.2E-2/y (G)

Table 5.3-1 (Continued)

Plant	GT	LOHS	LOFW
PWRs			
CE Plant 1	3.2E-1/y (G)	5.9E-2/y (G)	5.2E-2/y (G)
CE Plant 2	8.8E-1/y (G)	2.9E-1/y (W)	5.2E-2/y (G)
CE Plant 3	3.2E-1/y (G)	5.9E-2/y (G)	5.2E-2/y (G)
CE Plant 4	3.0E-1/y (G)	5.9E-2/y (G)	5.2E-2/y (G)
CE Plant 5	1.2E+0/y (G)	8.4E-2/y (G)	No data (G)
CE Plant 10	3.1E-1/y (G)	6.0E-2/y (G)	5.3E-2/y (G)
CE Plant 11	9.2E-1/y (G)	1.8E-1/y (W)	5.3E-2/y (G)
CE Plant 12	2.1E+0/y (W)	9.0E-2/y (G)	1.6E-1/y (G)
WE 2-Lp Plant 5	3.1E-1/y (G)	1.8E-1/y (W)	5.3E-2/y (G)
WE 2-Lp Plant 6	2.8E-1/y (G)	5.9E-2/y (G)	5.4E-2/y (G)
WE 3-Lp Plant 5	2.0E+0/y (W)	5.8E-2/y (G)	5.3E-2/y (G)
WE 3-Lp Plant 10	2.8E-1/y (G)	5.9E-2/y (G)	5.3E-2/y (G)
WE 3-Lp Plant 11	9.3E-1/y (G)	5.7E-2/y (G)	5.1E-2/y (G)
WE 4-Lp Plant 1	2.8E-1/y (G)	5.9E-2/y (G)	5.3E-2/y (G)
WE 4-Lp Plant 2	2.1E+0/y (W)	5.8E-2/y (G)	5.2E-2/y (G)
WE 4-Lp Plant 22	2.8E-1/y (G)	5.8E-2/y (G)	1.6E-1/y (G)
WE 4-Lp Plant 23	2.9E-1/y (G)	5.7E-2/y (G)	1.5E-1/y (G)
WE 4-Lp Plant 28	3.1E-1/y (G)	5.8E-2/y (G)	1.6E-1/y (G)

a. Plant performance bands are the following: green (G) - ΔCDF < 1.0E-6/y, white (W) - 1.0E-6/y < ΔCDF <1 .0E-5/y, yellow (Y) - 1.0E-5/y < ΔCDF <1 .0E-4/y, red (R) - ΔCDF > 1.0E-4/y.

b. A 1-year data collection interval applies (1999). The 1999 data were obtained from the ROP.

c. A 3-year data collection interval applies (1997–1999). The 1997 and 1998 data were obtained from the initiating events study update, while the 1999 data were obtained from the ROP.

d. This RBPI is not covered under the ROP, so the results presented in this table include only 1997 and 1998. (1999 licensee event reports will need to be reviewed to identify scrams that are LOFW, as defined in the initiating events study.)

e. Since the models and data in these tables have not completed formal peer review, plant-specific inferences regarding green or nongreen performance from these calculations would be inappropriate.

Table 5.3-2 Plant Performance Bands for Mitigating System Unavailability RBPIs (1999)[b]

Plant	EPS	HPI/ HPCI/ HPCS	AFW/ RCIC	RHR	SWS [a]	CCW [a]	PORV [a]
BWRs							
BWR 123 Plant 1	1.7E-2 (G)	1.4E-2 (G)	NA	6.2E-2 (G)	No data	NA	NA
BWR 123 Plant 2	1.5E-2 (G)	1.7E-2 (G)	NA	9.5E-3 (G)	No data	NA	NA
BWR 3/4 Plant 1	6.1E-2 (G)	1.4E-2 (G)	1.9E-2 (G)	7.4E-2 (G)	No data	NA	NA
BWR 3/4 Plant 2	6.1E-2 (G)	8.4E-3 (G)	3.7E-3 (G)	3.3E-2 (G)	No data	NA	NA
BWR 3/4 Plant 3	1.5E-2 (G)	4.1E-3 (G)	3.5E-3 (W)	1.6E-2 (G)	No data	NA	NA
BWR 3/4 Plant 4	1.5E-2 (G)	6.8E-3 (G)	2.1E-2 (G)	2.4E-2 (G)	No data	NA	NA
BWR 3/4 Plant 5	2.9E-3 (G)	2.4E-3 (G)	5.5E-3 (G)	0.0E+0 (G)	No data	NA	NA
BWR 3/4 Plant 6	1.3E-2 (G)	2.1E-3 (G)	1.0E-2 (G)	8.4E-3 (G)	No data	NA	NA
BWR 3/4 Plant 8	1.9E-2 (G)	2.8E-2 (G)	5.0E-2 (G)	7.8E-3 (G)	No data	NA	NA
BWR 3/4 Plant 11	7.4E-3 (G)	1.8E-2 (G)	1.8E-2 (W)	1.2E-2 (G)	No data	NA	NA
BWR 3/4 Plant 12	7.9E-2 (W)	8.2E-2 (G)	1.8E-2 (G)	1.0E-2 (G)	No data	NA	NA
BWR 3/4 Plant 13	7.1E-2 (W)	1.4E-2 (G)	1.5E-2 (G)	6.5E-3 (G)	No data	NA	NA
BWR 3/4 Plant 15	1.5E-2 (G)	1.6E-2 (G)	8.6E-3 (G)	9.1E-3 (G)	No data	NA	NA
BWR 3/4 Plant 16	2.2E-2 (G)	2.1E-2 (G)	7.9E-3 (G)	1.3E-2 (G)	No data	NA	NA
BWR 3/4 Plant 18	2.1E-2 (W)	4.5E-1 (W)	1.7E-2 (G)	5.4E-3 (G)	No data	NA	NA
BWR 3/4 Plant 19	1.8E-2 (W)	1.7E-2 (G)	1.8E-2 (G)	7.5E-3 (G)	No data	NA	NA
BWR 5/6 Plant 2	3.6E-2 (W)	4.6E-3 (G)	1.5E-2 (G)	4.4E-3 (G)	No data	NA	NA
BWR 5/6 Plant 8	5.7E-3 (G)	1.7E-2 (G)	1.7E-2 (G)	1.4E-2 (G)	No data	NA	NA
PWRs							
B&W Plant 3	2.3E-2 (G)	3.8E-3 (G)	MDP (No data) TDP (7.8E-3) (G)	9.1E-3 (G)	No data	No data	No data
B&W Plant 4	2.3E-2 (G)	5.3E-3 (G)	MDP (4.0E-3) (G) TDP (0.0E+0) (G)	1.8E-2 (G)	No data	No data	NA
B&W Plant 5	2.4E-2 (G)	3.0E-3(G)	MDP (3.3E-3) (G) TDP (3.1E-3) (G)	1.4E-2 (G)	No data	No data	NA
B&W Plant 6	2.2E-2 (G)	2.5E-3 (G)	MDP (6.8E-3) (G) TDP (8.9E-4) (G)	1.1E-2 (G)	No data	No data	NA
B&W Plant 7	2.8E-2 (G)	1.1E-2 (G)	MDP (6.6E-3) (G) TDP (1.5E-3) (G)	7.2E-2 (W)	No data	No data	No data
CE Plant 1	4.0E-3 (G)	1.8E-5 (G)	MDP (4.7E-3) (G) TDP (6.7E-4) (G)	9.5E-3 (G)	No data	No data	No data
CE Plant 2	6.6E-3 (G)	7.2E-3 (G)	MDP (0.0E+0) (G) TDP (2.9E-3) (G)	1.0E-2 (G)	No data	No data	No data

Table 5.3-2 (Continued)

Plant	EPS	HPI/ HPCI/ HPCS	AFW/ RCIC	RHR	SWS[a]	CCW[a]	PORV[a]
PWRs							
CE Plant 3	7.5E-3 (G)	1.1E-2 (G)	MDP (2.4E-3) (G) TDP (4.5E-3) (G)	1.4E-2 (G)	No data	No data	No data
CE Plant 4	9.5E-3 (G)	1.3E-3 (G)	MDP (9.8E-4) (G) TDP (6.2E-3) (G)	2.1E-3 (G)	No data	No data	No data
CE Plant 5	1.1E-2 (G)	8.3E-3 (G)	MDP (4.9E-3) (W) TDP (6.4E-3) (G)	4.1E-3 (G)	No data	No data	No data
CE Plant 10	3.7E-2 (G)	3.6E-3 (G)	MDP (2.2E-2) (W) TDP (8.2E-3 (G)	1.9E-2 (G)	No data	No data	No data
CE Plant 11	2.1E-2 (G)	1.3E-2 (G)	MDP (2.2E-2) (W) TDP (1.0E-2) (G)	4.0E-3 (G)	No data	No data	No data
CE Plant 12	5.1E-3 (G)	7.3E-3 (G)	MDP (5.3E-3) (W) TDP (4.6E-3) (G)	7.1E-3 (G)	NA	No data	No data
WE 2-Lp Plant 5	1.3E-2 (G)	1.4E-3 (G)	MDP (4.4E-3) (G) TDP (6.7E-3) (G)	1.6E-2 (G)	No data	No data	No data
WE 2-Lp Plant 6	1.0E-2 (G)	1.2E-3 (G)	MDP (4.2E-3) (G) TDP (2.5E-3) (G)	2.6E-3 (G)	No data	No data	No data
WE 3-Lp Plant 5	1.5E-2 (G)	1.6E-2 (G)	MDP (3.2E-3) (G) TDP (1.3E-3) (G)	5.9E-3 (G)	No data	No data	No data
WE 3-Lp Plant 10	5.2E-2 (G)	1.6E-3 (G)	MDP (4.9E-3) (G) TDP (1.9E-3) (G)	0.0E+0 (G)	No data	No data	No data
WE 3-Lp Plant 11	4.5E-2 (G)	7.8E-4 (G)	MDP (5.5E-3) (G) TDP (5.3E-3) (G)	2.1E-3 (G)	No data	No data	No data
WE 4-Lp Plant 1	3.5E-3 (G)	SI 1.1E-3 (G) CVC 5.4E-3 (G)	MDP (3.4E-3) (G) TDP (4.3E-2) (Y)	9.1E-5 (G)	No data	No data	No data
WE 4-Lp Plant 2	3.3E-3 (G)	SI 8.5E-3 (G) CVC 2.1E-2 (G)	MDP (2.4E-3) (G) TDP (1.1E-2) (G)	8.0E-3 (G)	No data	No data	No data
WE 4-Lp Plant 10	No data	No data	MDP (No data) TDP (No data)	No data	NA	No data	No data
WE 4-Lp Plant 11	No data	No data	MDP (No data) TDP (No data)	No data	No data	No data	No data
WE 4-Lp Plant 22	9.6E-3 (G)	SI 7.7E-3 (G) CVC 4.5E-2 (G)	MDP (3.8E-3) (W) TDP (1.2E-2) (G)	4.4E-3 (G)	No data	No data	No data
WE 4-Lp Plant 23	1.2E-2 (G)	SI 4.9E-3 (G) CVC 5.1E-3 (G)	MDP (6.6E-3) (W) TDP (1.7E-2) (G)	8.2E-3 (G)	No data	No data	No data

Table 5.3-2 (Continued)

Plant	EPS	HPI/ HPCI/ HPCS	AFW/ RCIC	RHR	SWS [a]	CCW [a]	PORV [a]
PWRs							
WE 4-Lp Plant 28	1.8E-2 (G)	2.2E-2 (G)	MDP (3.7E-3) (G) TDP (1.0E-3) (G)	9.2E-3 (G)	No data	No data	No data

a. Unavailability data are not available (not covered by the ROP) at this time. Eventually, EPIX may contain such data.
b. Since the models and data in these tables have not completed formal peer review, plant-specific inferences regarding green or nongreen performance from these calculations would be inappropriate.
c. "NA" in a system cell for a given plant indicates that the system does not exist at that plant.

Table 5.3-3 Plant Performance Bands for Mitigating System Unreliability RBPIs (1997 - 1999)[c]

Plant	EPS	HPI/ HPCI/ HPCS	AFW/ RCIC	RHR[a]	SWS	CCW	PORV
BWRs							
BWR Plant 123 Plant 1	< baseline (G)[b]	< baseline (G)	< baseline (G)	< baseline (G)	No data[c]	NA	NA
BWR Plant 123 Plant 2	< baseline (G)	< baseline (G)	< baseline (G)	< baseline (G)	No data	NA	NA
BWR 3/4 Plant 1	< baseline (G)	< baseline (G)	< baseline (G)	< baseline (G)	No data	NA	NA
BWR 3/4 Plant 2	< baseline (G)	< baseline (G)	< baseline (G)	< baseline (G)	No data	NA	NA
BWR 3/4 Plant 3	< baseline (G)	< baseline (G)	< baseline (G)	< baseline (G)	No data	NA	NA
BWR 3/4 Plant 4	< baseline (G)	< baseline (G)	< baseline (G)	< baseline (G)	No data	NA	NA
BWR 3/4 Plant 5	< baseline (G)	< baseline (G)	< baseline (G)	< baseline (G)	No data	NA	NA
BWR 3/4 Plant 6	< baseline (G)	< baseline (G)	< baseline (G)	< baseline (G)	No data	NA	NA
BWR 3/4 Plant 8	< baseline (G)	< baseline (G)	< baseline (G)	< baseline (G)	No data	NA	NA
BWR 3/4 Plant 11	< baseline (G)	< baseline (G)	< baseline (G)	< baseline (G)	No data	NA	NA
BWR 3/4 Plant 12	< baseline (G)	< baseline (G)	< baseline (G)	< baseline (G)	No data	NA	NA
BWR 3/4 Plant 13	< baseline (G)	< baseline (G)	< baseline (G)	< baseline (G)	No data	NA	NA
BWR 3/4 Plant 15	< baseline (G)	< baseline (G)	< baseline (G)	< baseline (G)	No data	NA	NA
BWR 3/4 Plant 16	< baseline (G)	< baseline (G)	< baseline (G)	< baseline (G)	No data	NA	NA
BWR 3/4 Plant 18	< baseline (G)	< baseline (G)	< baseline (G)	< baseline (G)	No data	NA	NA
BWR 3/4 Plant 19	< baseline (G)	< baseline (G)	1.1E-1 (W) (0.07)[d]	< baseline (G)	No data	NA	NA
BWR 5/6 Plant 2	< baseline (G)	< baseline (G)	< baseline (G)	< baseline (G)	< baseline (G)	NA	NA
BWR 5/6 Plant 5	< baseline (G)	< baseline (G)	< baseline (G)	< baseline (G)	No data	NA	NA
BWR 5/6 Plant 8	< baseline (G)	< baseline (G)	1.2E-1(W) (0.05)[d]	< baseline (G)	No data	NA	NA
PWRs							
B&W Plant 3	< baseline (G)	< baseline (G)	< baseline (G)	< baseline (G)	< baseline (G)	NA	No data
B&W Plant 4	< baseline (G)	< baseline (G)	< baseline (G)	< baseline (G)	< baseline (G)	< baseline (G)	NA
B&W Plant 5	< baseline (G)	< baseline (G)	< baseline (G)	< baseline (G)	< baseline (G)	< baseline (G)	NA
B&W Plant 6	< baseline (G)	< baseline (G)	< baseline (G)	< baseline (G)	< baseline (G)	< baseline (G)	NA
B&W Plant 7	< baseline (G)	< baseline (G)	< baseline (G)	< baseline (G)	< baseline (G)	< baseline (G)	No data
CE Plant 1	< baseline (G)	< baseline (G)	< baseline (G)	< baseline (G)	< baseline (G)	< baseline (G)	No data
CE Plant 2	< baseline (G)	< baseline (G)	< baseline (G)	< baseline (G)	No data	No data	No data
CE Plant 3	< baseline (G)	< baseline (G)	< baseline (G)	< baseline (G)	No data	No data	No data
CE Plant 4	< baseline (G)	< baseline (G)	< baseline (G)	< baseline (G)	< baseline (G)	No data	< baseline (G)
CE Plant 5	< baseline (G)	< baseline (G)	< baseline (G)	No data	No data	< baseline (G)	No data
CE Plant 10	< baseline (G)	< baseline (G)	< baseline (G)	< baseline (G)	< baseline (G)	< baseline (G)	No data
CE Plant 11	< baseline (G)	< baseline (G)	< baseline (G)	< baseline (G)	< baseline (G)	< baseline (G)	No data

Table 5.3-3 (Continued)

Plant	EPS	HPI/ HPCI/ HPCS	AFW/ RCIC	RHR[a]	SWS	CCW	PORV
PWRs							
CE Plant 12	< baseline (G)	< baseline (G)	< baseline (G)	< baseline (G)	No data	< baseline (G)	No data
WE 2-Lp Plant 5	< baseline (G)	< baseline (G)	< baseline (G)	< baseline (G)	No data	< baseline (G)	< baseline (G)
WE 2-Lp Plant 6	< baseline (G)	No data	< baseline (G)	< baseline (G)	< baseline (G)	No data	< baseline (G)
WE 3-Lp Plant 5	< baseline (G)	< baseline (G)	< baseline (G)	< baseline (G)	< baseline (G)	< baseline (G)	< baseline (G)
WE 3-Lp Plant 6	< baseline (G)	< baseline (G)	< baseline (G)	< baseline (G)	< baseline (G)	< baseline (G)	< baseline (G)
WE 3-Lp Plant 10	< baseline (G)	< baseline (G)	< baseline (G)	< baseline (G)	< baseline (G)	< baseline (G)	< baseline (G)
WE 4-Lp Plant 1	< baseline (G)	No data	< baseline (G)	< baseline (G)	No data	No data	No data
WE 4-Lp Plant 2	< baseline (G)	No data	< baseline (G)	< baseline (G)	No data	No data	No data
WE 4-Lp Plant 10	< baseline (G)	< baseline (G)	< baseline (G)	< baseline (G)	< baseline (G)	< baseline (G)	< baseline (G)
WE 4-Lp Plant 11	< baseline (G)	< baseline (G)	< baseline (G)	< baseline (G)	< baseline (G)	< baseline (G)	< baseline (G)
WE 4-Lp Plant 22	< baseline (G)	< baseline (G)	< baseline (G)	< baseline (G)	< baseline (G)	< baseline (G)	< baseline (G)
WE 4-Lp Plant 23	< baseline (G)	< baseline (G)	1.6E-2 (MDP) (W) (0.13)[d]	< baseline (G)	< baseline (G)	< baseline (G)	< baseline (G)
WE 4-Lp Plant 28	< baseline (G)	< baseline (G)	< baseline (G)	< baseline (G)	< baseline (G)	< baseline (G)	< baseline (G)

a. Reflects pump data. Valve data still need to be collected and evaluated.
b. "< baseline" indicates that there were not enough failures to result in a train unreliability greater than the baseline.
c. "No data" indicates that either EPIX has no data on this system or the RADS data load of the EPIX file did not include this system.
d. Unreliability RBPIs have the potential for false-positive indications. Therefore, for white indications, the probability of observing this performance if the plant is actually at its baseline (G) is also presented. For example, a 0.25 probability indicates that there is a 25% chance of experiencing the observed performance, even with the plant at baseline.
e. Since the models and data in these tables have not completed formal peer review, plant-specific inferences regarding green or nongreen performance from these calculations would be inappropriate.
f. "NA" in a system cell for a given plant indicates that the system does not exist at that plant.

Table 5.3-4 Plant Performance Bands for Component Class RBPIs (1997 - 1999)[e]

Plant	AOV	MOV	MDP
BWRs			
BWR 123 Plant 1	No data[a]	< baseline (G)[b]	9.6E-3 (2.6X)
BWR 123 Plant 2	< baseline (G)	< baseline (G)	9.7E-3 (2.6X)
BWR 3/4 Plant 1	No data	< baseline (G)	< baseline (G)
BWR 3/4 Plant 2	No data	< baseline (G)	< baseline (G)
BWR 3/4 Plant 3	No data	< baseline (G)	6.7E-3 (1.8X)
BWR 3/4 Plant 4	No data	< baseline (G)	4.3E-3 (1.2X)
BWR 3/4 Plant 5	No data	< baseline (G)	< baseline (G)
BWR 3/4 Plant 6	< baseline (G)	< baseline (G)	< baseline (G)
BWR 3/4 Plant 8	No data	< baseline (G)	< baseline (G)
BWR 3/4 Plant 11	No data	< baseline (G)	< baseline (G)
BWR 3/4 Plant 12	No data	< baseline (G)	< baseline (G)
BWR 3/4 Plant 13	No data	< baseline (G)	< baseline (G)
BWR 3/4 Plant 15	No data	3.6E-3 (1.2X)	< baseline (G)
BWR 3/4 Plant 16	No data	< baseline (G)	< baseline (G)
BWR 3/4 Plant 18	No data	< baseline (G)	< baseline (G)
BWR 3/4 Plant 19	No data	< baseline (G)	< baseline (G)
BWR 5/6 Plant 2	No data	< baseline (G)	< baseline (G)
BWR 5/6 Plant 5	No data	< baseline (G)	< baseline (G)
BWR 5/6 Plant 8	No data	7.4E-3 (2.5X) (Y)[c] (0.0013)	6.5E-3 (1.8X)
PWRs			
B&W Plant 3	< baseline (G)	< baseline (G)	8.5E-3 (2.3X)
B&W Plant 4	< baseline (G)	< baseline (G)	< baseline (G)
B&W Plant 5	< baseline (G)	< baseline (G)	< baseline (G)
B&W Plant 6	< baseline (G)	4.8E-3 (1.6X)	< baseline (G)
B&W Plant 7	< baseline (G)	< baseline (G)	4.5E-3 (1.2X)
CE Plant 1	< baseline (G)	< baseline (G)	< baseline (G)
CE Plant 2	< baseline (G)	5.2E-3 (1.7X)	< baseline (G)
CE Plant 3	< baseline (G)	< baseline (G)	< baseline (G)
CE Plant 4	3.8E-3 (3.8X) (G)[c]	< baseline (G)	< baseline(G)
CE Plant 5	No data	< baseline (G)	< baseline (G)
CE Plant 10	< baseline (G)	< baseline (G)	< baseline (G)
CE Plant 11	< baseline (G)	< baseline (G)	< baseline (G)
CE Plant 12	2.9E-3 (2.9X) (G)	4.5E-3 (1.5X) (W)[c] (0.14)[d]	< baseline (G)
WE 2-Lp Plant 5	< baseline (G)	< baseline (G)	< baseline (G)
WE 2-Lp Plant 6	< baseline (G)	< baseline (G)	< baseline (G)
WE 3-Lp Plant 5	1.1E-3 (1.1X) (G)	< baseline (G)	< baseline (G)
WE 3-Lp Plant 10	4.9E-3 (4.9X) (W) (0.0001)	< baseline (G)	< baseline (G)
WE 3-Lp Plant 11	< baseline (G)	< baseline (G)	< baseline (G)
WE 4-Lp Plant 1	No data	No data	< baseline (G)
WE 4-Lp Plant 2	No data	No data	< baseline (G)
WE 4-Lp Plant 10	< baseline (G)	< baseline (G)	< baseline (G)
WE 4-Lp Plant 11	< baseline (G)	< baseline (G)	< baseline (G)

Table 5.3-4 (Continued)

Plant	AOV	MOV	MDP
PWRs			
WE 4-Lp Plant 22	< baseline (G)	< baseline (G)	< baseline (G)
WE 4-Lp Plant 23	< baseline (G)	< baseline (G)	< baseline (G)
WE 4-Lp Plant 28	< baseline (G)	< baseline (G)	< baseline (G)

a. "No data" indicates that either EPIX has no data on this component class or the RADS data load of the EPIX file did not include this component class.

b. "< baseline" indicates that there were not enough failures to result in a train unreliability greater than the baseline.

c. The number in parentheses "3.8X" indicates that the unreliability is 3.8 times the baseline.

d. The component class RBPIs have the potential for false-positive indications. Therefore, for white indications, the probability of observing this performance if the plant is actually at its baseline (G) value is also presented.

e. Since the models and data in these tables have not completed formal peer review, plant-specific inferences regarding green or nongreen performance from these calculations would be inappropriate.

The results in Tables 5.3-1 through 5.3-4 are intended to show that RBPIs can be calculated using readily available data and models to produce potential indicators that reflect plant performance in a manner consistent with the current ROP philosophy. These tables clearly show that performance data can be used to calculate indicators that fit in the ROP concept. They demonstrate the feasibility of the process, but not necessarily the accuracy of the results. In order for these potential indicators to be used in the ROP, implementation issues relating to model fidelity and data quality need to be resolved so that there is sufficient alignment among stakeholders regarding the accuracy of both the thresholds and the calculated performance indicators.

The risk models and associated baseline performance values should be peer-reviewed by stakeholders to ascertain that they reasonably reflect the risk profile for the plants modeled. This is required to assure that thresholds derived from the models reasonably represent the risk significance of potential performance degradations. Similarly, the data inputs to the indicator calculations need to have sufficient accuracy to reasonably represent the risk significance of potential performance degradations. The accuracy should be consistent with the nominal uncertainties associated with unreliability and risk measurements so that errors in data collection do not result in mischaracterizing risk performance as measured by the ROP (i.e., characterizing green when actually nongreen or vice versa).

Since the models and data in these tables have not been formally peer reviewed, plant- specific inferences regarding green or nongreen performance from these calculations would be inappropriate. The data are presented to demonstrate that the process can be followed to produce potential indicators. The accuracy of the RBPI results sufficient for use in NRC decisionmaking remains to be determined through the ROP change process.

Tables 5.3-1 through 5.3-4 show how performance data can be used along with thresholds derived from risk models to produce indicators that are consistent with the ROP framework. Potential benefits derived from this exercise that relate to the practicality of calculation and credibility of results include:

- more precise accounting for the risk-significant design features of plants
- more plant-specific thresholds
- more appropriate accounting for the risk impact of fault exposure time in indicator formulation.

By evaluating indicators at a train level and accounting for diverse design features separately, the RBPIs can more precisely account for the risk significance of design features. For example, AFW systems in PWRs have turbine-driven, diesel-driven, and/or motor-driven pump trains. Turbine-driven or diesel-driven trains have risk significance in station blackout (SBO) sequences that motor-driven trains do not. By accounting for these effects separately, rather than combining them in a single indicator, RBPIs can more precisely account for risk-significant design features.

The use of plant-specific models to set thresholds allows the indicators for a plant to more closely reflect the risk significance of potential performance degradations. As noted earlier, the models used in the RBPI development need to be reviewed by licensees and other external stakeholders to determine if they represent a reasonable characterization of the plant risk profile.

Fault exposure time data collection and analysis is one method of estimating the probability of standby components, trains, or systems failing to perform their risk-significant safety function when needed. Assessing the probability through analysis of failure and demand counts is another method. Both methods produce the same result over a long period of time. However, counting fault exposure time over the shorter periods of time typical of the ROP sampling intervals can be problematic due to the increased likelihood of false positive and false negative indication. As noted in Appendix F, the RBPIs process fault exposure data and failure and demand count data in a manner that provides the most timely indication of potential performance degradation without undue occurrence of false positive or false negative indications. In addition, the RBPIs account for fault exposure time impacts on the risk-significant safety functions which can be different from the design basis functions of components, trains, or systems. For example, many systems have automatic initiation capabilities as design basis features (without credit for manual operation). However, to achieve the risk-significant safety function, either automatic or manual actuation is satisfactory. The RBPIs account for this case in the treatment of fault exposure time so that risk significance of events resulting in fault exposure time accumulation are more appropriately accounted for.

Testing of the RBPIs also included the monitoring of industry-wide performance. Industry-wide trending data are presented in Tables 5.3-5 through 5.3-8. The industry-wide averages were determined using only the 44 plants covered in this study, 25 PWRs and 19 BWRs. Statistical trending analyses have not been performed yet because only approximately two-fifths of the entire industry is represented at present, and 3 years of data are generally not sufficient to discern statistically significant trends, unless performance is changing rapidly. However, almost all of the yearly industry-wide averages lie below the 95th percentile of the distributions of the 1996 industry-average baselines. (The only exception is the AFW motor-driven pump train UA, where the yearly averages range from two to five times the baseline value. In this case the baseline value might need to be modified.) Similar to the testing of the RBPIs on a plant-specific basis, the industry-wide trending was accomplished using existing databases and software. In general, the trending data presented in Table 5.3-5 through 5.3-8 indicate that the values chosen to represent 1996 industry-average performance are reasonable, and that industry performance during 1997–1999 was comparable to or better than the 1996 baseline.

INPO's EPIX database, used to support evaluation of mitigating system and component class unreliabilities, is relatively new. A review of the data collection effort indicates that approximately 40% of the plants considered were missing some data for the four main types of systems considered—EPS, HPI/HPCI/HPCS, AFW/RCIC, and RHR. In addition, approximately 50% of the plants did not have data for the other systems considered—SWS, CCW, and PORV. Therefore, the EPIX database needs to be improved in this area before all of the proposed RBPIs could be implemented.

Unavailability data for the four main types of systems were obtained from the ROP. However, the ROP does not include other systems such as SWS, CCW, and PORV. The industry is considering the inclusion of unavailability data for these systems in the EPIX database. The addition of unavailability data to EPIX would help to support the RBPI program, especially for systems not covered by the ROP.

Table 5.3-5 Industry Trends for Initiating Event RBPIs (1997 through 1999)

	Industry-Wide Initiating Event Frequency [a]			
Initiating Event	1996 Baseline	1997	1998	1999
General transient (GT)	9.6E-1/y (PWRs) 1.2/y (BWRs)	6.7E-1/y 4.9E-1/y	7.4E-1/y 6.5E-1/y	8.5E-1/y 8.3E-1/y
Loss of Heat Sink (LOHS)	9.6E-2/y (PWRs) 2.3E-1/y (BWRs)	5.1E-2/y 6.5E-2/y	5.0E-2/y 1.1E-1/y	1.1E-1/y 1.1E-1/y
Loss of Feedwater (LOFW)	6.8E-2/y	4.9E-2/y	1.1E-1/y	NA [b]

a. The "industry-wide" results are the average of the 44 plants considered in this data review. The PWR results are for 25 plants. The BWR results are for 19 plants.

b. Data not available (without a review of the LERs for 1999).

Table 5.3-6 Industry Trends for Mitigating System Unavailability RBPIs (1997 through 1999)

	Industry-Wide Unavailability			
Mitigating System and Level	1996 Baseline	1997	1998	1999
EPS (train)	9.7E-3	1.2E-2	1.1E-2	1.1E-2
PWRs				
HPI (train)	4.2E-3	4.9E-3	4.3E-3	5.3E-3
AFW (MDP train)	1.1E-3	5.5E-3	2.8E-3	4.1E-3
AFW (TDP train)	4.6E-3	4.9E-3	6.4E-3	5.3E-3
AFW (DDP train)	1.5E-2	6.9E-3	1.7E-3	7.4E-3
RHR (train)	7.3E-3	9.3E-3	6.1E-3	8.2E-3
BWRs				
HPCI (train)	9.7E-3	1.3E-2	1.8E-2	4.5E-2
HPCS (train)	3.4E-3	9.0E-3	3.9E-3	1.0E-2
RCIC (train)	1.3E-2	9.0E-3	1.6E-2	1.6E-2
RHR (train)	1.0E-2	1.3E-2	1.5E-2	8.7E-3

Table 5.3-7 Industry Trends for Mitigating System Unreliability RBPIs (1997 through 1999)

	Industry-Wide Unreliability[a]			
Mitigating System and Level	1996 Baseline	1997	1998	1999
EPS (train)	4.1E-2	1.5E-2	9.3E-3	6.8E-3
PWRs				
HPI (train)	7.9E-3	6.2E-4	1.2E-3	4.0E-4
AFW (MDP train)	7.8E-3	4.1E-3	7.5E-3	4.5E-3
AFW (TDP train)	2.0E-1	2.6E-2	5.1E-2	7.9E-2
AFW (DDP train)	5.7E-2	2.7E-2	2.7E-2	5.5E-2
RHR (train)	1.1E-2	?[b]	?	?
BWRs				
HPCI (train)	4.3E-2	3.0E-2	2.5E-2	3.4E-2
HPCS (train)	6.8E-2	4.9E-2	3.1E-2	3.1E-2
RCIC (train)	4.4E-2	2.7E-2	2.2E-2	2.7E-2
RHR (train)	1.6E-2	?[b]	?	?

a. Train unreliability models vary by plant. For the industry-wide trending, the train unreliability was simplified to include the pump Fail to Start (FTS) and Fail to Run (FTR) (or EDG FTS, Fail to Load and Run (FTLR), and FTR), single-failure valves within the train, and train unavailability (kept at the baseline value). A 4-hour mission time was assumed for EDGs, and a 24-hour mission time for all other trains.

b. Valve data still need to be collected to evaluate this properly.

Table 5.3-8 Industry Trends for Component Class RBPIs (1997 through 1999)

	Industry-Wide Unreliability			
Component Class	1996 Baseline	1997	1998	1999
AOV	1.0E-3	2.6E-3	2.8E-3	1.3E-3
MOV	3.0E-3	6.5E-4	6.6E-4	3.4E-3
MDP[a]	3.7E-3	5.7E-4	1.1E-3	7.8E-4
TDP[a]	1.0E-1	3.4E-2	4.2E-2	3.4E-2

a. Unreliability includes FTS (baseline of 3.0E-3) and FTR (baseline λ of 3.0E-5/h and a mission time of 24 hours).

b. TDP is not an RBPI, but is trended at the industry level. Unreliability includes FTS (baseline of 1.4E-2, which is a weighted average of AFW, HPCI, and RCIC TDPs) and FTR (baseline λ of 3.7E-3/h, which is a weighted average, and a mission time of 24 hours).

6. KEY ISSUES AFFECTING RBPI DEVELOPMENT AND IMPLEMENTATION

The following subsections describe issues that have emerged in the course of the development described in this report. This work is part of the development and evolution of performance indicators in the current ROP and is closely coordinated with existing ROP efforts. There are several key implementation issues summarized in the executive summary and in this section, including the verification of risk models and data. The potential integration of RBPIs into the ROP would follow the guidelines in IMC-0608, "Performance Indicator Program." This would include a pilot program prior to the implementation of any or all RBPIs and interaction with stakeholders to resolve implementation issues raised in this report or from external stakeholders during the review of this report.

6.1 Program Coordination Issues

The following specific issues have been considered by the stakeholders:

- Are additional RBPIs needed in the ROP?
- Is the number of potential new indicators appropriate?
- Do the data sources for RBPIs exist and have sufficient quality for use in the ROP?
- Will additional SPAR Revision 3i models be available for setting plant-specific thresholds for all plants?
- Will SPAR LERF models be available for setting thresholds for mitigating and containment systems?

<u>Are any additional performance indicators needed in the ROP?</u>

Interactions with stakeholders commenting on the White Paper indicated differing views on this subject. Industry representatives questioned whether NRC needed to have a broader coverage of risk measured in the ROP indicators, especially if it did not result in a corresponding reduction in the inspection program. Other external stakeholder comments favored more indicators as well as additional inspections.

The RBPI development program is focused on demonstrating the technical feasibility of providing additional objective indicators that cover a broader spectrum of risk-significant plant performance. Future work may identify additional candidates. Any potential new performance indicators will be assessed in a pilot program consistent with the change process described in IMC-0608 prior to implementation.

Subsequent to the closing of the comment period for this report, the agency and industry (through the continuing ROP interactions) have identified several aspects of unreliability and unavailability indicators from the RBPI development that will be piloted in 2002 for potential implementation in the ROP. These involve unreliability and unavailability indicators associated with the six SSUPIs under the mitigating system cornerstone of the current ROP.

Is the number of potential new indicators appropriate?/Which of the proposed indicators would be most beneficial?

The RBPI Phase 1 development identified 22 potential indicators for PWRs and 17 potential indicators for BWRs. If all of these performance indicators were implemented, they could potentially replace 8 (3 initiating event and 5 mitigating system) of 18 existing indicators in whole or in part bringing the total number of indicators per plant to about 30. In addition to the issue of the appropriate risk scope of ROP indicators (noted above), it will be necessary to assess whether potentially expanding the total number of indicators to approximately 30 (approximately 25 based on currently available data) per plant is reasonable from a logistics/process point of view. For example, the criteria that result in plants entering various columns of the Action Matrix would have to be reconsidered. Section 6.5 discusses results of preliminary work to examine the feasibility of developing indicators at a higher level (system or cornerstone level) by combining results of lower level data and models. In follow-on work, higher-level indicators may be investigated further.

Do the data sources for RBPIs exist and have sufficient quality for use in the ROP?

A significant portion of the RBPIs require access to and use of data from the Equipment Performance and Information Exchange (EPIX) system. These data are voluntarily provided by industry in response to the Commission decision to forgo the Reliability Data Rule. Full industry participation, verification and validation of existing EPIX, and development of guidelines for consistent reporting are important to the feasibility of many RBPIs as potential improvements to the ROP.

Performance data are not readily available from EPIX for several of the proposed indicators. The NRC is working with industry groups to expand the unreliability data collection in this voluntary system to include data that will support evaluation of performance in these areas.

Data accuracy and licensee burden in this area are recognized as important implementation issues, which will be further investigated during the implementation phase using the change process in IMC-0608.

Will SPAR Revision 3i models be used for setting plant-specific thresholds for all plants?

Approximately 50 Standardized Plant Accident Risk (SPAR) Revision 3i models are currently available. Completion of all 70 SPAR Revision 3i models is scheduled for the end of calendar year 2002. As more models are made available for use in the RBPI development program, it will be possible to determine if plants can be grouped so that a few models can be used to set thresholds for all plants or individual models will be needed for each. The RBPI development program will continue to use the SPAR Revision 3i models as they are developed. External stakeholder comments on the White Paper indicated that peer review by licensees should be included in the development of these models. An additional implementation issue relates to whether licensees or NRC will calculate the thresholds and indicators as well as whether licensee models (meeting as yet to be developed NRC specifications) could be used instead of the SPAR models.

It is yet to be determined whether a plant-specific model will be required to set performance thresholds for each plant or a representative model is sufficient for a group of plants. Furthermore, it has not been determined whether the calculation for thresholds and indicators will be routinely performed by NRC staff using SPAR Revision 3i models, licensees using SPAR Revision 3i models, or licensees using their own risk models that meet some specifications agreed upon and reviewed by the NRC. These are potential options that will be dealt with through IMC-0608.

Will LERF models be used for setting thresholds for mitigating and containment systems?

There are a limited number of large early release frequency (LERF) models available to set thresholds for performance of systems that impact the integrity of the containment barrier. In addition, currently available data are inadequate for establishing performance measures for the containment systems. Also, for some systems under the mitigating systems cornerstone, the thresholds associated with changes in core damage frequency (CDF) due to performance degradations may not be limiting compared to changes in LERF. To assess that condition, LERF models that reflect the impact of potential CDF changes are needed. The current plan for developing LERF models over the next several years will support limited capability for identifying RBPIs or setting plant-specific LERF thresholds.

6.2 Plant-Specific RBPI Formulation

Based on risk-significance, some systems warrant RBPI coverage only at certain plants. From a risk coverage point of view, it may be desirable to include these systems in RBPI development. However, this leads to different numbers of indicators at different plants, and calls for more performance data to be collected through EPIX.

Options:

- Develop RBPIs for all systems satisfying standard criteria, and upgrade the collection of performance information to support quantification
- Maintain a generic set of RBPIs that are applicable to specific plant groups and can be supported with currently available data and logic models

Within the partial set of SPAR models available to the Phase 1 development, it was not possible to identify groups of plants within PWRs or BWRs to which a specific set of RBPIs would be applicable. When a complete set of SPAR models becomes available, another effort will be made to identify such plant groups.

6.3 Selection of Risk Metrics for Use in Assessing Containment Barrier Performance

Large early release frequency (LERF) is one important metric used for assessing the risk significance of proposed changes to the licensing basis. However, many significant elements of containment barrier performance discussed in SECY 99-007 do not affect either CDF or LERF significantly, although they affect late release frequency or other post-accident considerations such as worker dose. Currently, the graded approach in SECY 99-007 defines performance bands in terms of changes in CDF and LERF. However, if performance bands for large late release frequency were derived from the QHOs in the same way that performance bands for LERF are derived, then performance thresholds for many of these elements would be implied.

Quantification of thresholds based on changes in late release frequency would require either additional SPAR model development, or formulation of approximate approaches such as those being developed as part of the SDP.

Options:

- Use LERF only
- Develop models and apply to RBPI development addressing large late release frequency (LLRF)

In follow-on work, the feasibility and usefulness of RBPIs that address LLRF will be investigated.

6.4 Formulation of G/W Threshold In Terms of Performance Percentile

In some cases, relatively small changes in element performance are capable of causing a 1E-6 change in CDF. For such elements, placing the G/W threshold at this performance level makes false positive indications more likely. An alternative approach is to define the G/W threshold in terms of performance relative to the operating fleet. However, at some plants, the 95th percentile of system performance corresponds to a ΔCDF in a white or even yellow performance band. The current plan is to continue to apply a ΔCDF threshold of 1E-6, and address high false positive probability on a case-specific basis by supplementing each nongreen RBPI indication with an evaluation of the probability of observing that performance, given that the actual performance is at the baseline level.

Options:

- Continue to use a ΔCDF threshold of 1E-6, and identify RBPIs with high false positive probabilities
- Use 95th percentile
- Use a different ΔCDF threshold
- Use a different percentile

The current approach is to use a ΔCDF threshold of 1E-6, and identify RBPIs with high false positive probabilities.

6.5 Development of RBPIs at Higher Level

Because of industry concerns that the number of component/train level RBPIs may be too high, some preliminary work was done to assess the potential for reduced sets of higher-level RBPIs. Three sets of higher-level RBPIs were considered for at-power, internal events: the cornerstone level (initiating events and mitigating systems), the functional level mitigation (initiator types with associated mitigating systems), and the functional level mitigation (systems). Each of these higher-level RBPIs uses a subset of the component/train level and initiating event RBPIs discussed in Section 3 of this report. These concepts were presented to the ACRS (May 10, 2001) and discussed with external stakeholders. ACRS and external stakeholder comments on these higher-level RBPIs are discussed in Appendix I.

These higher-level RBPIs were quantified for selected examples from existing data. In these cases, performance indications from the higher-level RBPIs tended to be green when the lower-level inputs were green, and white when some of the lower-level inputs were white. These examples suggest that indications from higher-level RBPIs will correspond appropriately to the inputs, but it is necessary to characterize the behavior of these RBPIs more carefully over the range of possible inputs before recommending them for trial.

Defining RBPIs at higher levels has both potential benefits and potential limitations. A potential benefit is the reduced number of RBPIs. However, higher-level RBPIs generally use all of the performance data collected for the lower level RBPIs discussed in Section 3. These higher-level RBPIs also can balance the impacts of individual lower level RBPIs. For example, a high system train unavailability can be balanced by a low train unreliability. This approach is more consistent with the Maintenance Rule philosophy, wherein a balance between unreliability and unavailability is sought. Also, for higher-level RBPIs, poor performance of one system can potentially be balanced by good performance of another system. A limitation of these higher-level RBPIs is that when potentially degraded performance is indicated, further analysis (at a lower level) is required to identify the major contributors to the degraded performance.

Options:

- Continue to study the potential for developing RBPIs at a higher level
- Focus on RBPIs at the lower level outlined in this report (Section 3)

In follow-on work, we will continue to study the potential for developing RBPIs at a higher level.

6.6 Issues Related to Shutdown RBPI Development

SECY 99-007 indicated that a PI would be developed to monitor configuration management at shutdown. The development summarized in Section 3.2, and described more fully in Appendix B, was aimed at directly monitoring the risk incurred during shutdown by monitoring the time spent in risk-significant configurations. Several issues emerged in the course of that development as a result of stakeholder interactions. They are summarized and discussed below.

(1) Operational exigencies drive variations in shutdown risk that are substantially greater than the risk changes that are associated with PI thresholds in the ROP. PIs that measure changes in shutdown risk therefore capture influences whose relationship to licensee performance in configuration management is indirect. This potentially leads to unintended consequences.

Variations in risk at full power are typically less than variations in risk at shutdown, because at full power, variations are limited by the physical conditions at full power and by technical specifications. Changes in plant condition that are associated with major variations in risk at full power promptly lead to shutdown, automatically or by procedure. However, at shutdown, risk can vary more as a function of configuration. Review of actual outage experience shows that shutdown risk does vary. As a result, even within a representative sample of "normal" outages, the variance around mean behavior is large compared to ROP threshold values for ΔCDF. Moreover, much of this variation is driven by operational needs that may relate only indirectly, if at all, to licensee performance.

(2) The risk in certain configurations can be reduced by enhancing licensee readiness to respond to initiating events in those configurations. NUMARC 91-06 describes such compensatory measures that utilities generally use. The assessment of configuration risk significance in Section 3.2 does not account for all such compensatory measures.

Current licensee practice implicitly takes credit for risk reduction through licensee readiness, in that configurations are entered that would be assigned a high conditional CDF if not for compensatory measures. Existing risk models do not credit all possible compensatory measures. Therefore, the risk calculated for certain configurations may be higher than it would be if credit were taken for all compensatory measures. As a result, promulgating the baselines and thresholds presented in Section 3.2 for the shutdown PIs may not appropriately credit current operational practice.

(3) Models comparable to SPAR models for full power are not available for shutdown. Therefore, a development of plant-specific PI thresholds comparable to that for full power is not currently practical.

Development of the RBPIs for shutdown as discussed in Section 3.2 would likely require plant-specific models for two reasons: (1) classifying configurations according to risk significance requires plant-specific models, and (2) specification of baselines for those RBPIs requires access to a representative sample of outage schedules that have been mapped through a risk model to generate representative dwell times in risk-significant configurations. At this time, a complete set of models is not available, and even if they were available, a significant effort would be needed to assemble a representative sample of outage schedules and propagate them through the appropriate models. Before undertaking this, it would be necessary to address the issue regarding modeling of all compensatory measures.

(4) Development of a baseline would require characterizing a nominal outage, based on review of a large number of outage schedules and processing them through risk models.

For reasons mentioned above under (1), defining an applicable baseline for shutdown is difficult. The development presented in Section 3.2 established nominal times spent in each configuration category. For some activities, such as time spent in mid-loop early in an outage, small changes from the nominal produce relatively larger changes in risk. Thus, the indicator would be very sensitive to any changes above nominal conditions. As there may be times when longer than nominal conditions are necessary, the indicator may become a *de facto* operating limit.

(5) Because risk changes from configuration management are detectable in short time intervals, monitoring shutdown risk has an episodic, SDP character, rather than a longer-term, trending-indicator character. This raises the question whether this development should be subsumed in the SDP.

An approach that may address the above issues is to have the shutdown performance indicators formulated to quantify the deviation during each outage from the governing outage plan, and to assess separately (by inspection) the merits of the outage plan itself. PI thresholds would be exceeded if *planned* durations of risk-significant configurations were exceeded by significant amounts in the actual outages. In effect, the outage plan becomes an outage-specific baseline. Configurations could continue to be classified as in Section 3.2, although work would need to be done to address the treatment of applicable compensatory measures, and the PI would be defined as excess time beyond the outage plan spent in each configuration category. The short threshold times associated with risk-significant configurations could be anticipated in the licensee's formulation of each outage plan, so that exceedance of scheduled time would signal a performance deviation.

Measuring configuration management relative to an outage plan would alter the priorities in model development, and would not require formulating a baseline in terms of "normal" outage behavior based on historical outage plans. This approach more closely comports with guidance in NUMARC 91-06, which recommends monitoring of adherence to outage plans as an effective way to assess licensee performance.

Options:

1. Instead of developing PIs for shutdown, rely entirely on inspection to sample licensee performance at shutdown.
2. Finalize the PIs developed in Section 3.2 now, using generic models.
3. Obtain plant-specific models, then finalize the PIs developed in Section 3.2.
4. Develop PIs that measure configuration management relative to outage plan; address outage plans themselves through inspection.
5. Use RBPI concepts to develop a better shutdown SDP process.

Follow-on work will use RBPI concepts to develop a better shutdown SDP process.

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APPENDIX A

RBPI DETERMINATION FOR INTERNAL EVENTS/ FULL POWER ACCIDENT RISK

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Appendix A: RBPI Determination for Internal Events / Full Power Accident Risk

A.1 Initiating Events Cornerstone

This section discusses development of RBPIs that address the initiating events cornerstone for full power, internal events. External events and non-power modes are addressed in other sections. Each subsection describes the analyses for the steps from Figure 2.1 of the main report.

A.1.1 Assess the Potential Risk Impact of Degraded Performance

The objective of the initiating events cornerstone is to limit the frequency of those events that upset plant stability and challenge critical safety functions. Six 'key attributes' that contribute to initiating event frequency are identified in SECY 99-007 (Ref. 1). These six attributes consist of configuration control, procedure quality, human performance, protection against external factors, equipment performance, and design.

A.1.1.1 Determine Attributes That Are Risk-Significant and Explicitly Modeled

Identification of 'risk-significant' or 'risk-based' performance indicators necessitates a means of quantifying the impact of that attribute. Initiating events are unique among the cornerstones of safety in that their performance is quantified at the cornerstone level rather than at lower level quantities (i.e., the attribute level). Since initiating events represent the highest level element of risk pertaining to the cornerstone, they are used directly. Risk-significance of initiating events was determined through evaluation of the Individual Plant Examination (IPE) submittals and the associated IPE Database[1]. The IPE Database provides a succinct summary of industry-wide IPE data including initiator specific conditional core damage probabilities (CCDPs) and core damage frequencies (CDF). From this database, initiators with a CCDP $\geq$ 1E-6 and a contribution to industry-wide CDF $\geq$ 1% were identified as risk-significant. An exception to this rule was made for transients. They were included as a candidate RBPI even though the CCDP is less than 1E-6. The complete list of risk-significant initiating events is shown below in Table A.1.1.1-1. Initiating events contained in this table are grouped according to the convention used in NUREG/CR-5750 (Ref. 2).

The RBPI white paper (Ref. 3) indicates that RBPI development will be performed in a manner to group similar plants so that a given set of RBPIs apply to the entire group. In accordance with the data analysis performed in NUREG/CR-5750, only three schemes for grouping initiating events were considered; industry-wide, pressurized water reactors (PWRs) and boiling water reactors (BWRs). The list of risk-significant initiating events and the plant groups to which they are generically applicable are listed in Table A.1.1.1-1.

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[1] Su, T. M., et al., "Individual Plant Examination Database – User's Guide," NUREG-1603, U.S. NRC, April 1997

Table A.1.1.1-1 Modeled Risk-Significant Initiators

BWR INITIATOR	NUREG/CR-5750 Initiator	CCDP≥1E-6	Industry CDF[1] ≥1%	Timely Detection of Performance Changes at the Plant Level
Flood	J1	YES	YES	NO (Trending Candidate)
High Energy Line Breaks	K	YES	NO	NO
Loss of Heat Sink	L	YES	YES	YES (Candidate RBPI)
Loss of Instrument Air	D1	YES	YES(Note 2)	NO (Trending Candidate)
Loss of MFW	P1	YES	YES	YES (Candidate RBPI)
Loss of Offsite Power	B1	YES	YES	NO (Trending Candidate)
Loss of Vital AC Bus	C1	YES	YES	NO (Trending Candidate)
Loss of Vital 125vdc Bus	C3	YES	YES	NO (Trending Candidate)
Loss of Service Water	E1	YES	YES	NO (Note 3)
Medium LOCA	G6	YES	YES	NO (Note 3)
Stuck Open Safety / Relief Valve	G2, G5	YES	YES	NO (Trending Candidate)
Transients	Q	NO (Note 5)	YES	YES (Candidate RBPI)
PWR INITIATOR	**NUREG/CR-5750 Initiator**	**CCDP≥1E-6**	**Industry CDF[1] ≥1%**	**Timely Detection of Performance Changes at the Plant Level**
Flood (Note 4)	J1	YES	YES	NO (Trending Candidate)
High Energy Line Breaks	K	YES	NO	NO
Large LOCA	G7	YES	YES	NO (Note 3)
Loss of Cooling Water	E1	YES	YES	NO (Note 3)
Loss of Heat Sink	L	YES	YES	YES (Candidate RBPI)
Loss of Instrument Air	D1	YES	YES(Note 2)	NO (Trending Candidate)
Loss of MFW	P1	YES	YES	YES (Candidate RBPI)
Loss of Offsite Power	B1	YES	YES	NO (Trending Candidate)
Loss of Vital 125vdc Bus	C3	YES	YES	NO (Trending Candidate)
Loss of Vital AC Bus	C1	YES	YES	NO (Trending Candidate)
Medium LOCA	G6	YES	YES	NO (Note 3)
Reactor Coolant Pump Seal LOCA	G8	YES	YES(Note 6)	NO (Note 3)
Small/Very-Small LOCA	G1, G3	YES	YES	NO (Trending Candidate)
Steam Generator Tube Rupture	F1	YES	YES	NO (Trending Candidate)
Transients	Q	YES	YES	YES (Candidate RBPI)

1. The 'Industry CDF' value was extracted from the IPE database. It is the summation of the initiator specific CDF contributions from all plants modeling that initiator in their IPE.
2. Several plants did not report CDF contribution by specific initiator but rather combined initiators into groups. In such instances initiator specific CDF contributions cannot be determined, however, industry CDF for this initiator is likely ≥ 1%.
3. To be selected for trending the candidate initiators must be risk-significant and actually occur in the industry (at least one occurrence since 1987 as recorded in NUREG/CR-5750). There were no occurrences of these initiators since 1987.
4. Industry flooding frequency dominated (80%) by single event at Surry
5. Transient initiators did not meet the CCDP criteria, however, their high occurrence frequency in conjunction with their nominal CCDP give them the ability to effect changes into the white and yellow performance bands. Therefore, transient initiators were included in the list of potentially risk-significant initiators.
6. Most RCP seal LOCAs modeled as consequential events.

A.1.2 Obtain Performance Data for Risk-Significant, Equipment-Related Elements

The analysis of initiating event data and calculation of initiating event frequencies also relied on several data sources. The three data sources used in the selection, and their contribution to the analysis, of initiating event RBPIs are described below:

NUREG/CR-5750, Rates of Initiating Events at U.S. Nuclear Power Plants: 1987 - 1995, presents an analysis of initiating event frequencies at U. S. nuclear power plants. This report provides two key sets of information essential to the RBPI process. One set of information consists of generic initiating event frequencies calculated for various initiators. These initiating event frequencies were incorporated into Standardized Plant Analysis Risk (SPAR, Ref. 6) models as part of the process of establishing plant-specific baseline core damage frequencies. Another set of information extracted from the report includes the definitions of initiators and related functional impact groupings. Use of these definitions ensure that initiating event frequencies calculated in future updates are comparable with those used in the baseline SPAR models.

The Sequence Coding and Search System (SCSS) is a database maintained at Oak Ridge National Laboratory that provides access to full text electronic copies of Licensee Event Reports (LERs) dating back to 1980. Per the Code of Federal Regulations 10CFR50.73, LERs are required each time a plant is scrammed. Therefore, LERs present a comprehensive set of data addressing plant scrams. Licensee Event Reports (LERs), accessed through the SCSS database, comprised the primary source of data used in identification of scrams and trips in NUREG/CR-5750. Similarly, this database will be used to identify trips and scrams used in future calculations of initiating event frequencies and corresponding RBPI thresholds.

Advanced features associated with the SCSS database allow screening on various coding schemes to greatly reduce the number of LERs that must be manually reviewed. Review by experienced engineers is then performed to screen and group the data by functional failures. The lag time between the occurrence of the event and its entry into the SCSS database is approximately 10 weeks. LERs can also be obtained directly from the NRC in hard copy form and reduce this process to approximately eight weeks.

Monthly Operating Reports (MORs) are summaries of operating experience that are filed with the NRC on a monthly basis. These reports contain detailed information on plant operation including hours that the reactor was critical and type, duration and cause of shutdowns and power reductions. This information is tabulated in various databases maintained at the INEEL.

Initiating event frequencies reported in NUREG/CR-5750 and subsequently incorporated into the SPAR models are reported in terms of per critical hour/year. Therefore, knowledge of plant-specific critical hour data is essential in calculating these values. NUREG/CR-5750 utilized one of the INEEL databases built on MOR data (MORP1) as the primary data source used in identification of critical hours. Similarly, this database will be used to identify critical hours used in future calculations of initiating event frequencies and corresponding RBPI thresholds.

A.1.3 Identify Indicators Capable of Detecting Performance Changes in a Timely Manner

In addition to being risk-significant (see Table A.1.1.1-1), initiating event performance indicators must be capable of detecting performance changes in a timely manner. An initiating event performance indicator involves collection of data during some monitoring period, and a decision rule, which declares that a plant is in a certain performance band based on observed data. This monitoring period must be long enough to reduce the probabilities of false negatives and false positives to acceptable levels, but no longer. When only one type of event is considered, such as initiating events, the decision rule is straightforward. It is to estimate the event occurrence rate, compare the estimate to the thresholds of the performance bands, and classify the plant accordingly. These analyses were performed with the results, including monitoring periods, documented in Appendices E and F.

In accordance with the preceding discussions, three initiating events/groups that met the criteria for risk-significance and timely monitoring were selected as candidates to be monitored as Initiating Event RBPIs. These initiators consist of Loss of Main Feedwater (LOFW), Loss of Heat Sink (LOHS), and General Transients (GT). These initiators met the criteria of risk-significance as outlined in section A.1.1. Monitoring periods of reasonable length were also calculated based on acceptable levels of false positives and negatives. Additionally, changes in their frequencies can be readily quantified with the current SPAR models. These three initiator categories account for over 90% of all reactor trips.

The remaining initiators identified in Table A.1.1.1-1 are not considered good candidates for initiating event RBPIs due to the excessive monitoring periods required to yield statistically significant trends in performance. However, because of their potential risk-significance, these initiators cannot be ignored. These initiators account for a very small fraction of the plant trips recorded in the industry yet they are significant contributors to industry risk associated with nuclear power plants. For example, Loss-of-Coolant-Accidents are postulated as significant contributors to risk yet only five LOCA events are identified between 1987 and 1998. These were all 'very-small' LOCAs. There has never been recorded a medium or large LOCA event in the U. S. nuclear power industry. While monitoring these events at the plant level is not practical, trending them at the industry-wide level may provide important insights.

A.1.3.1 Industry-wide Trending of Initiating Events

The RBPI development program also provides industry-wide trends of the initiating events that are RBPIs as well as risk-significant performance elements that are not possible to trend on a plant-specific basis. Since more data are available at the industry level, trends emerging at the industry level may be apparent before plant-specific changes can be determined. The Loss of Offsite Power (LOOP) initiator is an example of a performance element that is difficult to trend at a plant-specific level yet will yield valuable information at the industry level. The IPE results indicate that LOOP is the dominant contributor to risk at U.S. nuclear power plants, however, plant-specific performance indicators are not practicable because of the excessive period required to monitor this initiator.

Initiators evaluated as Accident Sequence Precursors (ASP) will also be trended on an industry-wide basis. ASP events are a set of precursor events screened from the industry that have an increased potential for severe core damage. Trending of these events provides a better understanding of the risk-significant events occurring at U.S. commercial reactors. The Annual ASP Index for initiating events was selected as the figure of merit to trend. This index is based on the sum of the CCDPs of at power precursors involving initiating events divided by the number of reactor operating years.

To be selected for trending the candidate initiators must be risk-significant and actually occur in the industry (at least one occurrence since 1987 as recorded in NUREG/CR-5750). Thirteen initiating event types/groups meet these conditions and are identified as candidates for industry-wide trending. These initiating event types/groups and their respective NUREG/CR-5750 category are listed below:

Internal Flood (J1)
General Transients (Q)
Stuck Open Safety / Relief Valve - BWR (G2)
Initiators Evaluated as Accident Sequence Precursors (ASP)
Loss of Feedwater Initiators (P1)
Loss of Heat Sink Initiators (L)
Loss of Instrument/Control Air - BWR (D1)
Loss of Instrument/Control Air - PWR (D1)
Loss of Offsite Power Events (B1)
Loss of Vital AC Bus (C1, C2)
Loss of Vital DC Bus (C3)
Small/Very Small LOCA (G1, G3)
Steam Generator Tube Rupture (F1)

The Initiating Event RBPIs (General Transients, Loss of Feedwater, and Loss of Heat Sink) are trended in Table 5.3-6 of the main body of the report. Trends associated with non-RBPI events are shown below in Figures A.1.3.1-1 and A.1.3.1-10.

A.1.4 Identify Performance Thresholds Consistent with a Graded Approach to Performance Evaluation from SECY 99-007

A graded approach to identifying performance thresholds is built around four performance bands (green, white, yellow, red) whose boundaries correspond to plant-specific changes in CDF equal to 1E-6/yr, 1E-5/yr and 1E-4/yr. The two higher level thresholds (ΔCDF = 1E-5/yr and 1E-4/yr) were set in accordance with acceptance guidelines outlined in Regulatory Guide 174 (Ref. 7).

SECY 99-007 proposed a lower level threshold determined by choosing a value to no more than two significant figures such that about 95% of the plants would have observed data values that would be in the green zone. This process establishes a generic value that is applied to each plant. The weakness of this method is that it depends only on the number of plants with less than acceptable performance but not on how much their performance exceeds the norm (i.e., actual risk). Additionally, due to the large plant-to-plant variability in the importance of systems, this value correlates to changes in CDF in excess of 1E-5/year at some plants. After considerable analysis, the alternative lower level threshold (green/white) of ΔCDF = 1E-6/yr was chosen. This value is consistent with the order of magnitude decrements associated with the higher level thresholds. It is also consistent with the green/white interval associated with inspection findings evaluated in the Significance Determination Process (SDP).

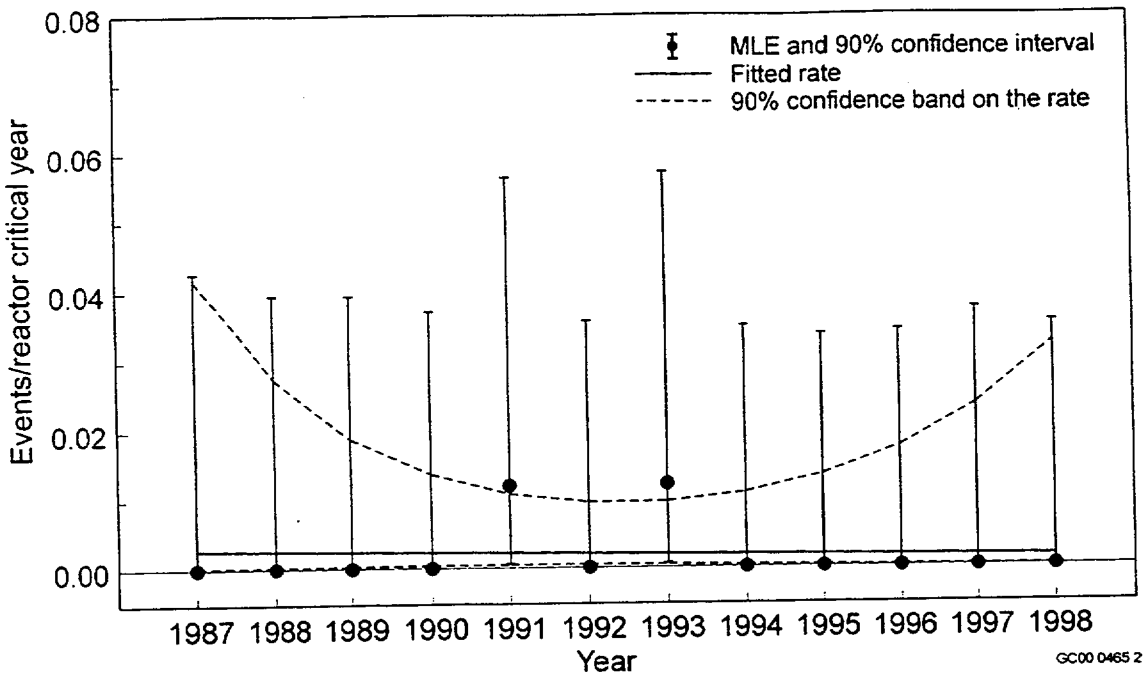

Figure A.1.3.1-1 Time-dependent Trending of Internal Flood Initiating Events (Trend is not close to statistically significant, p-value = 0.8)

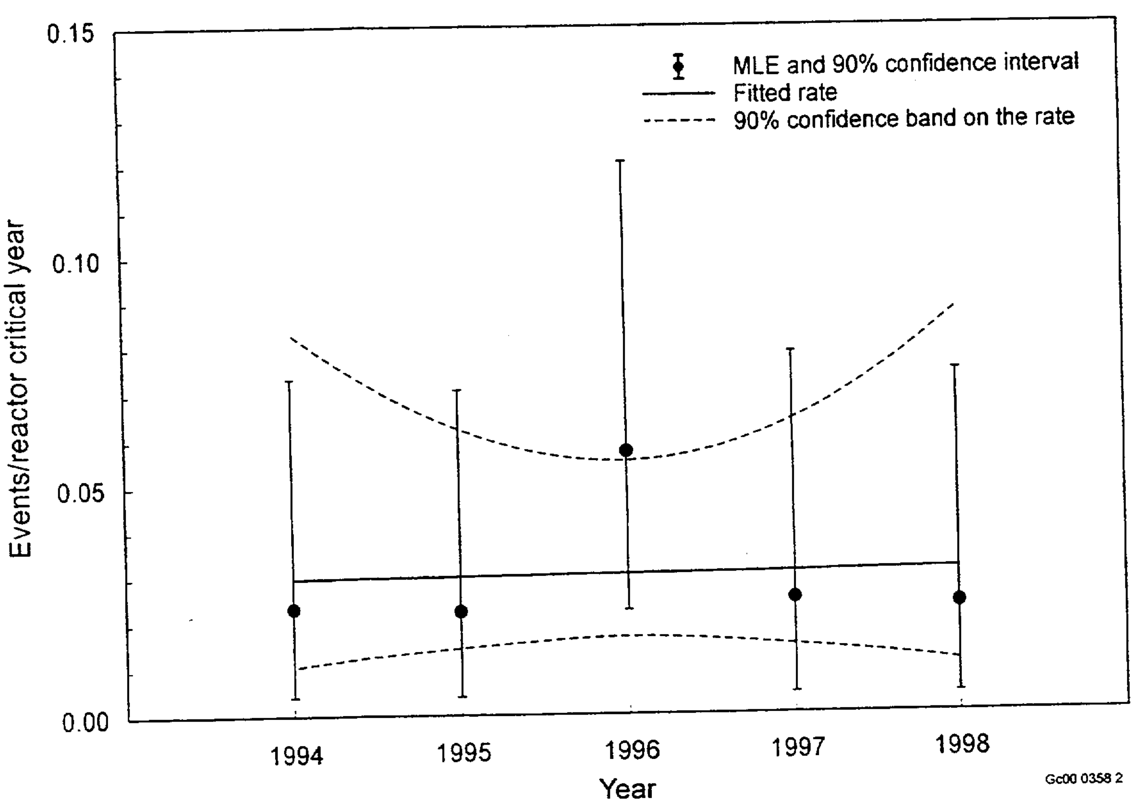

Figure A.1.3.1-2 Time-dependent Trending of Annual Initiating Event ASP Index (The trend is not close to statistically significant, p-value = 0.9)

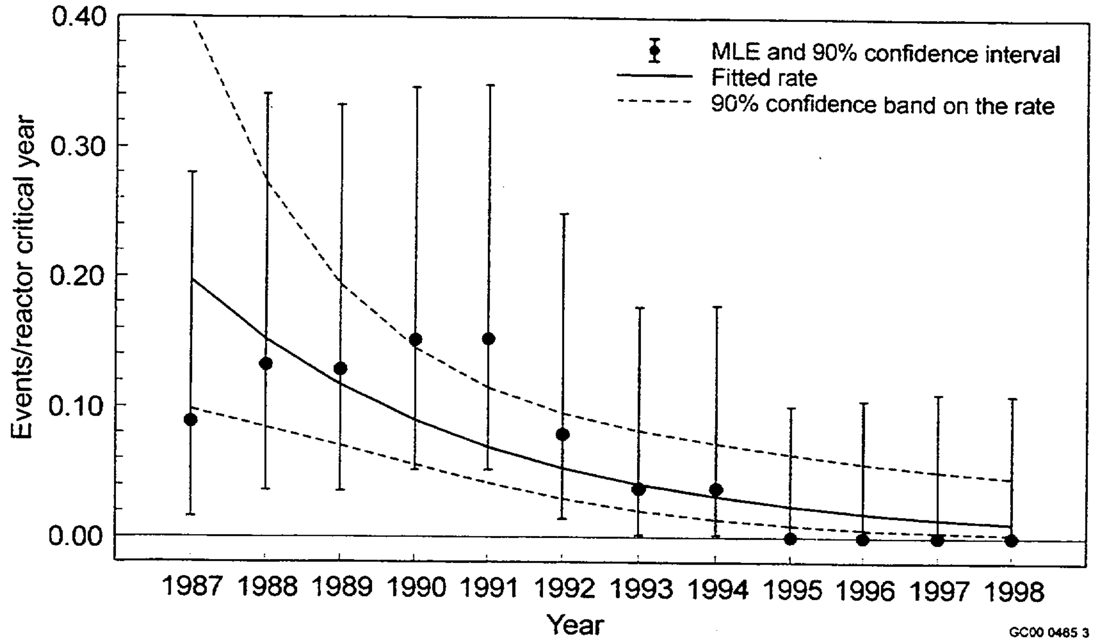

Figure A.1.3.1-3 Time-dependent Trending of Loss of Instrument/Control Air (BWR) Initiators (The trend is statistically very significant, p-value = 0.0016)

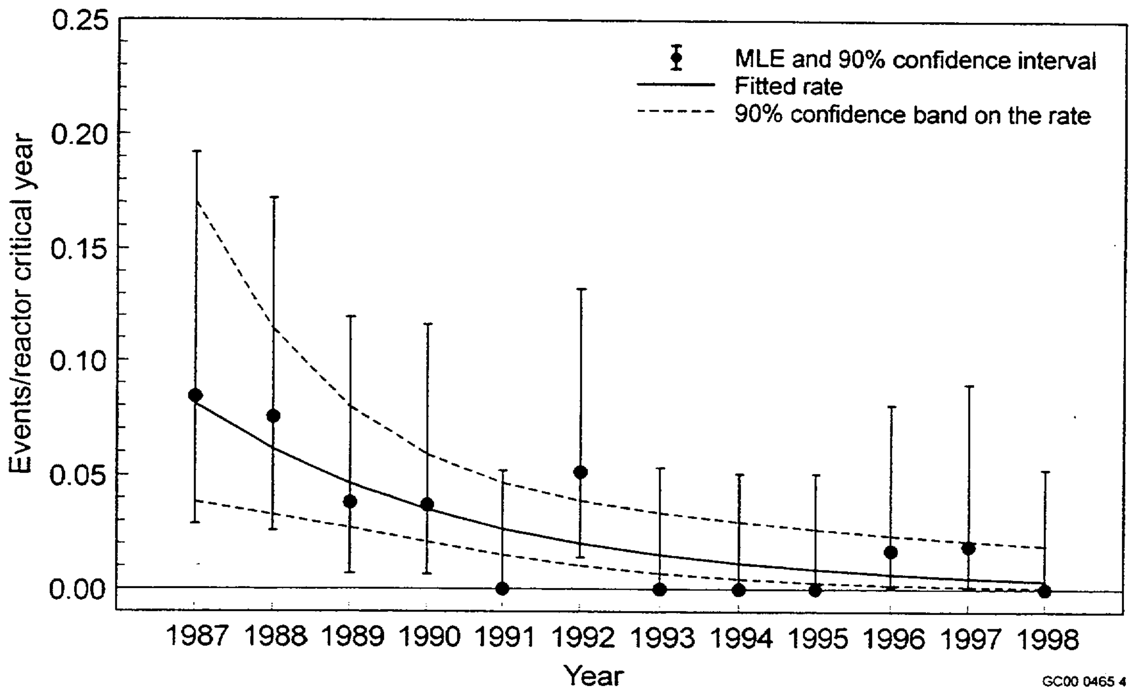

Figure A.1.3.1-4 Time-dependent Trending of Loss of Instrument/Control Air (PWR) Initiators (The trend is statistically very significant, p-value = 0.0016)

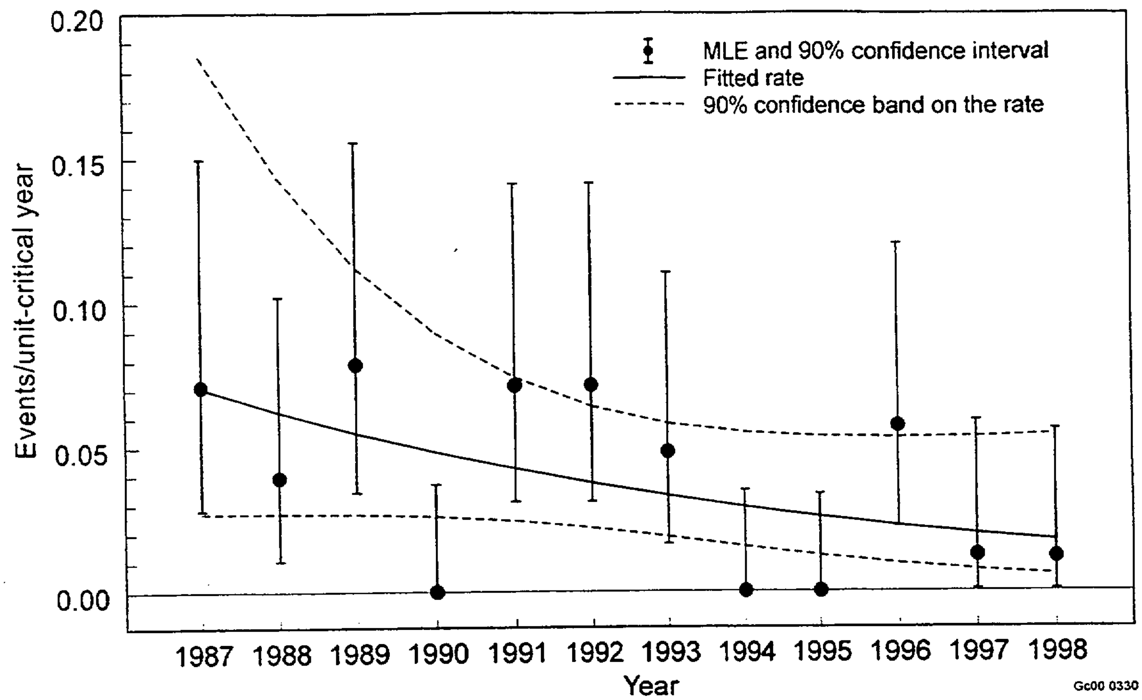

Figure A.1.3.1-5 Time-dependent Trending of Loss of Offsite Power Initiating Events (When extra-Poisson scatter is accounted for, trend is not statistically significant, p-value = 0.10)

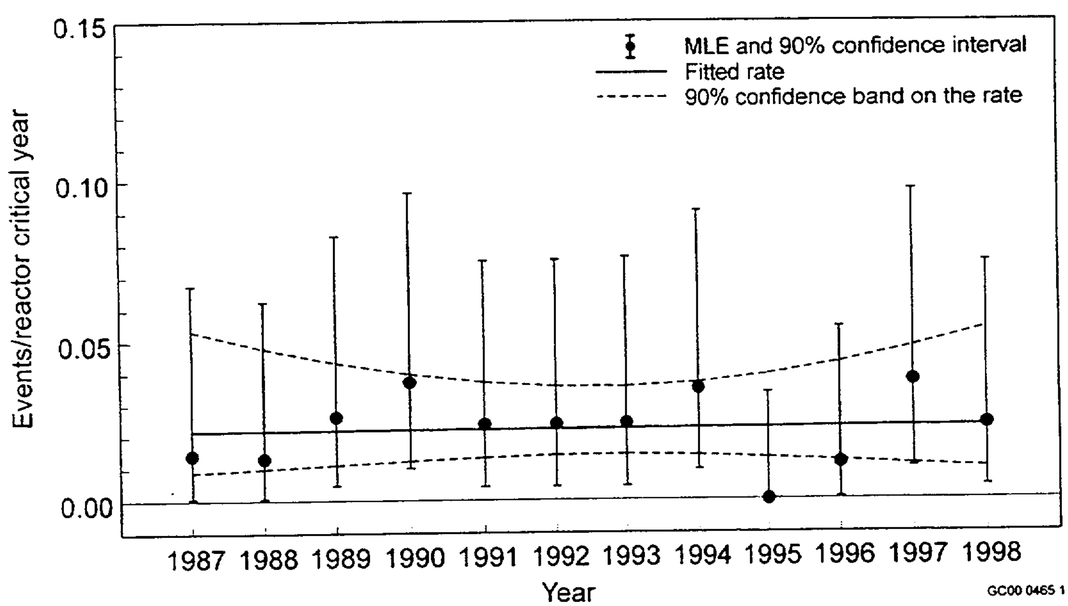

Figure A.1.3.1-6 Time-dependent Trending of Loss of Safety Related Vital AC Bus Initiators (Trend is not close to statistically significant, p-value = 0.9)

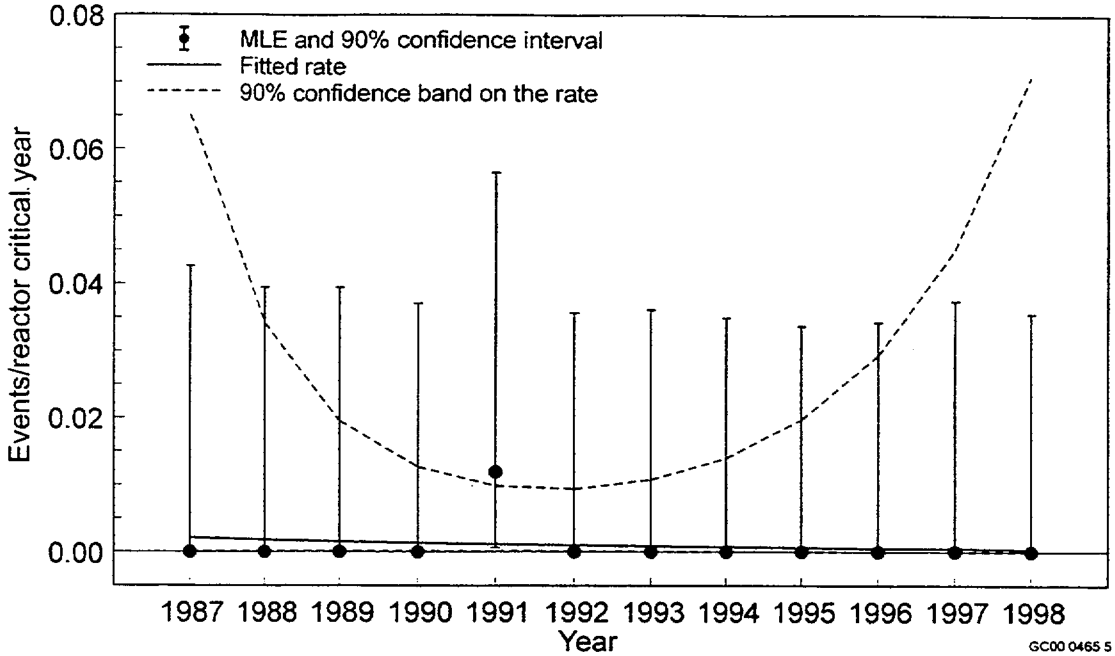

Figure A.1.3.1-7 Time-dependent Trending of Loss of Safety Related Vital DC Bus Initiators (Trend is not close to statistically significant, p-value = 0.6)

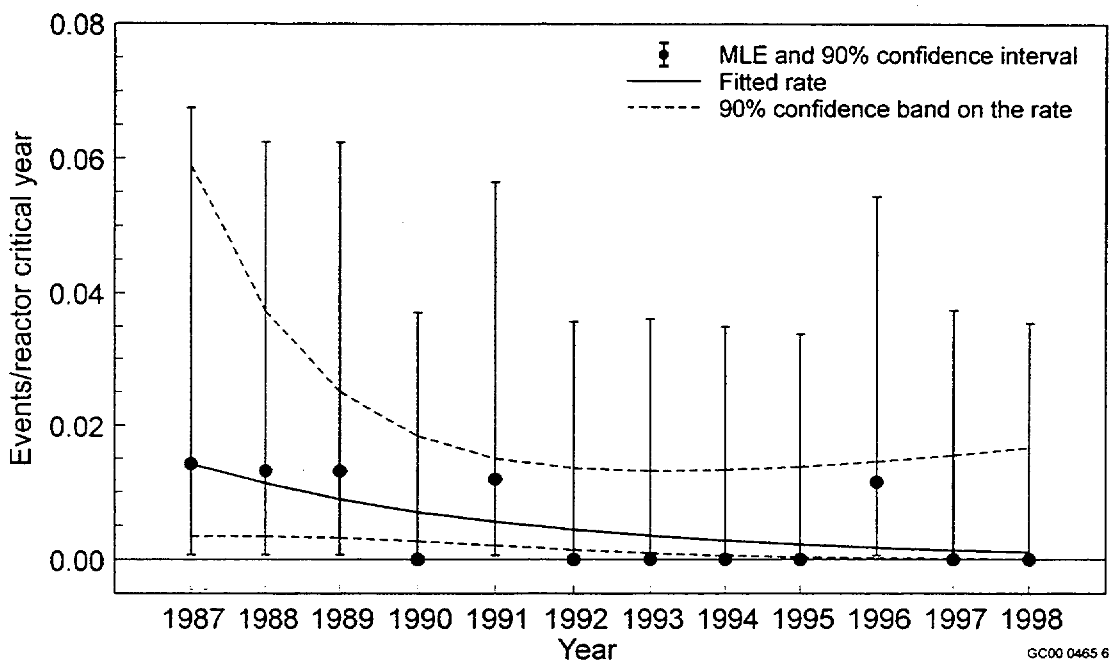

Figure A.1.3.1-8 Time-dependent Trending of Loss of Small/Very Small LOCA Initiators (Trend is not statistically significant, p-value = 0.17)

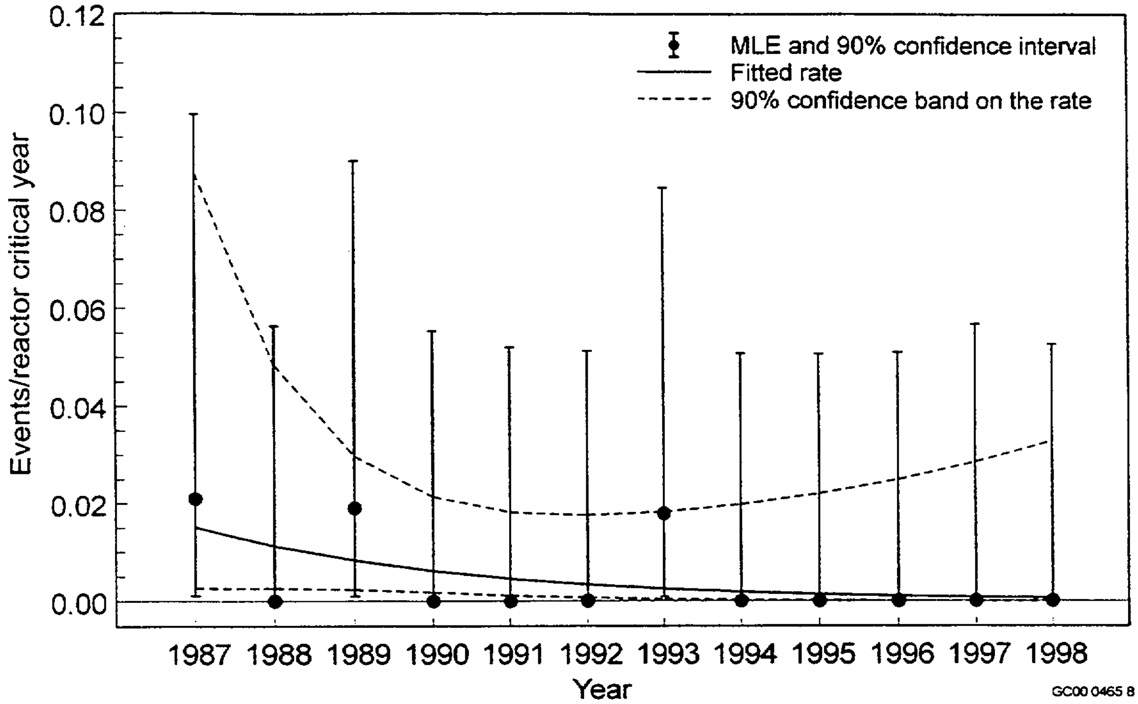

Figure A.1.3.1-9 Time-dependent Trending of Steam Generator Tube Rupture Initiators (Trend is not statistically significant, p-value = 0.17)

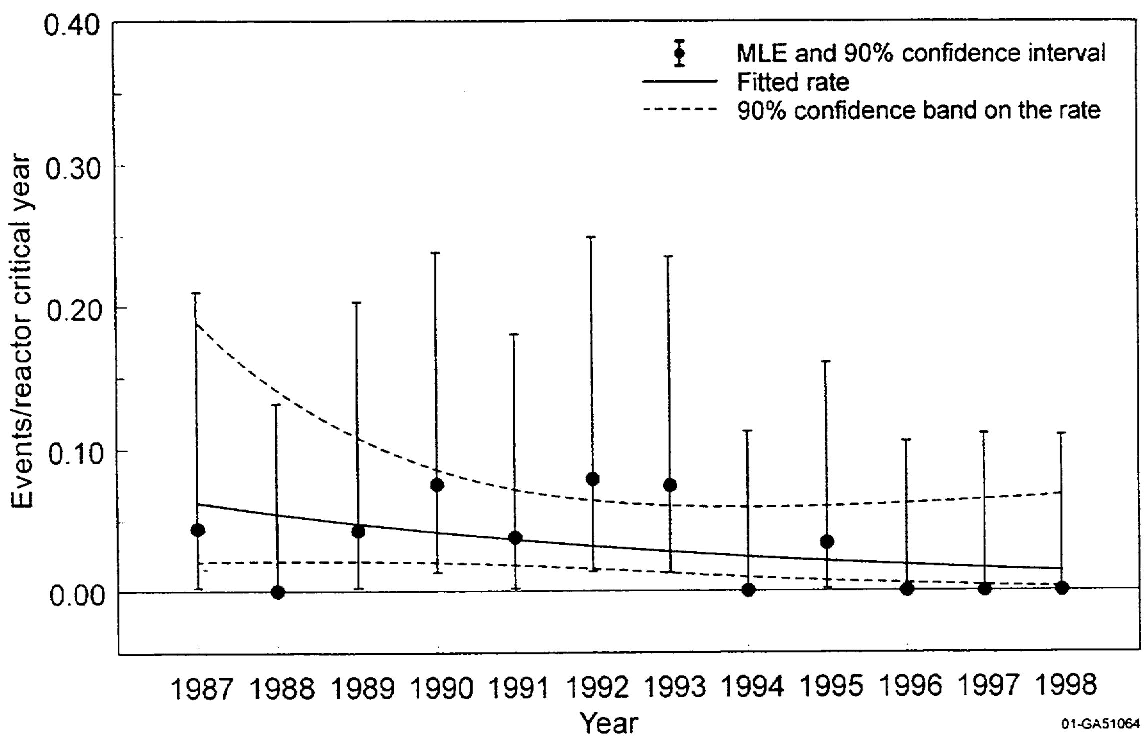

Figure A.1.3.1-10 Time-dependent Trending of Stuck Open Safety/Relief Valve - BWR Initiating Event (Trend is not statistically significant, p-value = 0.16)

To evaluate changes in performance as well as current thresholds and future performance trends, a fixed reference point (i.e., performance baseline) corresponding to current nominal performance is required. To facilitate plant-specific threshold values, a 'baseline' model was constructed for each plant analyzed in the RBPI program. Plant-specific logic (i.e., the SPAR models) was used to allow plant-specific design and operational characteristics to be credited. These models were 'baselined' to 1996 performance by incorporating appropriate unavailability data from the World Association of Nuclear Operators (WANO, Ref. 8), and reliability data from the system reliability studies (References 9, 10, 11, 12, 13, and 14). (Note: EPIX/RADs (References 15 and 16) will provide the failure data used in future performance trending and was the preferred data source for the baseline models.) In some cases minor modifications to the logic were also made to ensure that the logic structure of the models matched the available data.

An iterative technique is employed to determine the exact thresholds. The frequency of the initiator is increased until the plant core damage frequency increases by an amount correlating to the performance action bands limits (i.e., 1E-6, 1E-5, 1E-4). Calculation of the Transient initiating event thresholds is straightforward using this process. Calculation of the LOFW and LOHS initiating event thresholds is obtained in a similar fashion, however, the process is somewhat more complex since they are conditional events within the Transient event tree and do not have their own explicit event trees.

Initiating event RBPIs were selected and their threshold values calculated for 30 sites (44 plants). These sites are comprised of 19 BWR and 25 PWR plants. Detailed threshold information for each analyzed plant is contained in Tables A.1.4-1 through A.1.4-30.

A.1.5 Inspection Areas Covered by New RBPIs

The RBPIs developed in this report for the initiating events cornerstone were compared with the performance indicators in the ROP to identify those RBPIs that are not currently in the ROP. The inspection areas that could be impacted by the new initiating event RBPIs were then determined. The results are summarized in Table A.1.5-1.

Table A.1.4-1 BWR 123 Plant 1/2 Initiating Events Threshold Summary

BWR 123 Plant 1/2 SPAR 3i (3.7E-9/hr, 2.6E-5/calendar year[1]) BWR 123 Plant 1/2 RBPI Baseline (2.9E-9/hr, 2.0E-5/calendar year[1])					
Initiator	Baseline Initiator Frequency (NUREG/CR-5750)	Green/White Threshold 95^{th}%ile	Green/White Threshold (ΔCDF =1E-6/year)	White/Yellow Threshold (ΔCDF =1E-5/year)	Yellow/Red Threshold (ΔCDF =1E-4/year)
Transient Initiator	1.2 / calendar year	2.2 / calendar year	1.9 / calendar year	8.4 / calendar year	73 / calendar year
Loss of Feedwater	6.8E-2 / calendar year	2.0E-1 / calendar year	1.3E-1 / calendar year	1.1 / calendar year	11 / calendar year
Loss of Heat Sink	2.3E-1 / calendar year	3.1E-1 / calendar year	3.9E-1 / calendar year	2.7 / calendar year	25 / calendar year

1. Calendar year is defined as 7000 critical hours.

Table A.1.4-2 BWR 3/4 Plant 1 Initiating Events Threshold Summary

BWR 3/4 Plant 1 SPAR 3i (3.5E-10/hr, 2.4E-6/calendar year[1]) BWR 3/4 Plant 1 RBPI BASELINE (3.1E-10/hr, 2.2E-6/calendar year[1])					
Initiator	Baseline Initiator Frequency (NUREG/CR-5750)	Green/White Threshold 95^{th}%ile	Green/White Threshold (ΔCDF =1E-6/year)	White/Yellow Threshold (ΔCDF =1E-5/year)	Yellow/Red Threshold (ΔCDF =1E-4/year)
Transient Initiator	1.2 / calendar year	2.2 / calendar year	2.4 / calendar year	13 / calendar year	120 / calendar year
Loss of Feedwater	6.8E-2 / calendar year	2.0E-1 / calendar year	3.4E-1 / calendar year	3.2 / calendar year	32 / calendar year
Loss of Heat Sink	2.3E-1 / calendar year	3.1E-1 / calendar year	4.6E-1 / calendar year	3.3 / calendar year	32 / calendar year

1. Calendar year is defined as 7000 critical hours.

Table A.1.4-3 BWR 3/4 Plant 2 Initiating Events Threshold Summary

BWR 3/4 Plant 2 SPAR 3i (5.1E-10/hr, 3.5E-6/calendar year[1]) BWR 3/4 Plant 2 RBPI Baseline (3.8E-10/hr, 2.7E-6/calendar year[1])					
Initiator	Baseline Initiator Frequency (NUREG/CR-5750)	Green/White Threshold 95th%ile	Green/White Threshold (ΔCDF =1E-6/year)	White/Yellow Threshold (ΔCDF =1E-5/year)	Yellow/Red Threshold (ΔCDF =1E-4/year)
Transient Initiator	1.2 / calendar year	2.2 / calendar year	2.5 / calendar year	14.3 / calendar year	126 / calendar year
Loss of Feedwater	6.8E-2 / calendar year	2.0E-1 / calendar year	3.6E-1 / calendar year	3.5 / calendar year	34 / calendar year
Loss of Heat Sink	2.3E-1 / calendar year	3.1E-1 / calendar year	4.8E-1 / calendar year	3.5 / calendar year	33 / calendar year

1. Calendar year is defined as 7000 critical hours.

Table A.1.4-4 BWR 3/4 Plant 3/4 Initiating Events Threshold Summary

BWR 3/4 Plant 3/4 SPAR 3i (3.5E-9/hr, 2.4E-5/calendar year[1]) BWR 3/4 Plant 3/4 RBPI Baseline (2.2E-9/hr, 1.5E-5/calendar year[1])					
Initiator	Baseline Initiator Frequency (NUREG/CR-5750)	Green/White Threshold 95th%ile	Green/White Threshold (ΔCDF =1E-6/year)	White/Yellow Threshold (ΔCDF =1E-5/year)	Yellow/Red Threshold (ΔCDF =1E-4/year)
Transient Initiator	1.2 / calendar year	2.2 / calendar year	1.5 / calendar year	3.7 / calendar year	26 / calendar year
Loss of Feedwater	6.8E-2 / calendar year	2.0E-1 / calendar year	1.0 E-1 / calendar year	7.0E-1 / calendar year	6.8 / calendar year
Loss of Heat Sink	2.3E-1 / calendar year	3.1E-1 / calendar year	2.0E-1 / calendar year	8.0E-1 / calendar year	7.1 / calendar year

1. Calendar year is defined as 7000 critical hours.

Table A.1.4-5 BWR 3/4 Plant 5 Initiating Events Threshold Summary

BWR 3/4 Plant 5 SPAR 3i (2.0E-9/hr, 1.4E-5/calendar year[1]) BWR 3/4 Plant 5 RBPI Baseline (2.2E-9/hr, 1.5E-5/calendar year[1])					
BWR Initiator	Baseline IE Frequency (NUREG/CR-5750)	Green/White Threshold 95^{th}%ile	Green/White Threshold (ΔCDF =1E-6/year)	White/Yellow Threshold (ΔCDF =1E-5/year)	Yellow/Red Threshold (ΔCDF =1E-4/year)
Transient Initiator	1.2 / calendar year	2.2 / calendar year	1.5 / calendar year	3.9 / calendar year	28 / calendar year
Loss of Feedwater	6.8E-2 / calendar year	2.0E-1 / calendar year	1.4E-1 / calendar year	8.0E-1 / calendar year	8.0 / calendar year
Loss of Heat Sink	2.3E-1 / calendar year	3.1E-1 / calendar year	3.0E-1 / calendar year	9.6E-1 / calendar year	8.0 / calendar year

1. Calendar year is defined as 7000 critical hours.

Table A.1.4-6 BWR 3/4 Plant 6 Initiating Events Threshold Summary

BWR 3/4 Plant 6 SPAR 3i (2.8E-9/hr, 2.0E-5/calendar year[1]) BWR 3/4 Plant 6 RBPI Baseline (2.4E-9/hr, 1.7E-5/calendar year[1])					
BWR Initiator	Baseline IE Frequency (NUREG/CR-5750)	Green/White Threshold 95^{th}%ile	Green/White Threshold (ΔCDF =1E-6/year)	White/Yellow Threshold (ΔCDF =1E-5/year)	Yellow/Red Threshold (ΔCDF =1E-4/year)
Transient Initiator	1.2 / calendar year	2.2 / calendar year	1.5 / calendar year	3.9 / calendar year	30 / calendar year
Loss of Feedwater	6.8E-2 / calendar year	2.0E-1 / calendar year	1.5E-1 / calendar year	9.6E-1 / calendar year	8.8 / calendar year
Loss of Heat Sink	2.3E-1 / calendar year	3.1E-1 / calendar year	3.0E-1 /calendar year	9.4E-1 / calendar year	8.0 / calendar year

1. Calendar year is defined as 7000 critical hours.

Table A.1.4-7 BWR 3/4 Plant 8 Initiating Events Threshold Summary

BWR 3/4 Plant 8 SPAR 3i (8.7E-10/hr, 6.1E-6/calendar year[1]) BWR 3/4 Plant 8 RBPI Baseline (7.6E-10/hr, 5.3E-6/calendar year[1])					
BWR Initiator	Baseline IE Frequency (NUREG/CR-5750)	Green/White Threshold 95th%ile	Green/White Threshold (ΔCDF =1E-6/year)	White/Yellow Threshold (ΔCDF =1E-5/year)	Yellow/Red Threshold (ΔCDF =1E-4/year)
Transient Initiator	1.2 / calendar year	2.2 / calendar year	3.1 / calendar year	13 / calendar year	113 / calendar year
Loss of Feedwater	6.8E-2 / calendar year	2.0E-1 / calendar year	2.6E-1 / calendar year	2.0 / calendar year	19 / calendar year
Loss of Heat Sink	2.3E-1 / calendar year	3.1E-1 / calendar year	4.2E-1 / calendar year	2.1 / calendar year	19 / calendar year

1. Calendar year is defined as 7000 critical hours.

Table A.1.4-8 BWR 3/4 Plant 11 Initiating Events Threshold Summary

BWR 3/4 Plant 11 SPAR 3i (4.9E-9/hr, 3.4E-5/calendar year[1]) BWR Plant 11 RBPI Baseline (5.6E-9/hr, 3.9E-5/calendar year[1])					
BWR Initiator	Baseline IE Frequency (NUREG/CR-5750)	Green/White Threshold 95th%ile	Green/White Threshold (ΔCDF =1E-6/year)	White/Yellow Threshold (ΔCDF =1E-5/year)	Yellow/Red Threshold (ΔCDF =1E-4/year)
Transient Initiator	1.2 / calendar year	2.2 / calendar year	1.3 / calendar year	2.4 / calendar year	14 / calendar year
Loss of Feedwater	6.8E-2 / calendar year	2.0E-1 / calendar year	1.0E-1 / calendar year	4.3E-1 / calendar year	3.8 / calendar year
Loss of Heat Sink	2.3E-1 / calendar year	3.1E-1 / calendar year	2.6E-1 / calendar year	6.0E-1 / calendar year	4.0 / calendar year

1. Calendar year is defined as 7000 critical hours.

Table A.1.4-9 BWR 3/4 Plant 12/13 Initiating Events Threshold Summary

BWR 3/4 Plant 12/13 SPAR 3i (4.2E-9/hr, 3.0E-5/calendar year[1]) BWR 3/4 Plant 12/13 RBPI Baseline (3.4E-9/hr, 2.4E-5/calendar year[1])					
Initiator	Baseline Initiator Frequency (NUREG/CR-5750)	Green/White Threshold 95^{th}%ile	Green/White Threshold (ΔCDF =1E-6/year)	White/Yellow Threshold (ΔCDF =1E-5/year)	Yellow/Red Threshold (ΔCDF =1E-4/year)
Transient Initiator	1.2 / calendar year	2.2 / calendar year	1.3 / calendar year	2.0 / calendar year	9.4 / calendar year
Loss of Feedwater	6.8E-2 / calendar year	2.0E-1 / calendar year	5.0E-2 / calendar year	2.4E-1 / calendar year	2.0 / calendar year
Loss of Heat Sink	2.3E-1 / calendar year	3.1E-1 / calendar year	1.7E-1 / calendar year	3.6E-1 / calendar year	2.2 / calendar year

1. Calendar year is defined as 7000 critical hours.

Table A.1.4-10 BWR 3/4 Plant 15/16 Initiating Events Threshold Summary

BWR 3/4 Plant 15/16 SPAR 3i (5.8E-10/hr, 4.1E-6/calendar year[1]) BWR 3/4 Plant 15/16 RBPI Baseline (5.3E-10/hr, 3.7E-6/calendar year[1])					
BWR Initiator	Baseline IE Frequency (NUREG/CR-5750)	Green/White Threshold 95^{th}%ile	Green/White Threshold (ΔCDF =1E-6/year)	White/Yellow Threshold (ΔCDF =1E-5/year)	Yellow/Red Threshold (ΔCDF =1E-4/year)
Transient Initiator	1.2 / calendar year	2.2 / calendar year	2.1 / calendar year	10 / calendar year	90 / calendar year
Loss of Feedwater	6.8E-2 / calendar year	2.0E-1 / calendar year	3.1E-1 / calendar year	2.6 / calendar year	25 / calendar year
Loss of Heat Sink	2.3E-1 / calendar year	3.1E-1 / calendar year	4.8E-1 / calendar year	2.6 / calendar year	25 / calendar year

1. Calendar year is defined as 7000 critical hours.

Table A.1.4-11 BWR 3/4 Plant 18/19 Initiating Events Threshold Summary

BWR 3/4 Plant 18/19 SPAR 3i (3.7E-9/hr, 2.6E-5/calendar year[1]) BWR 3/4 Plant 18/19 RBPI Baseline (2.9E-9/hr, 2.0E-5/calendar year[1])					
BWR Initiator	Baseline IE Frequency (NUREG/CR-5750)	Green/White Threshold 95th%ile	Green/White Threshold (ΔCDF =1E-6/year)	White/Yellow Threshold (ΔCDF =1E-5/year)	Yellow/Red Threshold (ΔCDF =1E-4/year)
Transient Initiator	1.2 / calendar year	2.2 / calendar year	1.9 / calendar year	7.8 /calendar year	67 /calendar year
Loss of Feedwater	6.8E-2 / calendar year	2.0E-1 / calendar year	3.0E-1 / calendar year	2.5 / calendar year	24 / calendar year
Loss of Heat Sink	2.3E-1 / calendar year	3.1E-1 / calendar year	4.1E-1 / calendar year	3.4 / calendar year	33 / calendar year

1. Calendar year is defined as 7000 critical hours.

Table A.1.4-12 BWR 5/6 Plant 2 Initiating Events Threshold Summary

BWR 5/6 Plant 2 SPAR 3i (1.2E-9/hr, 8.6E-6/calendar year[1]) BWR 5/6 Plant 2 RBPI Baseline (1.4E-9/hr, 9.9E-6/calendar year[1])					
BWR Initiator	Baseline IE Frequency (NUREG/CR-5750)	Green/White Threshold 95th%ile	Green/White Threshold (ΔCDF =1E-6/year)	White/Yellow Threshold (ΔCDF =1E-5/year)	Yellow/Red Threshold (ΔCDF =1E-4/year)
Transient Initiator	1.2 / calendar year	2.2 / calendar year	1.8 / calendar year	7.2 / calendar year	60 / calendar year
Loss of Feedwater	6.8E-2 / calendar year	2.0E-1 / calendar year	2.2E-1 / calendar year	1.7 / calendar year	16 / calendar year
Loss of Heat Sink	2.3E-1 / calendar year	3.1E-1 / calendar year	3.8E-1 / calendar year	1.8 / calendar year	16 / calendar year

1. Calendar year is defined as 7000 critical hours.

Table A.1.4-13 BWR 5/6 Plant 5 Initiating Events Threshold Summary

BWR 5/6 Plant 5 SPAR 3i (1.5E-9/hr, 1.1E-5/calendar year[1]) BWR 5/6 Plant 5 RBPI Baseline (1.2E-9/hr, 8.5E-6/calendar year[1])					
Initiator	Baseline Initiator Frequency (NUREG/CR-5750)	Green/White Threshold 95th%ile	Green/White Threshold (ΔCDF =1E-6/year)	White/Yellow Threshold (ΔCDF =1E-5/year)	Yellow/Red Threshold (ΔCDF =1E-4/year)
Transient Initiator	1.2 / calendar year	2.2 / calendar year	2.3 / calendar year	12 / calendar year	106 / calendar year
Loss of Feedwater	6.8E-2 / calendar year	2.0E-1 / calendar year	2.4E-1 / calendar year	2.2 / calendar year	22 / calendar year
Loss of Heat Sink	2.3E-1 / calendar year	3.1E-1 / calendar year	5.2E-1 / calendar year	3.9 / calendar year	37 / calendar year

1. Calendar year is defined as 7000 critical hours.

Table A.1.4-14 BWR 5/6 Plant 8 Initiating Events Threshold Summary

BWR 5/6 Plant 8 SPAR 3i (3.7E-9/hr, 2.6E-5/calendar year[1]) BWR 5/6 Plant 8 RBPI Baseline (7.9E-9/hr, 5.6E-5/calendar year[1])					
Initiator	Baseline Initiator Frequency (NUREG/CR-5750)	Green/White Threshold 95th%ile	Green/White Threshold (ΔCDF =1E-6/year)	White/Yellow Threshold (ΔCDF =1E-5/year)	Yellow/Red Threshold (ΔCDF =1E-4/year)
Transient Initiator	1.2 / calendar year	2.2 / calendar year	1.3 / calendar year	1.5 / calendar year	3.9 / calendar year
Loss of Feedwater	6.8E-2 / calendar year	2.0E-1 / calendar year	4.0E-2 / calendar year	1.0E-1 / calendar year	7.5E-1 / calendar year
Loss of Heat Sink	2.3E-1 / calendar year	3.1E-1 / calendar year	1.5E-1 / calendar year	2.2E-1 / calendar year	8.6E-1 / calendar year

1. Calendar year is defined as 7000 critical hours.

Table A.1.4-15 B&W Plant 3 Initiating Events Threshold Summary

B&W Plant 3 SPAR 3i (2.2E-9/hr, 1.6E-5/calendar year[1]) B&W Plant 3 Baseline (2.3E-9/hr, 1.6E-5/calendar year[1])					
Initiator	Baseline Initiator Frequency (NUREG/CR-5750)	Green/White Threshold 95^{th}%ile	Green/White Threshold (ΔCDF =1E-6/year)	White/Yellow Threshold (ΔCDF =1E-5/year)	Yellow/Red Threshold (ΔCDF =1E-4/year)
Transient Initiator	9.6E-1 / calendar year	1.8 / calendar year	2.4 / calendar year	14 / calendar year	134 / calendar year
Loss of Feedwater	6.8E-2 / calendar year	2.0E-1 / calendar year	2.5E-1 / calendar year	2.4 / calendar year	22 / calendar year
Loss of Heat Sink	9.6E-2 / calendar year	2.6E-1 / calendar year	1.1E-1 / calendar year	5.3E-1 / calendar year	4.7 / calendar year

1. Calendar year is defined as 7000 critical hours.

Table A.1.4-16 B&W Plant 4/5/6 Initiating Events Threshold Summary

B&W Plant 4/5/6 SPAR 3i (2.1E-9/hr, 1.5E-5/calendar year[1]) B&W Plant 4/5/6 RBPI Baseline (2.5E-9/hr, 1.7E-5/calendar year[1])					
PWR Initiator	Baseline IE Frequency (NUREG/CR-5750)	Green/White Threshold 95^{th}%ile	Green/White Threshold (ΔCDF =1E-6/year)	White/Yellow Threshold (ΔCDF =1E-5/year)	Yellow/Red Threshold (ΔCDF =1E-4/year)
Transient Initiator	9.6E-1 / calendar year	1.8 / calendar year	1.2 / calendar year	2.7 / calendar year	17 / calendar year
Loss of Feedwater	6.8E-2 / calendar year	2.0E-1 / calendar year	2.0E-1 / calendar year	1.9 / calendar year	17 / calendar year
Loss of Heat Sink	9.6E-2 / calendar year	2.6E-1 / calendar year	1.0E-1 / calendar year	4.0E-1 / calendar year	3.4 / calendar year

1. Calendar year is defined as 7000 critical hours.

Table A.1.4-17 B&W Plant 7 Initiating Events Threshold Summary

B&W Plant 7 SPAR 3i (1.9E-9/hr, 1.4E-5/calendar year[1]) B&W Plant 7 RBPI Baseline (1.8E-9/hr, 1.2E-5/calendar year[1])					
Initiator	Baseline Initiator Frequency (NUREG/CR-5750)	Green/White Threshold 95th%ile	Green/White Threshold (ΔCDF =1E-6/year)	White/Yellow Threshold (ΔCDF =1E-5/year)	Yellow/Red Threshold (ΔCDF =1E-4/year)
Transient Initiator	9.6E-1 / calendar year	1.8 / calendar year	5.4 / calendar year	45 / calendar year	438 / calendar year
Loss of Feedwater	6.8E-2 / calendar year	2.0E-1 / calendar year	1.1 / calendar year	10 / calendar year	102 / calendar year
Loss of Heat Sink	9.6E-2 / calendar year	2.6E-1 / calendar year	3.0E-1 / calendar year	2.5 / calendar year	24 / calendar year

1. Calendar year is defined as 7000 critical hours.

Table A.1.4-18 CE Plant 1 Initiating Events Threshold Summary

CE Plant 1 SPAR 3i (4.2E-9/hr, 3.0E-5/calendar year[1]) CE Plant 1 RBPI Baseline (4.2E-9/hr, 3.0E-5/calendar year[1])					
Initiator	Baseline Initiator Frequency (NUREG/CR-5750)	Green/White Threshold 95th%ile	Green/White Threshold (ΔCDF =1E-6/year)	White/Yellow Threshold (ΔCDF =1E-5/year)	Yellow/Red Threshold (ΔCDF =1E-4/year)
Transient Initiator	9.6E-1 / calendar year	1.8 / calendar year	3.2 / calendar year	23.0 / calendar year	222 / calendar year
Loss of Feedwater	6.8E-2 / calendar year	2.0E-1 / calendar year	7.3E-1 / calendar year	7.2 / calendar year	71.2 / calendar year
Loss of Heat Sink	9.6E-2 / calendar year	2.6E-1 / calendar year	2.6E-1 / calendar year	2.0 / calendar year	19.2 / calendar year

1. Calendar year is defined as 7000 critical hours.

Table A.1.4-19 CE Plant 2/3 Initiating Events Threshold Summary

CE Plant 2/3 SPAR 3i (2.6E-9/hr, 1.8E-5/calendar year[1]) CE Plant 2/3 RBPI Baseline (2.1E-9/hr, 1.4E-5/calendar year[1])					
PWR Initiator	Baseline IE Frequency (NUREG/CR-5750)	Green/White Threshold 95th%ile	Green/White Threshold (ΔCDF =1E-6/year)	White/Yellow Threshold (ΔCDF =1E-5/year)	Yellow/Red Threshold (ΔCDF =1E-4/year)
Transient Initiator	9.6E-2 / calendar year	1.8 / calendar year	8.2 / calendar year	72 /calendar year	720 / calendar year
Loss of Feedwater	6.8E-2 / calendar year	2.0E-1 / calendar year	8.0E-1 / calendar year	12 / calendar year	120 / calendar year
Loss of Heat Sink	9.6E-2 / calendar year	2.6E-1 / calendar year	2.8E-1 / calendar year	2.9 / calendar year	28 / calendar year

1. Calendar year is defined as 7000 critical hours.

Table A.1.4-20 CE Plant 4 Initiating Events Threshold Summary

CE Plant 4 SPAR 3i (2.6E-9/hr, 1.8E-5/calendar year[1]) CE Plant 4 RBPI Baseline (2.2E-9/hr, 1.6E-5/calendar year[1])					
PWR Initiator	Baseline IE Frequency (NUREG/CR-5750)	Green/White Threshold 95th%ile	Green/White Threshold (ΔCDF =1E-6/year)	White/Yellow Threshold (ΔCDF =1E-5/year)	Yellow/Red Threshold (ΔCDF =1E-4/year)
Transient Initiator	9.6E-1 / calendar year	1.8 / calendar year	1.9 / calendar year	9.3 / calendar year	88 / calendar year
Loss of Feedwater	6.8E-2 / calendar year	2.0E-1 / calendar year	5.4E-1 / calendar year	4.8 / calendar year	48 / calendar year
Loss of Heat Sink	9.6E-2 / calendar year	2.6E-1 / calendar year	2.0E-1 / calendar year	1.1 / calendar year	10 / calendar year

1. Calendar year is defined as 7000 critical hours.

Table A.1.4-21 CE Plant 5 Initiating Events Threshold Summary

CE Plant 5 SPAR 3i (4.0E-9/hr, 2.8E-5/calendar year[1]) CE Plant 5 RBPI Baseline (2.6E-9/hr, 1.8E-5/calendar year[1])					
PWR Initiator	Baseline IE Frequency (NUREG/CR-5750)	Green/White Threshold 95th%ile	Green/White Threshold (ΔCDF =1E-6/year)	White/Yellow Threshold (ΔCDF =1E-5/year)	Yellow/Red Threshold (ΔCDF =1E-4/year)
Transient Initiator	9.6E-1 / calendar year	1.8 / calendar year	1.3 / calendar year	4.1 / calendar year	32 / calendar year
Loss of Feedwater	6.8E-2 / calendar year	2.0E-1 / calendar year	2.3E-1 / calendar year	1.8 / calendar year	17 / calendar year
Loss of Heat Sink	9.6E-2 / calendar year	2.6E-1 / calendar year	1.3E-1 / calendar year	4.4E-1 / calendar year	3.6 / calendar year

1. Calendar year is defined as 7000 critical hours.

Table A.1.4-22 CE Plant 10/11 Initiating Events Threshold Summary

CE Plant 10/11 SPAR 3i (7.4E-9/hr, 5.1E-5/calendar year[1]) CE Plant 10/11 RBPI Baseline (8.6E-9/hr, 6.0E-5/calendar year[1])					
Initiator	Baseline Initiator Frequency (NUREG/CR-5750)	Green/White Threshold 95th%ile	Green/White Threshold (ΔCDF =1E-6/year)	White/Yellow Threshold (ΔCDF =1E-5/year)	Yellow/Red Threshold (ΔCDF =1E-4/year)
Transient Initiator	9.6E-1 / calendar year	1.8 / calendar year	1.4/ calendar year	4.2 / calendar year	33 / calendar year
Loss of Feedwater	6.8E-2 / calendar year	2.0E-1 / calendar year	2.2E-1 / calendar year	2.0 / calendar year	20 / calendar year
Loss of Heat Sink	9.6E-2 / calendar year	2.6E-1 / calendar year	1.1E-1 / calendar year	5.0E-1 / calendar year	4.1 / calendar year

1. Calendar year is defined as 7000 critical hours.

Table A.1.4-23 CE Plant 12 Initiating Events Threshold Summary

CE Plant 12 SPAR 3i (4.0E-9/hr, 2.8E-5/calendar year[1]) CE Plant 12 RBPI Baseline (2.7E-9/hr, 1.9E-5/calendar year[1])					
PWR Initiator	Baseline IE Frequency (NUREG/CR-5750)	Green/White Threshold 95th%ile	Green/White Threshold (ΔCDF =1E-6/year)	White/Yellow Threshold (ΔCDF =1E-5/year)	Yellow/Red Threshold (ΔCDF =1E-4/year)
Transient Initiator	9.6E-1 / calendar year	1.8 / calendar year	1.6 / calendar year	7.2 / calendar year	62 / calendar year
Loss of Feedwater	6.8E-2 / calendar year	2.0E-1 / calendar year	4.2E-1 / calendar year	3.7 / calendar year	35 / calendar year
Loss of Heat Sink	9.6E-2 / calendar year	2.6E-1 / calendar year	1.2E-1 / calendar year	9.6E-1 / calendar year	8.8 / calendar year

1. Calendar year is defined as 7000 critical hours.

Table A.1.4-24 WE 2-Lp Plant 5/6 Initiating Events Threshold Summary

WE 2-Lp Plant 5/6 SPAR 3i (2.1E-9/hr, 1.4E-5/calendar year[1]) WE 2-Lp Plant 5/6 RBPI Baseline (2.1E-9/hr, 1.5E-5/calendar year[1])					
PWR Initiator	Baseline IE Frequency (NUREG/CR-5750)	Green/White Threshold 95th%ile	Green/White Threshold (ΔCDF =1E-6/year)	White/Yellow Threshold (ΔCDF =1E-5/year)	Yellow/Red Threshold (ΔCDF =1E-4/year)
Transient Initiator	9.6E-1 / calendar year	1.8 / calendar year	1.4 / calendar year	4.7 / calendar year	38 / calendar year
Loss of Feedwater	6.8E-2 / calendar year	2.0E-1 / calendar year	4.0E-1 / calendar year	3.2 / calendar year	32 / calendar year
Loss of Heat Sink	9.6E-2 / calendar year	2.6E-1 / calendar year	1.7E-1 / calendar year	9.6E-1 / calendar year	8.8 / calendar year

1. Calendar year is defined as 7000 critical hours.

Table A.1.4-25 WE 3-LP Plant 5 Initiating Events Threshold Summary

WE 3-LP Plant 5 SPAR 3i (6.3E-9/hr, 4.4E-5/calendar year[1]) WE 3-LP Plant 5 RBPI Baseline (6.7E-9/hr, 4.7E-5/calendar year[1])					
Initiator	Baseline Initiator Frequency (NUREG/CR-5750)	Green/White Threshold 95th%ile	Green/White Threshold (ΔCDF =1E-6/year)	White/Yellow Threshold (ΔCDF =1E-5/year)	Yellow/Red Threshold (ΔCDF =1E-4/year)
Transient Initiator	9.6E-1 / calendar year	1.8 / calendar year	1.7 / calendar year	7.9 / calendar year	69 / calendar year
Loss of Feedwater	6.8E-2 / calendar year	2.0E-1 / calendar year	6.3E-1 / calendar year	6.1 / calendar year	61 / calendar year
Loss of Heat Sink	9.6E-2 / calendar year	2.6E-1 / calendar year	1.9E-1 / calendar year	1.4 / calendar year	13 / calendar year

1. Calendar year is defined as 7000 critical hours.

Table A.1.4-26 WE 3-LP Plant 10/11 Initiating Events Threshold Summary

WE 3-LP PLANT 10/11 SPAR 3i (3.2E-9/hr, 2.3E-5/calendar year[1]) WE 3-LP PLANT 10/11 RBPI BASELINE (2.9E-9/hr, 2.1E-5/calendar year[1])					
Initiator	Baseline Initiator Frequency (NUREG/CR-5750)	Green/White Threshold 95th%ile	Green/White Threshold (ΔCDF =1E-6/year)	White/Yellow Threshold (ΔCDF =1E-5/year)	Yellow/Red Threshold (ΔCDF =1E-4/year)
Transient Initiator	9.6E-1 / calendar year	1.8 / calendar year	1.4 / calendar year	5.0 / calendar year	41 / calendar year
Loss of Feedwater	6.8E-2 / calendar year	2.0E-1 / calendar year	3.6E-1 / calendar year	3.5 / calendar year	36 / calendar year
Loss of Heat Sink	9.6E-2 / calendar year	2.6E-1 / calendar year	1.4E-1 / calendar year	8.6E-1 / calendar year	8.0 / calendar year

1. Calendar year is defined as 7000 critical hours.

Table A.1.4-27 WE 4-Lp Plant 1/2 Initiating Events Threshold Summary

WE 4-Lp Plant 1/2 SPAR 3i (1.0E-8/hr, 7.2E-5/calendar year[1]) WE 4-Lp Plant 1/2 RBPI Baseline (1.1E-8/hr, 7.5E-5/calendar year[1])					
PWR Initiator	Baseline IE Frequency (NUREG/CR-5750)	Green/White Threshold 95th%ile	Green/White Threshold (ΔCDF =1E-6/year)	White/Yellow Threshold (ΔCDF =1E-5/year)	Yellow/Red Threshold (ΔCDF =1E-4/year)
Transient Initiator	9.6E-1 / calendar year	1.8 / calendar year	1.2 / calendar year	3.2 / calendar year	24 / calendar year
Loss of Feedwater	6.8E-2 / calendar year	2.0E-1 / calendar year	1.9E-1 / calendar year	2.1 / calendar year	20 / calendar year
Loss of Heat Sink	9.6E-2 / calendar year	2.6E-1 / calendar year	9.7E-2 / calendar year	4.8E-1 / calendar year	4.0 / calendar year

1. Calendar year is defined as 7000 critical hours.

Table A.1.4-28 WE 4-LP Plant 10/11 Initiating Events Threshold Summary

WE 4-LP Plant 10/11 SPAR 3i (3.5E-9/hr, 2.5E-5/calendar year[1]) WE 4-LP Plant 10/11 RBPI Baseline (3.6E-9/hr, 2.5E-5/calendar year[1])					
Initiator	Baseline Initiator Frequency (NUREG/CR-5750)	Green/White Threshold 95th%ile	Green/White Threshold (ΔCDF =1E-6/year)	White/Yellow Threshold (ΔCDF =1E-5/year)	Yellow/Red Threshold (ΔCDF =1E-4/year)
Transient Initiator	9.6E-1 / calendar year	1.8 / calendar year	1.8 / calendar year	8.0 / calendar year	73 / calendar year
Loss of Feedwater	6.8E-2 / calendar year	2.0E-1 / calendar year	6.8E-1 / calendar year	6.4 / calendar year	64 / calendar year
Loss of Heat Sink	9.6E-2 / calendar year	2.6E-1 / calendar year	2.1E-1 / calendar year	1.5 / calendar year	14 / calendar year

1. Calendar year is defined as 7000 critical hours.

Table A.1.4-29 WE 4-Lp Plant 22/23 Initiating Events Threshold Summary

WE 4-Lp Plant 22/23 SPAR 3i (4.7E-9/hr, 3.3E-5/calendar year[1]) WE 4-Lp Plant 22/23 RBPI Baseline (4.9E-9/hr, 3.4E-5/calendar year[1])					
PWR Initiator	Baseline IE Frequency (NUREG/CR-5750)	Green/White Threshold 95th%ile	Green/White Threshold (ΔCDF =1E-6/year)	White/Yellow Threshold (ΔCDF =1E-5/year)	Yellow/Red Threshold (ΔCDF =1E-4/year)
Transient Initiator	9.6E-1 / calendar year	1.8 / calendar year	1.8 / calendar year	8.8 / calendar year	78 / calendar year
Loss of Feedwater	6.8E-2 / calendar year	2.0E-1 / calendar year	8.0E-1 / calendar year	7.2 / calendar year	74 / calendar year
Loss of Heat Sink	9.6E-2 / calendar year	2.6E-1 / calendar year	2.4E-1 / calendar year	1.5 / calendar year	15 / calendar year

1. Calendar year is defined as 7000 critical hours.

Table A.1.4-30 WE 4-LP Plant 28 Initiating Events Threshold Summary

WE 4-LP Plant 28 SPAR 3i (5.0E-9/hr, 3.5E-5/calendar year[1]) WE 4-LP Plant 28 RBPI Baseline (3.8E-9/hr, 2.7E-5/calendar year[1])					
Initiator	Baseline Initiator Frequency (NUREG/CR-5750)	Green/White Threshold 95th%ile	Green/White Threshold (ΔCDF =1E-6/year)	White/Yellow Threshold (ΔCDF =1E-5/year)	Yellow/Red Threshold (ΔCDF =1E-4/year)
Transient Initiator	9.6E-1 / calendar year	1.8 / calendar year	2.0 / calendar year	10 / calendar year	93 / calendar year
Loss of Feedwater	6.8E-2 / calendar year	2.0E-1 / calendar year	1.0 / calendar year	9.9 / calendar year	99 / calendar year
Loss of Heat Sink	9.6E-2 / calendar year	2.6E-1 / calendar year	2.7E-1 / calendar year	2.1 / calendar year	20/ calendar year

1. Calendar year is defined as 7000 critical hours.

Table A.1.5-1 Summary of Inspection Areas Impacted by New RBPIs for Initiating Event Cornerstone

RBPI	Attribute	Inspection Area
General Transient	Equipment Performance	71111.12, Maintenance Rule Implementation 71111.08, Inservice Inspection Activities 71111.20, Refueling and Outage Activities 71152, Identification and Resolution of Problems
	Human Performance	71111.14, Personnel Performance During Non-routine Evolutions
LOFW		None
LOHS	Protection Against External Factors	71111.07, Heat Sink Performance

A.2 Mitigating Systems Cornerstone

This section discusses development of RBPIs that address the mitigating systems cornerstone for full power, internal events. External events and non-power modes are addressed in other sections.

A.2.1 Assess the Potential Risk Impact of Degraded Performance

The objective of the mitigating system cornerstone is to ensure adequate performance (availability, reliability, and capability) of systems that mitigate initiating events to prevent reactor accidents. Six 'key attributes' that contribute to mitigating system performance are identified in SECY 99-007 (Ref. 1). These six attributes consist of configuration control, procedure quality, human performance, protection against external events, equipment performance, and design.

A.2.1.1 Determine Attributes That Are Risk-Significant and Explicitly Modeled

Determination of 'risk-significant' or 'risk-based' performance indicators necessitates a means of quantifying the impact of that attribute. However, of the mitigating system attributes listed above, only equipment performance and some aspects of human performance (i.e., post initiator actions) are explicitly modeled and can be quantified in currently available risk models (IPE and SPAR). Potential performance indicators are further reduced by the fact that even though human performance is modeled and is shown to be risk-significant, changes in performance are not readily measurable. Currently there is no established method of identifying changes in operator performance and then feeding this information back into the SPAR models. As a result, equipment performance is the only mitigating system attribute that will be evaluated in this analysis.

Risk-significance of modeled mitigating systems was determined through analysis of Revision 3i SPAR models supplemented by quantification results found in the Individual Plant Examination (IPE) submittals and the associated IPE Database (Ref. 5). Risk-significance of mitigating systems was based on importance measures. Importance information resulting from quantification of the models was summarized on a plant-specific basis by system/component and evaluated for importance to overall plant risk. Importance measure values in accordance with those specified in the PSA Applications Guide (Ref. 18) and Regulatory Guide 1.160 (Maintenance Rule, Ref. 19) were utilized in the determination of risk-significance. A system was considered to be risk-significant at the plant level if its system level Fussell-Vesely Importance (FV) > 0.05. A system was also considered risk-significant if a component within that system yielded a Risk Achievement Worth (RAW) > 2.0 in conjunction with a component level FV > 0.005. Systems that met either of these criteria were considered risk-significant at the individual plant level. Support systems identified in the IPE database as contributing in excess of five percent to overall core damage frequency were also considered important at the plant level.

In addition to risk-significant systems, risk-significant component classes were also identified using a similar process. The same importance criteria were used to select component class indicators, however, the system level Fussell-Vesely Importance values were determined using the multi-variable or group function available in SAPHIRE. There are two main benefits for identifying component group RBPIs. The first is that trends and impacts on CDF that might not be detected at the individual system level might be picked up at the component group level. The second benefit is that the component group RBPIs can be trended across plant groups or the entire industry to detect early signs of deteriorating performance. Three component classes were identified as risk-significant. These classes include air-operated valves (AOVs), motor-driven pumps (MDPs), and motor-operated valves (MOVs).

The RBPI white paper (Ref. 3) indicates that RBPI development will be performed in a manner to group similar plants so that a given set of RBPIs apply to the entire group. This task was performed in two steps. The first step was performed prior to determining risk-significance of specific systems. In this step all plants were grouped according to similarities in configuration and/or design that were expected to result in differences in systems selected as important. This step facilitated identification of a preliminary plant grouping based on systems that may be important at only a subset of plants having a particular design characteristic. The second step was to validate the plant groupings based on actual system importance results obtained from the quantified models. Due to the limited number of plants in the pilot program, only two distinct plant groups were identified and then validated (BWR and PWR). Additional plant grouping are anticipated following evaluation of the remaining plants. Additionally, as more plants are evaluated, it is expected that some mitigating system RBPIs may be eliminated from some plant groups (e.g., CCW). The list of risk-significant mitigating systems and the plant groups to which they are generically applicable are listed in Table A.2.1.1-1. Systems that are risk-significant at only a single plant or a limited number of plants were identified in Tables A.2.4-1 through A.2.4-30 as plant-specific inspection candidates.

Table A.2.1.1-1 Modeled Risk-Significant Mitigating Systems

	Plant Group #1 (BWR)	Plant Group #2 (PWR)	Timely Detection of Performance Changes at the Plant Level and Availability of Performance Data
Auxiliary/Emergency Feedwater		X	Yes
Component Cooling Water		X	Yes[1]
Emergency AC Power	X	X	Yes
High Pressure Coolant Injection Systems (HPCI, HPCS)	X		Yes
High Pressure Heat Removal Systems (RCIC, IC)	X		Yes[2]
High Pressure Safety Injection		X	Yes
Main Feedwater	X		Yes (As LOFW RBPI)
Main Steam/Main Steam Isolation		X	Yes (As LOFW/LOHS RBPI)
Power Conversion System	X		Yes (As LOHS RBPI)
Power Operated Relief Valve		X	Yes[1]
Primary Pressure Relief	X	X	No[3]
Reactor Protection System	X	X	No[4]
Residual/Decay Heat Removal	X	X	Yes
Service Water (To EDG/RHR)	X	X	Yes
Risk-Significant Component Classes			
Air-Operated Valves	X	X	Yes
Motor-Operated Valves	X	X	Yes
Motor-Driven Pumps	X	X	Yes

1 Marginal RBPI candidate, may be removed following evaluation of additional plants and/or data.

2 The Isolation Condenser was provisionally added as a 'Mitigating System' performance indicator at the five units that comprise the BWR 1/2/3 class based on importances calculated in original IPE submittals. The inclusion of this system as an RBPI will be re-evaluated following completion of Revision 3 SPAR models for these plants.

3 Timely detection of performance at the plant level is not feasible due to sparseness of data.

4 The Reactor Protection System (RPS) has substantial safety implications if performance degrades significantly. However, the RPS is not included as a candidate RBPI due to significant differences between the level of detail found in the SPAR 3i logic and level at which failure data is reported in EPIX. The current SPAR 3i models through which the RBPI thresholds are calculated have limited detail in the RPS system logic. The BWR models contain four hardware events and the PWR models contain three events. EPIX contains extensive amounts of failure data associated with dozens of components in the RPS system but at a much lower level of detail. Without significant modification to the SPAR 3i RPS logic to incorporate lower levels of data, it is not feasible to incorporate updated EPIX failure data into the RPS models so that changes in performance can be quantified and tracked.

A.2.1.2 Determine Monitoring Levels for Each Element

Performance can be monitored using indicators at different levels, ranging from the function level comprising multiple systems down to the level of the individual component failure mode. Higher level (e.g., function or system level) indicators have certain positive attributes: they allow for more licensee flexibility than lower level indicators, and provide more apparent coverage per indicator, resulting in fewer indicators for a given level of apparent coverage than would be needed using lower-level indicators. However, in some areas, certain practical considerations compel the selection of indicators at a lower level. In these areas, train-level indicators are used. Train-level indicators are further broken down into unreliability and unavailability indicators. The following discussion addresses the practical considerations that lead to selection of train-level indicators.

The use of a single indicator above the train level is inappropriate for systems with dissimilar trains. Table A.2.1.2-1 illustrates this point with an example of an Auxiliary Feedwater System consisting of a diesel-driven pump train and a motor-driven pump train. The dominant accident scenarios associated with this plant are associated with LOOP events, especially events in which on-site AC power is also lost. If AC motive power is not available, the AC-driven pump train performance is moot, and the diesel-driven pump train performance is especially important. This argument suggests that changes in CDF due to decreases in AFW system performance are much more sensitive to degradation of the diesel driven train performance than to an equivalent change in AFW system performance due to degradation of performance of the motor-driven train. This is reflected in the values in Table A.2.1.2-1. These differences are due to the mission specific nature of the different trains. Therefore, to accurately reflect the risk implications of a given change in performance, separate indicators are required for dissimilar trains of a given system.

For similar reasons, systems that have train specific loads such as emergency AC power lend themselves well to train level unavailability indicators. Many service water and component cooling water systems also have train specific loads (i.e., lack of a single common header) and are better addressed with train level indicators.

Additionally, failures at the train level are much more frequent than system-level failures of multiple-train systems. Thus, the timely detection of performance trends at the train level is typically much more feasible than at the system level.

Table A.2.1.2-1 Auxiliary Feedwater System Example of the Differing Impacts of Dissimilar Trains on CDF

AFW System Top Event Probability	3.7E-4 (Nominal)	5.0E-4	1.0E-3	5.0E-3
ΔCDF Associated with Degradation of Diesel-Driven Pump (DDP) Train Performance Only.	ΔCDF = 0.0/year	ΔCDF = 5.6E-6/year	ΔCDF = 1.1E-4/year	ΔCDF = 8.7E-4/year
ΔCDF Associated with Degradation of Motor-Driven Pump (MDP) Train Performance Only	ΔCDF = 0.0/year	ΔCDF = 3.5E-7/year	ΔCDF = 6.7E-6/year	ΔCDF = 5.8E-5/year

Separate indicators for unreliability and unavailability are also appropriate because the relationship between system performance and CDF is highly dependent on whether reliability or availability is causing the change in system performance. The difference arises because train unavailability is somewhat constrained by Technical Specifications, while reliability is not. In the calculation of CDF, Technical Specifications are assumed to be followed explicitly, and cutsets with disallowed maintenance combinations are eliminated from the CDF cutset tabulations. Table A.2.1.2-2 illustrates the results of this process. This table shows that CDF is more sensitive to EPS reliability than to EPS availability.

Another difference between the significance of unavailability changes and unreliability changes arises as a result of common cause failure (CCF). As modeled, an increase in a specific train's unreliability affects CDF not only through the increased probability of failures of that train, but also through the increased probability of common cause failures of redundant trains. Unavailability does not behave in the same way; as discussed above, concurrent unavailability of redundant trains is limited by technical specifications.

Therefore, a single train-level performance indicator that combines unreliability and unavailability is inadequate to address the risk implications of changing system performance.

Table A.2.1.2-2 Emergency Power System Example of the Differing Impacts of Unavailability and Unreliability on CDF

Emergency Diesel Generator Top Event Probability	4.6E-2 (Nominal)	5.0E-2	1.0E-1	5.0E-1
ΔCDF Associated with Degradation of Diesel Generator Availability (UA) Only	ΔCDF = 0.0/year	ΔCDF = 6.3E-7/year	ΔCDF = 9.1E-6/year	ΔCDF = 7.6E-5/year
ΔCDF Associated with Degradation of Diesel Generator Reliability (UR) Only	ΔCDF = 0.0/year	ΔCDF = 1.1E-6/year	ΔCDF = 2.4E-5/year	ΔCDF = 5.8E-4/year

Other considerations also support the use of train level unavailability indicators. SECY 99-007 also identifies reliability and availability as the two specific elements associated with equipment performance. Maintenance is normally performed on the train level and is intrinsically recognizable as such to plant personnel. This fact is also incorporated in the SPAR models with their placement of test and maintenance events at the train level. Additionally, WANO reports unavailability at the train level.

There are some shortcomings in using train level performance indicators. A few system fault trees in the SPAR models include common cause failures (CCF) at the system level. Since CCF events are often significant contributors to overall system unreliability, system level unreliability indicators would more closely mimic actual CDF changes. Finally, non-redundant systems are typically best addressed at the system level. For example, the key safety function of PORVs at some plants requires success of 2/2 PORVs, so that if either PORV fails, the function fails.

Table A.2.1.2-3 identifies the risk-significant systems, elements and the level of the associated performance indicator.

Table A.2.1.2-3 Candidate Mitigating System RBPIs and Monitoring Level

BWR RBPI SYSTEMS	RBPI Level
Emergency AC Power (EPS)	Unreliability and unavailability both monitored at the train level.
High Pressure Coolant Injection Systems • High Pressure Coolant Injection (HPCI) • High Pressure Core Spray (HPCS)	Unreliability and unavailability both monitored at the train level.
High Pressure Heat Removal Systems • Isolation Condenser (IC) • Reactor Core Isolation Cooling (RCIC)	Unreliability and unavailability both monitored at the train level.
Residual Heat Removal (SPC, RHR)	Unreliability and unavailability both monitored at the train level.
Service Water (SWS)	Unreliability and unavailability both monitored at the train level.
PWR RBPI SYSTEMS	
Auxiliary Feedwater (AFW/EFW) • Motor-driven Pump Train • Turbine-driven Pump Train	 Unreliability and unavailability both monitored at the train level. Unreliability and unavailability both monitored at the train level.
Component Cooling Water (CCW)	Unreliability and unavailability both monitored at the train level.
Emergency AC Power (EPS)	Unreliability and unavailability both monitored at the train level.
High Pressure Injection (HPI)	Unreliability and unavailability both monitored at the train level.
Power Operated Relief Valve (PORV)	Unreliability monitored at the system level.
Residual/Decay Heat Removal (RHR)	Unreliability and unavailability both monitored at the train level.
Service Water (SWS)	Unreliability and unavailability both monitored at the train level.
COMPONENT CLASSES (all plants)	
Air-Operated Valves (AOVs)	Unreliability monitored at the component level.
Motor-Operated Valves (MOVs)	Unreliability monitored at the component level.
Motor-Driven Pumps (MDPs)	Unreliability monitored at the component level.

A.2.2 Obtain Performance Data for Risk-Significant, Equipment-Related Elements

The analysis of mitigating system performance also relies on several data sources. The primary data sources used in the selection of, and their contribution to, the analysis of mitigating system RBPIs are described below:

The Equipment Performance and Information Exchange database (EPIX) is an industry-sponsored effort to collect performance information for key components in or affecting risk-significant systems as identified in plant maintenance rule programs. EPIX (Ref. 15) is a replacement for the Nuclear Plant Reliability Data System (NPRDS) database. (Data reporting to NPRDS stopped at the end of 1996.) All nuclear utilities have submitted some reliability data for entry into EPIX. The current RBPI pilot effort uses EPIX data to support the evaluation of mitigating system unreliability RBPIs.

The Reliability and Availability Data System (RADS) interfaces with established data sources to provide risk analysis capability for use with risk-informed applications and regulations. RADS (Ref. 16) takes this raw failure, demand, and unavailability information, and manipulates it to yield reliability parameters that can be used in PRA analyses. Availability data will also be available in RADS in the near future. RADS reports these data on a component or train level for a specified selection of key systems and components. RADS also estimates CCF rates and performs trending analyses. Other uses include monitoring maintenance rule implementation, supporting plant-specific licensing actions, and improving accident sequence precursor analyses. The current RBPI pilot effort uses RADs to screen data to support the evaluation of mitigating system unreliability RBPIs.

System reliability studies (Refs. 9, 10, 11, 12, 13 and 14) have been and are being conducted to systematically evaluate operational data of risk-significant systems at nuclear power plants. The primary objectives of the studies are twofold. The first objective is to estimate system unreliability based on operational data and then to compare the results with data, models, and assumptions used in IPEs. The second is to provide an engineering analysis of the factors affecting system unreliability and to determine any trends or patterns. Other objectives include identification of failure trends over time and generation of baseline performance data from which to compare industry-wide and plant-specific performance. In addition to containing the most current data failure available for these systems, the failure data contained in these studies has been extensively analyzed to retain only valid failures and to accurately characterize the nature of those failures. This data was incorporated into the SPAR models as part of the process of establishing plant-specific 'baseline' models and associated core damage frequencies.

A.2.3 Identify Indicators Capable of Detecting Performance Changes in a Timely Manner

In addition to being risk-significant (see Table A.2.1.1-1), mitigating system performance indicators must be capable of detecting performance changes in a timely manner. A mitigating system performance indicator involves collection of data during some monitoring period, and a decision rule, which declares that a plant is in a certain performance band based on observed data. This monitoring period must be long enough to reduce the probabilities of false negatives and false positives to acceptable levels, but no longer. Appendices E and F document these statistical analyses.

In accordance with the preceding discussion and the statistical analyses documented in Appendices E and F, several mitigating systems/component classes met the criteria for risk-significance and timely monitoring and were selected as candidates to be monitored as mitigating system RBPIs. Monitoring periods of reasonable length were also calculated based on acceptable levels of false positives and negatives. Additionally, changes in their frequencies can be readily quantified with the current SPAR models. These systems and component classes are identified in Table A.2.1.2-1.

Some risk-significant systems were considered best monitored as initiating event RBPIs. These systems are discussed in Section A.2.3.1. In addition to these systems, one other risk-significant system identified in Table A.2.1.1-1 was not considered to be a good candidate for mitigating system RBPIs. This system, the primary pressure relief system, was excluded due to the sparseness of data and the resulting excessive monitoring periods. It will be consigned to risk-informed baseline inspections.

A.2.3.1 Treatment of Systems Whose Function Is Monitored under Initiating Event RBPIs

Main feedwater, power conversion and main steam are risk-significant systems whose functions are best monitored under the LOHS and LOFW initiating event RBPIs. Several factors combine to prevent these systems from being good mitigating system RBPI candidates and lead to monitoring of their performance within the LOHS and LOFW RBPIs.

First, these systems are continuously operational during normal power operations and function with little or no redundancy. This lack of redundancy precludes generating an unavailability indicator since there is no standby equipment. Additionally, since there is no standby equipment, some types of failure data associated with testing of standby equipment (e.g., failure to start) is sparse. Finally, failure of any major component within these systems results in an immediate plant trip or shutdown. The impact of these trips and shutdowns is explicitly monitored through the LOHS and LOFW RBPIs.

A.2.3.2 Industry-wide Trending of Mitigating Systems

Similar to mitigating system RBPIs, candidates for industry-wide trending must also be risk-significant. In addition to the mitigating system RBPIs identified in Table A.2.1.2-1, common cause failure (CCF) events were also included as potential candidates for industry-wide trending. Analysis of the SPAR model results indicate that CCF events associated with Auxiliary/ Emergency Feedwater pumps and emergency diesel generators are significant contributors to risk. Since these events do not occur frequently enough to track on a plant-specific basis they will be trended industry-wide. Other system specific CCF categories may be added as additional plants are evaluated. Finally, CCF events associated with all systems are as a group very risk-significant and will also be trended. The mitigating system industry-wide trending candidates are listed below:

- All systems and component classes identified in Table 3.1.2.1 as RBPIs
- Common Cause Failure Events for Auxiliary Feedwater Pumps
- Common Cause Failure Events for Emergency Diesel Generators
- Common Cause Failure Events for All Systems

The mitigating system RBPIs are trended in Table 5.3-7 of the main body of the report. Trends associated with non-RBPI events (Common Cause Failures) are shown below in Figures A.2.3.2-1 through A.2.3.2-3.

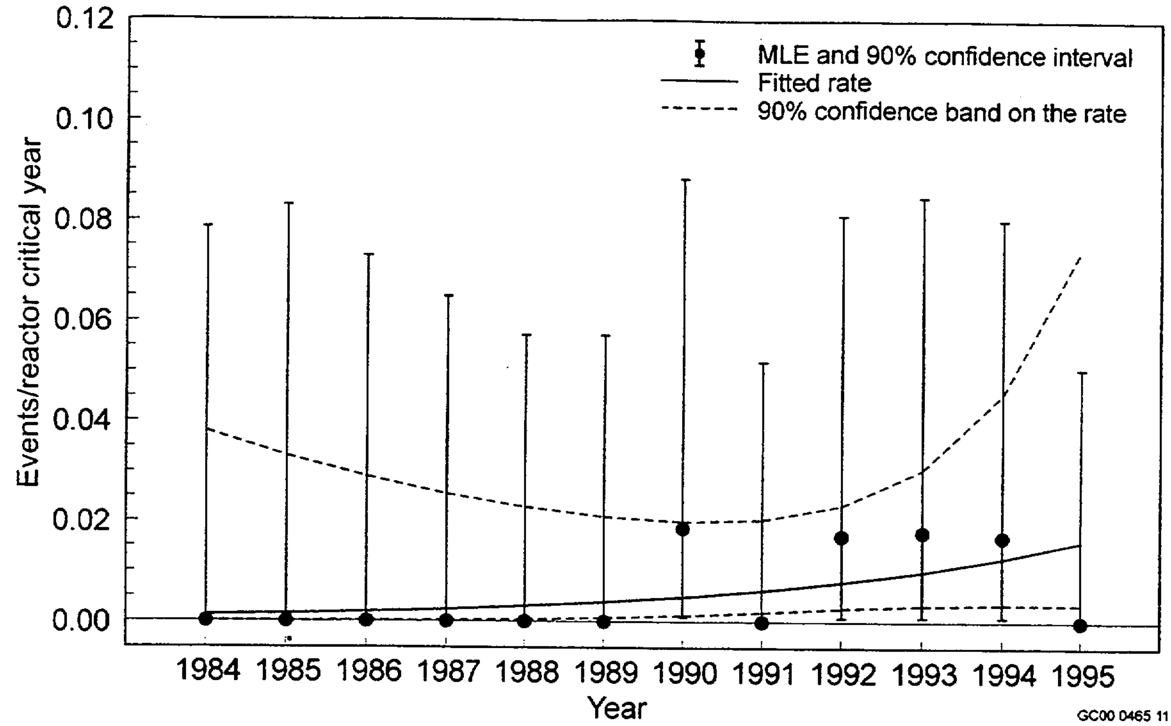

Figure A.2.3.2-1 Time-dependent Trending of CCF Events for Auxiliary Feedwater Pumps (Trend is not statistically significant, p-value = 0.2.)

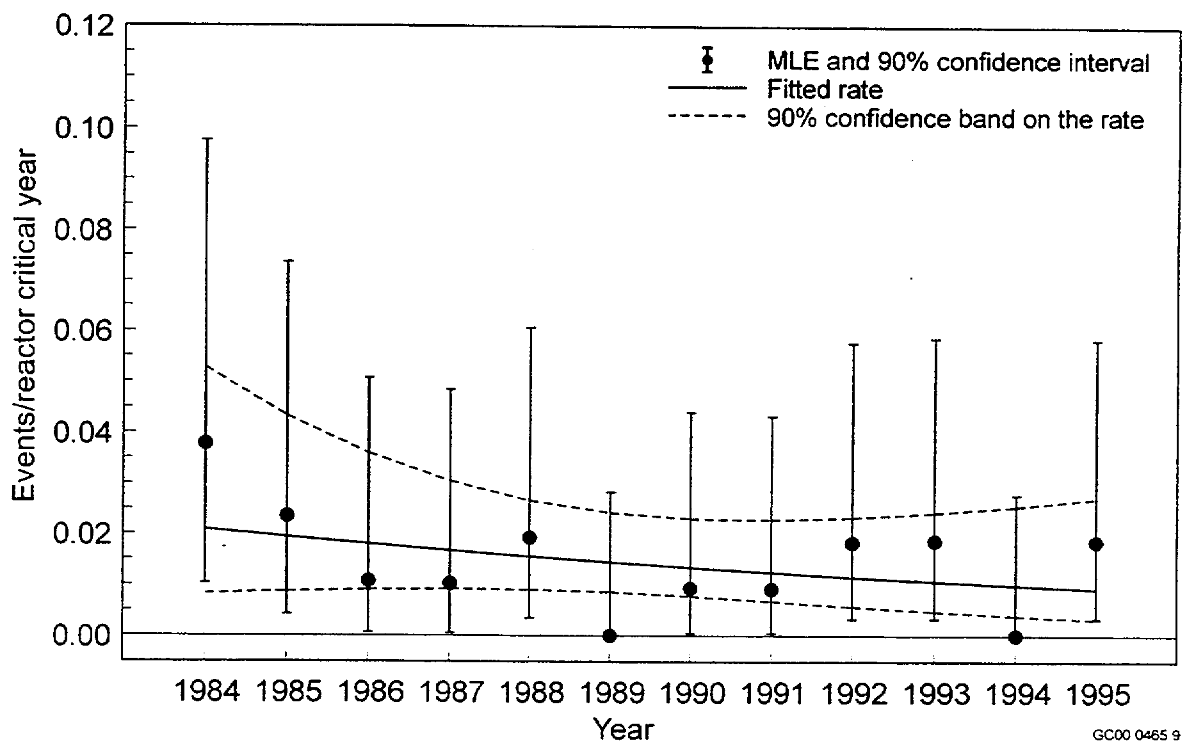

Figure A.2.3.2-2 Time-dependent Trending of CCF Events for Emergency Diesel Generators (Trend is not statistically significant, p-value = 0.3.)

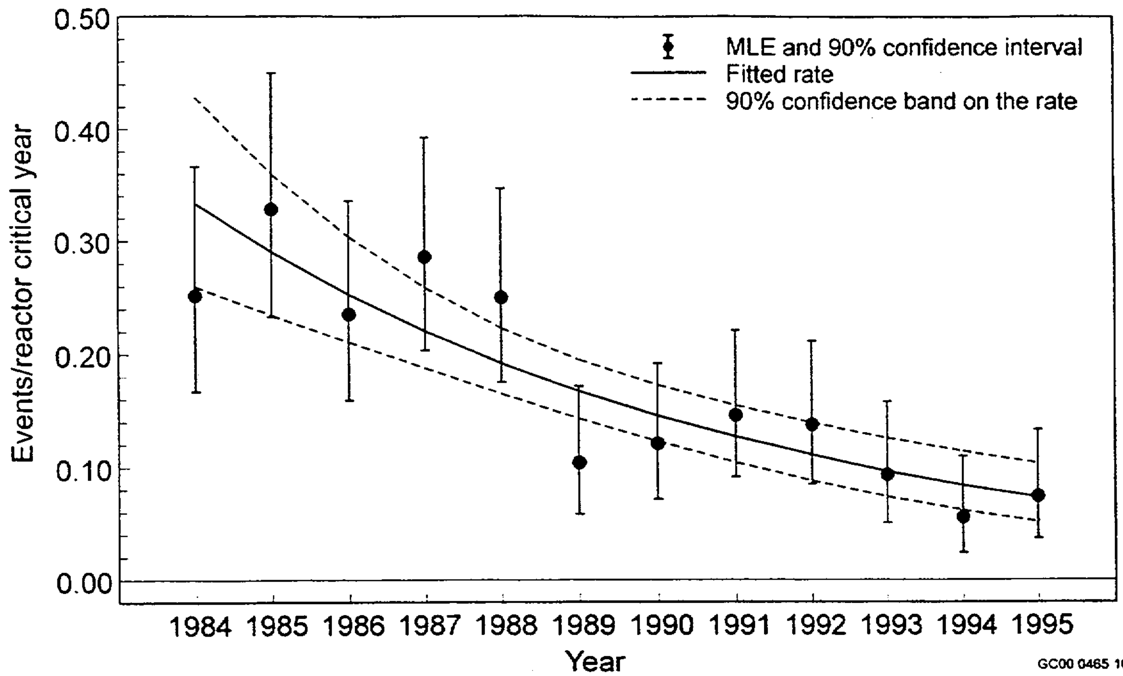

Figure A.2.3.2-3 Time-dependent Trending of CCF Events for All Systems (Trend is statistically very significant, p-value < 0.0001.)

A.2.4 Identify Performance Thresholds Consistent with a Graded Approach to Performance Evaluation from SECY 99-007

The same graded approach outlined in Section A.1.4 for Initiating Event RBPI thresholds is also used for setting mitigating system RBPI thresholds. This graded approach is built around four performance bands (green, white, yellow, red) whose boundaries correspond to plant-specific changes in CDF equal to 1E-6/yr, 1E-5/yr and 1E-4/yr.

Again, the same 'baseline' models defined in Section A.1.4 and used to identify Initiating Event thresholds are also used to identify mitigating system thresholds. An iterative technique was once more employed to determine the exact mitigating system thresholds. System specific unavailability thresholds were determined by simultaneously increasing all train level test and maintenance probabilities (for similar trains) within the subject system until the appropriate change in CDF was reached. Unreliability thresholds were determined by simultaneously increasing the random failure probabilities of all system specific equipment tracked in EPIX until the appropriate change in CDF was reached. For the unreliability calculations, the test and maintenance probabilities were kept constant at their baseline values. See Appendix H for more details.

Once the performance action band boundary is reached, the unreliability threshold value is calculated by quantifying the fault tree gate that corresponds to that train/system. The unavailability threshold is calculated similarly to the unreliability threshold except that only the test and maintenance (T&M) events are increased and the value of the T&M event then becomes the threshold. "Not Reached" in the threshold summary tables indicates an inability to reach the subject performance action band boundary with the train/system failed. In some instances (identified with an accompanying footnote), a "Not Reached" corresponding to an unavailability threshold indicates an inability to reach the subject performance action band boundary while staying within allowable Technical Specification maintenance combinations.

Mitigating system RBPIs were selected and their threshold values calculated for 30 sites (44 plants). These sites are comprised of 19 BWR and 25 PWR plants. Detailed threshold information for each analyzed plant is contained in Tables A.2.4-1 through A.2.4-30.

A.2.5 Inspection Areas Covered by New RBPIs

The RBPIs developed in this report for the mitigating system cornerstone were compared with the performance indicators in the ROP to identify those RBPIs that are not currently in the ROP. The inspection areas that could be impacted by the new mitigating system RBPIs were then determined. The results are summarized in Table A.2.5-1.

A.3 Barrier Integrity Cornerstone: Containment

This section presents the background for the preliminary RBPI development results that address the containment integrity portion of the barrier integrity cornerstone for full power, internal events. The scope of the structures, systems, and components related to the containment barrier includes the primary and secondary containment buildings, primary containment penetrations and associated isolation systems, and risk-significant systems and components necessary for containment heat removal, pressure control, and degraded core hydrogen control. This section is focused on the containment barrier itself, and bypass of the containment barrier (for example, by steam generator tube rupture) is not considered in this section.

The section is structured in a manner similar to that for Section A.2, Mitigating Systems Cornerstone, and follows the RBPI development process described in the main report. A special subsection is added, Section A.3.6, that addresses:

- The definition of LERF
- The justification for using LERF

In discussion of the initiating events cornerstone and the mitigating systems cornerstone, emphasis was placed on CDF as the metric for defining the risk significance of changes. In this section, LERF is used as the metric for determining the risk significance of changes in containment performance. However, the burden of this section is containment integrity, not LERF in general. Many influences on LERF need to be addressed under other cornerstones. To clarify this point, it is useful to classify hardware and human performance elements according to a recent development carried out for the SDP.

Table A.2.4-1 BWR 123 Plant 1/2 Mitigating Systems Threshold Summary

BWR 123 Plant 1/2 SPAR 3i (3.7E-9/hr, 2.6E-5/calendar year[1]) BWR 123 Plant 1/2 RBPI Baseline (6.9E-10/hr, 4.8E-6/calendar year[1])					
System/Component	Baseline Train Unavailability and Unreliability	Green/White Threshold 95th %ile	Green/White Threshold (ΔCDF =1E-6/year)	White/Yellow Threshold (ΔCDF =1E-5/year)	Yellow/Red Threshold (ΔCDF =1E-4/year)
Emergency AC Power (EPS)	(Train Unreliability[2]) 6.0E-2	1.4E-1	4.5E-2	7.6E-2	1.6E-1
	(Train Unavailability) 9.7E-3	6.0E-2	5.5E-2	1.9E-1	Not Reached[6]
Isolation Condenser[3,5]	(Train Unreliability) 2.1E-2	6.2E-2	4.8E-2	Not Reached	Not Reached
Diesel Generator Emergency Service Water (DG ESW)	(Train Unreliability[2]) 2.2E-2	7.7E-2	3.4E-2	1.0E-1	3.5E-1
	(Standby Train Unavail.) 1.8E-2	7.2E-2	4.2E-2	2.1E-1	Not Reached
HPCI (HC1)	(Train Unreliability[2,4]) 5.3E-2	1.0E-1	4.5E-1	Not Reached	Not Reached
	(Train Unavailability) 9.7E-3	6.1E-2	3.9E-1	Not Reached	Not Reached
Residual Heat Removal (SPC)	(Train Unreliability) 1.5E-2	4.1E-2	2.4E-2	7.2E-2	2.2E-1
	(Train Unavailability) 1.0E-2	9.8E-3	Not Reached[6]	Not Reached[6]	Not Reached[6]
Containment Cooling Service Water (CSW to RHR HTX's)	(Train Unreliability) 2.2E-2	7.7E-2	9.1E-2	3.0E-1	6.0E-1
	(Train Unavailability) 1.8E-2	7.7E-2	Not Reached	Not Reached	Not Reached
AOVs	Component Class Unreliability	N/A	Increase 40X	Not Reached	Not Reached
MOVs[4]	Component Class Unreliability	N/A	Increase 3.3X	Increase 9.7X	Increase 19X
MDPs	Component Class Unreliability	N/A	Increase 3.9X	Increase 20X	Increase 65X

1. Calendar year is defined as 7000 critical hours.
2. Total unreliability, includes test and maintenance (TM) contribution.
3. TM not modeled in this system.
4. HPCI Injection valve reopening failure was excluded from consideration due to its unique failure mechanism and probability
5. Isolation Condenser unavailability not calculated. (page13, INEL-95/0478)
6. This threshold can be reached if the T&M outages associated with this system are not confined to TECH SPEC allowable combinations.

Table A.2.4-2 BWR 3/4 Plant 1 Mitigating Systems Threshold Summary

BWR 3/4 Plant 1 SPAR 3i (3.5E-10/hr, 2.4E-6/calendar year[1]) BWR 3/4 Plant 1 RBPI Baseline (3.1E-10/hr, 2.2E-6/calendar year[1])					
System/Component	Baseline Train Unavailability and Unreliability	Green/White Threshold 95^{th}%ile	Green/White Threshold (ΔCDF =1E-6/year)	White/Yellow Threshold (ΔCDF =1E-5/year)	Yellow/Red Threshold (ΔCDF =1E-4/year)
Emergency AC Power (EPS)	(Train Unreliability[2]) 4.4E-2	9.6E-2	3.7E-1)	5.2E-1	7.1E-1
	(Train Unavailability) 9.7E-3	6.2E-2	Not Reached[5]	Not Reached[5]	Not Reached[5]
Reactor Core Isolation Cooling (RCI)	(Train Unreliability[2,4]) 8.8E-2	1.7E-1	1.4E-1	5.1E-1	Not Reached
	(Train Unavailability) 1.3E-2	6.4E-2	1.6E-1	7.9E-1	Not Reached
Essential/Emergency Service Water (ESW to EDGs)	(Train Unreliability[2]) 2.2E-2	7.4E-2	7.8E-1	Not Reached	Not Reached
	(Train Unavailability) 1.8E-2	5.3E-2	Not Reached[6]	Not Reached[6]	Not Reached[6]
HSW (RHR Service Water)	(Train Unreliability[2]) 2.2E-2	7.2E-2	6.2E-1	8.3E-1	Not Reached
	(Train Unavailability) 1.8-2	8.1E-2	Not Reached[5]	Not Reached[5]	Not Reached[5]
HPCI (HCI)	(Train Unreliability[2,3]) 2.3E-1	4.4E-1	2.7E-1	5.5E-1	Not Reached
	(Train Unavailability) 9.7E-3	6.0E-2	3.1E-1	Not Reached	Not Reached
Residual Heat Removal (SPC)	(Train Unreliability[2]) 1.5E-2	3.0E-2	2.3E-1	Not Reached	Not Reached
	(Train Unavailability) 1.0E-2	9.2E-3	Not Reached[5]	Not Reached[5]	Not Reached[5]
AOVs	Component Class Unreliability	N/A	Increase 220X	Not Reached	Not Reached
MOVs[3]	Component Class Unreliability	N/A	Increase 3.6X	Increase 19X	Not Reached
MDPs	Component Class Unreliability	N/A	Increase 27X	Increase 120X	Increase 195X

1. Calendar year is defined as 7000 critical hours.
2. Total unreliability, includes test and maintenance (TM) contribution.
3. HPCI Injection valve reopening failure was excluded from consideration due to its unique failure mechanism and probability
4. RCIC turbine restart failure was excluded from consideration due to its unique failure mechanism and probability.
5. This threshold can be reached if the T&M outages associated with this system are not confined to TECH SPEC allowable combinations.
6. Thresholds not reached due to redundancy of ESW and HSW from all 3 plants.

Table A.2.4-3 BWR 3/4 Plant 2 Mitigating Systems Threshold Summary

BWR 3/4 Plant 2 SPAR 3i (5.1E-10/hr, 3.5E-6/calendar year[1]) BWR 3/4 Plant 2 RBPI Baseline (3.8E-10/hr, 2.7E-6/calendar year[1])					
System/Component	Baseline Train Unavailability and Unreliability	Green/White Threshold 95th %ile	Green/White Threshold (ΔCDF =1E-6/year)	White/Yellow Threshold (ΔCDF =1E-5/year)	Yellow/Red Threshold (ΔCDF =1E-4/year)
Emergency AC Power (EPS)	(Train Unreliability[2]) 4.4E-2	9.6E-2	8.9E-2	2.4E-1	4.9E-1
	(Train Unavailability) 9.7E-3	6.0E-2	4.2E-1	Not Reached[5]	Not Reached[5]
Reactor Core Isolation Cooling (RCI)	(Train Unreliability[2,4]) 7.9E-2	1.7E-1	1.2E-1	4.7E-1	Not Reached
	(Train Unavailability) 1.3E-2	2.9E-2	1.4E-1	7.2E-1	Not Reached
Essential/Emergency Service Water (ESW to EDGs)	(Train Unreliability[2]) 2.2E-2	7.0E-2	7.9E-1	Not Reached	Not Reached
	(Train Unavailability) 1.8E-2	5.3E-2	Not Reached[6]	Not Reached[6]	Not Reached[6]
HSW (RHR Service Water)	(Train Unreliability[2]) 2.2E-2	7.5E-2	1.2E-1	4.6E-1	7.2E-1
	(Train Unavailability) 1.8E-2	7.6E-2	8.9E-1	Not Reached[5]	Not Reached[5]
HPCI (HCI)	(Train Unreliability[2,3]) 2.3E-1	4.4E-1	2.7E-1	5.6E-1	Not Reached
	(Train Unavailability) 9.7E-3	5.9E-2	3.2E-1	Not Reached	Not Reached
Residual Heat Removal (SPC)	(Train Unreliability[2]) 1.5E-2	3.0E-2	2.3E-1	7.8E-1	Not Reached
	(Train Unavailability) 1.0E-2	1.1E-2	Not Reached[5]	Not Reached[5]	Not Reached[5]
AOVs	Component Class Unreliability	N/A	Increase 210X	Not Reached	Not Reached
MOVs[3]	Component Class Unreliability	N/A	Increase 8.1X	Increase 37X	Increase 76X
MDPs	Component Class Unreliability	N/A	Increase 22X	Increase 87X	Increase 150X

1. Calendar year is defined as 7000 critical hours.
2. Total unreliability, includes test and maintenance (TM) contribution.
3. HPCI Injection valve reopening failure was excluded from consideration due to its unique failure mechanism and probability
4. RCIC turbine restart failure was excluded from consideration due to its unique failure mechanism and probability.
5. This threshold can be reached if the T&M outages associated with this system are not confined to TECH SPEC allowable combinations.
6. Thresholds not reached due to redundancy of ESW and HSW from all 3 plants.

Table A.2.4-4 BWR 3/4 Plant 3/4 Mitigating Systems Threshold Summary

BWR 3/4 Plant 3/4 SPAR 3i (3.5E-9/hr, 2.4E-5/calendar year[1]) BWR 3/4 Plant 3/4 RBPI Baseline (2.2E-9/hr, 1.5E-5/calendar year[1])					
System/Component	Baseline Train Unavailability and Unreliability	Green/White Threshold 95th %ile	Green/White Threshold (ΔCDF =1E-6/year)	White/Yellow Threshold (ΔCDF =1E-5/year)	Yellow/Red Threshold (ΔCDF =1E-4/year)
Emergency AC Power (EPS)	(Train Unreliability[2]) 3.9E-2	9.7E-2	4.1E-2	5.9E-2	1.4E-1
	(Train Unavailability) 9.7E-3	5.8E-2	4.2E-2	7.0E-2	3.6E-1
Reactor Core Isolation Cooling (RCI)	(Train Unreliability[2,5]) 7.9E-2	1.7E-1	1.7E-1	7.6E-1	Not Reached
	(Train Unavailability) 1.3E-2	2.9E-2	1.8E-1	Not Reached	Not Reached
Nuclear Service Water (NSW to EDGs)	(Train Unreliability[2]) 2.5E-2	7.7E-2	2.7E-3	6.9E-2	8.4E-1
	(Standby Train Unavail) 1.8E-2	6.8E-2	Not Reached	Not Reached	Not Reached
HPCI (HCI)	(Train Unreliability[2,4]) 2.4E-1	4.4E-1	2.5E-1	3.7E-1	7.5E-1
	(Train Unavailability) 9.7E-3	5.8E-2	2.5E-1	4.2E-1	Not Reached
Residual Heat Removal (SPC)	(Train Unreliability[7]) 5.0E-3	1.3E-2	7.0E-3	2.2E-2	9.6E-2
	(Train Unavailability[7]) 2.9E-2	6.3E-2	Not Reached[6]	Not Reached[6]	Not Reached[6]
AOVs	Component Class Unreliability	N/A	Increase 7.1X	Increase 3.7X	Increase 126X
MOVs[4]	Component Class Unreliability	N/A	Increase 1.1X	Increase 1.3X	Increase 4.6X
MDPs	Component Class Unreliability	N/A	Increase 3.1X	Increase 17X	Increase 57X

1. Calendar year is defined as 7000 critical hours.
2. Total unreliability, includes test and maintenance (TM) contribution.
3. TM not modeled in this system.
4. HPCI Injection valve reopening failure was excluded from consideration due to its unique failure mechanism and probability
5. RCIC turbine restart failure was excluded from consideration due to its unique failure mechanism and probability.
6. This threshold can be reached if the T&M outages associated with this system are not confined to TECH SPEC allowable combinations.
7. RHR train for UA is defined at the heat exchanger (2 pump) level, RHR train for UR is defined at the individual pump level.

Table A.2.4-5 BWR 3/4 Plant 5 Mitigating Systems Threshold Summary

BWR 3/4 Plant 5 SPAR 3i (2.0E-0/hr, 1.4E-5/calendar year[1]) BWR 3/4 Plant 5 RBPI Baseline (2.2E-9/hr, 1.5E-5/calendar year[1])					
System/Component	Baseline Train Unavailability And Unreliability	Green/White Threshold 95^{th} %ile	Green/White Threshold (ΔCDF =1E-6/year)	White/Yellow Threshold (ΔCDF=1E-5/year)	Yellow/Red Threshold (ΔCDF=1E-4/year)
Emergency AC Power	(Train Unreliability[2]) 4.5E-2	2.2E-1	5.0E-2	7.9E-2	2.4E-1
	(Train Unavailability) 9.7E-3	1.9E-2	1.7E-2	8.5E-2	7.2E-1
Reactor Core Isolation Cooling[5]	(Train Unreliability[2]) 7.9E-2	1.0E-1	1.1E-1	3.2E-1	Not Reached
	(Train Unavailability) 1.3E-2	4.0E-2	5.0E-2	3.7E-1	Not Reached
Service Water (ESW to EDGs and RHR)	(Train Unreliability[2]) 2.3E-2	7.4E-2	2.8E-2	6.6E-2	2.3E-1
	(Train Unavailability) 1.8E-2	5.6E-2	2.6E-2	1.1E-1	8.4E-1
High-Pressure Coolant Injection	(Train Unreliability[2]) 2.2E-1	4.2E-1	2.3E-1	3.0E-1	Not Reached
	(Train Unavailability) 9.7E-3	3.8E-2	6.3E-2	5.3E-1	Not Reached
Residual Heat Removal (SPC)	(Train Unreliability[2,3]) 5.0E-3	9.9E-3	6.5E-3	1.8E-2	8.6E-2
	(Train Unavailability) 1.0E-2	2.5E-2	2.2E-2	1.2E-1	Not Reached[6]
AOVs	Component Class Unreliability	N/A	Increase 3.0X	Increase 19X	Increase 115X
MOVs[4]	Component Class Unreliability	N/A	Increase 1.2X	Increase 2.6X	Increase 7.3X
MDPs	Component Class Unreliability	N/A	Increase 1.6X	Increase 6.2X	Increase 38X

1. Calendar year is defined as 7000 critical hours. (ΔCDF is calculated in calendar years).
2. Total unreliability includes test and maintenance (TM) contribution.
3. RHR train for UA is defined at the heat exchanger (2 pump) level; RHR train for UR is defined at the individual pump level.
4. HPCI injection valve reopening failure was excluded from consideration due to its unique failure mechanism and probability.
5. RCIC turbine restart failure was excluded from consideration due to its unique failure mechanism and probability.
6. This threshold can be reached if the T&M outages associated with this system are not confined to TECH SPEC allowable combinations.

Table A.2.4-6 BWR 3/4 Plant 6 Mitigating Systems Threshold Summary

BWR 3/4 Plant 6 SPAR 3i (2.8E-9/hr, 2.0E-5/calendar year[1]) BWR 3/4 Plant 6 RBPI Baseline (2.4E-9/hr, 1.7E-5/calendar year[1])					
System/Component	Baseline Train Unavailability And Unreliability	Green/White Threshold 95th %ile	Green/White Threshold (ΔCDF =1E-6/year)	White/Yellow Threshold (ΔCDF =1E-5/year)	Yellow/Red Threshold (ΔCDF =1E-4/year)
Emergency AC Power	(Train Unreliability[2]) 3.9E-2	2.3E-1	4.5E-2	8.8E-2	2.8E-1
	(Train Unavailability) 9.7E-3	1.9E-2	2.3E-2	1.5E-1	Not Reached[5]
Reactor Core Isolation Cooling	(Train Unreliability[2,4]) 7.9E-2	1.6E-1	1.7E-1	7.6E-1	Not Reached
	(Train Unavailability) 1.3E-2	4.0E-2	5.3E-2	Not Reached[5]	Not Reached
Essential/Emergency Service Water (to EDGs)	(Train Unreliability[2]) 2.5E-2	8.3E-2	2.7E-2	4.4E-2	1.6E-1
	(Train Unavailability) 1.9E-2	7.7E-2	2.3E-2	5.9E-2	1.6E-1
High-Pressure Coolant Injection (HPCI)	(Train Unreliability[2,3]) 2.4E-1	4.4E-1	2.8E-1	6.3E-1	Not Reached
	(Train Unavailability) 9.7E-3	3.8E-2	9.0E-2	8.1E-1	Not Reached
Residual Heat Removal (SPC, includes SSW)	(System Unreliability[2]) 1.7E-2	3.2E-2	2.2E-2	6.6E-2	3.0E-1
	(Train Unavailability) 1.0E-2	2.5E-2	3.9E-2	3.8E-1	Not Reached[5]
AOVs	Component Class Unreliability	N/A	Increase 1.25X	Increase 3.5X	Increase 17X
MOVs[3]	Component Class Unreliability	N/A	Increase 1.54X	Increase 5.3X	Increase 21X
MDPs	Component Class Unreliability	N/A	Increase 1.27X	Increase 3.5X	Increase 17X

1. Calendar year is defined as 7000 critical hours.
2. Total unreliability; includes test and maintenance (TM) contribution.
3. HPCI Injection valve reopening failure was excluded from consideration due to its unique failure mechanism and probability.
4. RCIC turbine restart failure was excluded from consideration due to its unique failure mechanism and probability.
5. This threshold can be reached if the T&M outages associated with this system are not confined to TECH SPEC allowable combinations.

Table A.2.4-7 BWR 3/4 Plant 8 Mitigating Systems Threshold Summary

BWR 3/4 Plant 8 SPAR 3i (8.7E-10/hr, 6.1E-6/calendar year[1]) BWR 3/4 Plant 8 RBPI Baseline (7.6E-10/hr, 5.3E-6/calendar year[1])					
System/Component	Baseline Train Unavailability And Unreliability	Green/White Threshold 95th %ile	Green/White Threshold (ΔCDF =1E-6/year)	White/Yellow Threshold (ΔCDF =1E-5/year)	Yellow/Red Threshold (ΔCDF =1E-4/year)
Emergency AC Power	(Train Unreliability[2]) 4.5E-2	2.3E-1	6.3E-2	1.7E-1	4.3E-1
	(Train Unavailability) 9.7E-3	1.9E-2	2.4E-1	Not Reached[6]	Not Reached[6]
Reactor Core Isolation Cooling	(Train Unreliability[2,5]) 7.9E-2	1.6E-1	1.2E-1	3.9E-1	Not Reached
	(Train Unavailability) 1.3E-2	4.0E-2	6.4E-2	5.3E-1	Not Reached
Essential Service Water (ESW to EDGs)	(Train Unreliability[2]) 2.3E-2	7.9E-2	2.9E-2	6.9E-2	2.5E-1
	(Train Unavailability) 1.8E-2	6.5E-2	6.5E-3	5.0E-2	Not Reached[6]
High-Pressure Coolant Injection (HPCI)	(Train Unreliability[2,4]) 2.4E-1	4.4E-1	2.7E-1	5.4E-1	Not Reached
	(Train Unavailability) 9.7E-3	3.8E-2	1.2E-1	Not Reached	Not Reached
Residual Heat Removal (SPC)	(Train Unreliability[2]) 5.1E-3	1.0E-2	2.4E-2	9.18E-2	2.7E-1
	(Train Unavailability) 1.0E-2	2.5E-2	1.4E-1	Not Reached[6]	Not Reached[6]
RHR Service Water (HSW)	(Train Unreliability[2,3]) 2.2E-2	7.7E-2	5.5E-2	2.1E-1	4.7E-1
	(Train Unavailability) 1.8E-2	7.1E-2	Not Reached[6]	Not Reached[6]	Not Reached[6]
AOVs	Component Class Unreliability	N/A	Increase 3.5X	Increase 24X	Increase 170X
MOVs	Component Class Unreliability	N/A	Increase 2.4X	Increase 8.4X	Increase 21X
MDPs	Component Class Unreliability	N/A	Increase 2.0X	Increase 9.9X	Increase 30X

1. Calendar year is defined as 7000 critical hours.
2. Total unreliability includes test and maintenance (TM) contribution.
3. RHR train for UA is defined at the heat exchanger (2 pump) level; RHR train for UR is defined at the individual pump level.
4. HPCI injection valve reopening failure was excluded from consideration due to its unique failure mechanism and probability.
5. RCIC turbine restart failure was excluded from consideration due to its unique failure mechanism and probability.
6. This threshold can be reached if the T&M outages associated with this system are not confined to TECH SPEC allowable combinations.

Table A.2.4-8 BWR 3/4 Plant 11 Mitigating Systems Threshold Summary

BWR 3/4 Plant 11 SPAR 3i (4.9E-9/hr, 3.4E-5/calendar year[1]) BWR 3/4 Plant 11 RBPI Baseline (5.6E-9/hr, 3.9E-5/calendar year[1])					
System/Component	Baseline Train Unavailability And Unreliability	Green/White Threshold 95th %ile	Green/White Threshold (ΔCDF =1E-6/year)	White/Yellow Threshold (ΔCDF =1E-5/year)	Yellow/Red Threshold (ΔCDF =1E-4/year)
Emergency AC Power	(Train Unreliability[2]) 4.6E-2	1.0E-1	5.2E-2	9.2E-2	2.6E-1
	(Train Unavailability) 9.7E-3	1.9E-2	1.9E-2	1.1E-1	9.5E-1
Reactor Core Isolation Cooling	(Train Unreliability[2,6]) 7.9E-2	1.8E-1	8.2E-2	1.0E-1	3.2E-1
	(Train Unavailability) 1.3E-2	4.0E-2	1.6E-2	4.7E-2	3.6E-1
Safety Auxiliaries Cooling Water (SACs) (Cools EDGs & RHR)	(Train Unreliability[2]) 3.8E-3	8.8E-3	2.2E-2	1.3E-1	3.8E-1
	(Train Unavailability) N/A[3]	N/A[3]	N/A[3]	N/A[3]	N/A[3]
High-Pressure Coolant Injection (HPCI)	(Train Unreliability[2,5]) 2.4E-1	4.4E-1	2.4E-1	2.7E-1	5.4E-1
	(Train Unavailability) 9.7E-3	3.8E-2	2.4E-2	3.1E-1	9.9E-1
Residual Heat Removal (SPC)	(Train Unreliability[2]) 2.2E-2	3.8E-2	3.8E-2	1.3E-1	4.5E-1
	(Train Unavailability) 1.0E-2	2.5E-2	2.6E-2	1.8E-1	Not Reached[7]
Station Service Water (SSW)	(Train Unreliability[2]) 3.2E-2	8.5E-2	3.5E-2	5.9E-2	1.6E-1
	(Train Unavailability) 1.8E-2	6.8E-2	2.3E-2	6.5E-2	4.9E-1
AOVs[4]	Component Class Unreliability	N/A	Increase 1.2X	Increase 3.3X	Increase 23X
MOVs	Component Class Unreliability	N/A	Increase 1.1X	Increase 2.2X	Increase 8.4X
MDPs	Component Class Unreliability	N/A	Increase 1.4X	Increase 4.0X	Increase 17X

1. Calendar year is defined as 7000 critical hours.
2. Total unreliability; includes test and maintenance (TM) contribution.
3. N/A – T&M events not included in SPAR logic.
4. AOVs System do not include failure of the reliefs to re-close.
5. HPCI Injection valve reopening failure was excluded from consideration due to its unique failure mechanism and probability.
6. RCIC turbine restart failure was excluded from consideration due to its unique failure mechanism and probability.
7. This threshold can be reached if the T&M outages associated with this system are not confined to TECH SPEC allowable combinations.

Table A.2.4-9 BWR 3/4 Plant 12/13 Mitigating Systems Threshold Summary

BWR 3/4 Plant 12/13 SPAR 3i (4.2E-9/hr, 3.0E-5/calendar year[1]) BWR 3/4 Plant 12/13 RBPI Baseline (3.4E-9/hr, 2.4E-5/calendar year[1])					
System/Component	Baseline Train Unavailability and Unreliability	Green/White Threshold 95th %ile	Green/White Threshold (ΔCDF =1E-6/year)	White/Yellow Threshold (ΔCDF =1E-5/year)	Yellow/Red Threshold (ΔCDF =1E-4/year)
Emergency AC Power (EPS)	(Train Unreliability[2]) 3.8E-2	9.1E-2	4.3E-2	7.5E-2	2.1E-1
	(Train Unavailability) 9.7E-3	6.2E-2	4.7E-2	1.1E-1	5.9E-1
Reactor Core Isolation Cooling (RCI)	(Train Unreliability[2,4]) 7.9E-2	1.7E-1	8.6E-2	1.5E-1	6.5E-1
	(Train Unavailability) 1.3E-2	2.9E-2	8.7E-2	1.6E-1	9.0E-1
Essential/Emergency Service Water (ESW to EDGs)	(Train Unreliability[2]) 2.2E-2	7.7E-2	7.6E-2	1.81E-1	3.5E-1
	(Standby Train Unavail.) 1.8E-2	5.6E-2	8.7E-2	2.7E-1	7.4E-1
HPCI (HCI)	(Train Unreliability[2,3]) 2.3E-1	4.4E-1	2.4E-1	2.7E-1	5.4E-1
	(Train Unavailability) 9.7E-3	5.8E-2	2.4E-1	2.8E-1	6.9E-1
Residual Heat Removal (SPC)	(Train Unreliability[6]) 2.1E-2	3.8E-2	3.2E-2	1.1E-1	3.4E-1
	(Train Unavailability) 1.0E-2	9.9E-3	1.7E-1	Not Reached[5]	Not Reached[5]
AOVs	Component Class Unreliability	N/A	Increase 4.1X	Increase 30X	Not Reached
MOVs	Component Class Unreliability	N/A	Increase 1.2X	Increase 3.0X	Increase 9.5X
MDPs	Component Class Unreliability	N/A	Increase 5.5X	Increase 28X	Increase 80X

1. Calendar year is defined as 7000 critical hours.
2. Total unreliability, includes test and maintenance (TM) contribution.
3. HPCI Injection valve reopening failure was excluded from consideration due to its unique failure mechanism and probability
4. RCIC turbine restart failure was excluded from consideration due to its unique failure mechanism and probability.
5. This threshold can be reached if the T&M outages associated with this system are not confined to TECH SPEC allowable combinations.
6. RHR train for UA is defined at the heat exchanger (2 pump) level, RHR train for UR is defined at the individual pump level.

Table A.2.4-10 BWR 3/4 Plant 15/16 Mitigating Systems Threshold Summary

BWR Plant 15/16 SPAR 3i (5.8E-10/hr, 4.1E-6/calendar year[1]) BWR Plant 15/16 RBPI Baseline (5.3E-10/hr, 3.7E-6/calendar year[1])					
System/Component	Baseline Train Unavailability And Unreliability	Green/White Threshold 95^{th} %ile	Green/White Threshold (ΔCDF =1E-6/year)	White/Yellow Threshold (ΔCDF =1E-5/year)	Yellow/Red Threshold (ΔCDF =1E-4/year)
Emergency AC Power	(Train Unreliability[2]) 3.9E-2	9.5E-2	1.4E-1	3.2E-1	6.7E-1
	(Train Unavailability) 9.7E-3	1.9E-2	7.5E-1	Not Reached[5]	Not Reached[5]
Reactor Core Isolation Cooling (RCI)	(Train Unreliability[2,4]) 7.9E-2	1.6E-1	2.0E-1	8.5E-1	Not Reached
	(Train Unavailability) 1.3E-2	4.0E-2	2.0E-1	Not Reached	Not Reached
Essential/Emergency Service Water (to EDGs)	(Train Unreliability[2]) 8.5E-3	2.1E-2	6.3E-2	1.5E-1	3.5E-1
	(Train Unavailability) 2.0E-3	7.8E-3	5.8E-1	Not Reached[5]	Not Reached[5]
High-Pressure Coolant Injection (HPCI)	(Train Unreliability[2,3]) 2.4E-1	4.5E-1	3.2E-1	7.5E-1	Not Reached
	(Train Unavailability) 9.7E-3	3.8E-2	2.6E-1	Not Reached	Not Reached
Residual Heat Removal (SPC)	(Train Unreliability[2]) 1.2E-2	2.7E-2	5.7E-2	2.3E-1	4.7E-1
	(Train Unavailability) 2.0E-3	8.0E-3	Not Reached[5]	Not Reached[5]	Not Reached[5]
High Pressure Service Water (HSW)	(Train Unreliability[2]) 8.2E-3	1.9E-2	2.7E-2	1.5E-1	4.1E-1
	(Train Unavailability) 2.0E-3	8.5E-3	5.8E-1	Not Reached[5]	Not Reached[5]
AOVs	Component Class Unreliability	N/A	Increase 1.7X	Increase 8.1X	Increase 59X
MOVs	Component Class Unreliability	N/A	Increase 3.5X	Increase 14X	Increase 35X
MDPs	Component Class Unreliability	N/A	Increase 3.4X	Increase 17X	Increase 47X

1. Calendar year is defined as 7000 critical hours.
2. Total unreliability; includes test and maintenance (TM) contribution.
3. HPCI Injection valve reopening failure was excluded from consideration due to its unique failure mechanism and probability.
4. RCIC turbine restart failure was excluded from consideration due to its unique failure mechanism and probability.
5. This threshold can be reached if the T&M outages associated with this system are not confined to TECH SPEC allowable combinations.

Table A.2.4-11 BWR 3/4 Plant 18/19 Mitigating Systems Threshold Summary

BWR 3/4 Plant 18/19 SPAR 3i (3.7E-9/hr, 2.6E-5/calendar year[1]) BWR 3/4 Plant 18/19 RBPI Baseline (2.9E-9/hr, 2.0E-5/calendar year[1])					
System/Component	Baseline Train Unavailability and Unreliability	Green/White Threshold 95th %ile	Green/White Threshold (ΔCDF =1E-6/year)	White/Yellow Threshold (ΔCDF =1E-5/year)	Yellow/Red Threshold (ΔCDF =1E-4/year)
Emergency AC Power	(Train Unreliability[2]) 4.0E-2	9.9E-2	4.2E-2	5.8E-2	1.5E-1
	(Train Unavailability) 9.7E-3	1.9E-2	1.4E-2	4.9E-2	3.9E-1
Reactor Core Isolation Cooling (RCIC)	(Train Unreliability[2,5]) 7.9E-2	1.7E-1	9.1E-2	2.0E-1	Not Reached
	(Train Unavailability) 1.3E-2	4.0E-2	2.8E-2	1.7E-1	Not Reached
Essential/Emergency Service Water (to EDGs)	(Train Unreliability[2]) 2.5E-2	8.0E-2	2.7E-2	4.2E-2	1.3E-1
	(Standby Train Unavail.) 1.9E-2	5.4E-2	2.2E-2	5.6E-2	3.9E-1
High-Pressure Coolant Injection (HPCI)	(Train Unreliability[2,4]) 2.4E-1	4.3E-1	2.6E-1	4.6E-1	Not Reached
	(Train Unavailability) 9.7E-3	3.8E-2	8.2E-2	7.3E-1	Not Reached
Residual Heat Removal (SPC)	(Train Unreliability[7]) 8.8E-3	2.3E-2	2.0E-2	6.8E-2	2.2E-1
	(Train Unavailability) 1.0E-2	2.5E-2	1.4E-1	Not Reached[6]	Not Reached[6]
AOVs	Component Class Unreliability	N/A	Increase 2.2X	Increase 13X	Increase 83X
MOVs	Component Class Unreliability	N/A	Increase 1.7X	Increase 7.0X	Increase 28X
MDPs	Component Class Unreliability	N/A	Increase 1.2X	Increase 5.1X	Increase 28X

1. Calendar year is defined as 7000 critical hours.
2. Total unreliability; includes test and maintenance (TM) contribution.
3. TM not modeled in this system.
4. HPCI Injection valve reopening failure was excluded from consideration due to its unique failure mechanism and probability
5. RCIC turbine restart failure was excluded from consideration due to its unique failure mechanism and probability.
6. This threshold can be reached if the T&M outages associated with this system are not confined to TECH SPEC allowable combinations.
7. RHR train for UA is defined at the heat exchanger (2 pump) level; RHR train for UR is defined at the individual pump level.

Table A.2.4-12 BWR 5/6 Plant 2 Mitigating Systems Threshold Summary

BWR 5/6 Plant 2 SPAR 3i (1.2E-9/hr, 8.6E-6/calendar year[1]) BWR 5/6 Plant 2 RBPI Baseline (1.4E-9/hr, 9.9E-6/calendar year[1])					
System/Component	Baseline Train Unavailability And Unreliability	Green/White Threshold 95th %ile	Green/White Threshold (ΔCDF =1E-6/year)	White/Yellow Threshold (ΔCDF =1E-5/year)	Yellow/Red Threshold (ΔCDF =1E-4/year)
Emergency AC Power	(Train Unreliability[2]) 4.1E-2	9.9E-2	4.8E-2	9.9E-2	3.7E-1
	(Train Unavailability) 9.7E-3	1.9E-2	3.2E-2	2.2E-1	Not Reached[4]
High-Pressure Core Spray (HPCS)	(Train Unreliability[2]) 1.0E-1	1.4E-1	1.5E-1	5.0E-1	Not Reached
	(Train Unavailability) 3.4E-3	1.2E-2	6.1E-2	5.4E-1	Not Reached
Reactor Core Isolation Cooling (RCI)	(Train Unreliability[2]) 7.9E-2	1.6E-1	1.4E-1	5.7E-1	Not Reached
	(Train Unavailability) 1.3E-2	4.0E-2	8.3E-2	7.0E-1	Not Reached
Residual Heat Removal (RHR, SPC)	(Train Unreliability[2]) 2.4E-2	4.2E-2	2.9E-2	7.1E-2	3.3E-1
	(Train Unavailability) 1.0E-2	2.5E-2	1.5E-2	5.9E-2	4.9E-1
Standby Service Water (SSW)	(Train Unreliability[2]) 1.5E-2	3.0E-2	2.0E-2	5.2E-2	2.2E-1
	(Train Unavailability) 2.0E-3	7.5E-3	2.8E-2	1.4E-1	Not Reached
AOVs	Component Class Unreliability	N/A	Increase 1.7X	Increase 7.5X	Increase 55X
MOVs	Component Class Unreliability	N/A	Increase 1.2X	Increase 2.3X	Increase 6.9X
MDPs	Component Class Unreliability	N/A	Increase 1.5X	Increase 5.0X	Increase 21X

1. Calendar year is defined as 7000 critical hours.
2. Total unreliability; includes test and maintenance (TM) contribution.
3. This threshold can be reached if the T&M outages associated with this system are not confined to TECH SPEC allowable combinations.

Table A.2.4-13 BWR 5/6 Plant 5 Mitigating Systems Threshold Summary

BWR 5/6 Plant 5 SPAR 3i (1.5E-9/hr, 1.1E-5/calendar year[1]) BWR 5/6 Plant 5 RBPI Baseline (1.2E-9/hr, 8.5E-6/calendar year[1])					
System/Component	Baseline Train Unavailability and Unreliability	Green/White Threshold 95th%ile	Green/White Threshold (ΔCDF =1E-6/year)	White/Yellow Threshold (ΔCDF =1E-5/year)	Yellow/Red Threshold (ΔCDF =1E-4/year)
Emergency AC Power (EPS)	(Train Unreliability[2]) 4.9E-2	1.3E-1	6.7E-2	1.5E-1	4.5E-1
	(Train Unavailability) 9.7E-3	6.3E-2	8.5E-2	4.1E-1	Not Reached[3]
Reactor Core Isolation Cooling (RCI)	(Train Unreliability[2]) 7.9E-2	1.7E-1	1.5E-1	7.9E-2	Not Reached
	(Train Unavailability) 1.3E-2	2.9E-2	1.7E-1	9.6E-1	Not Reached
Essential/Emergency Service Water (SWS to EDGs)	(Train Unreliability[2]) 2.3E-2	7.2E-2	4.5E-2	2.1E-1	5.3E-1
	(Standby Train Unavail.) 1.9E-2	6.2E-2	Not Reached[3]	Not Reached[3]	Not Reached[3]
HPCS (HCS)	(Train Unreliability[2])6.3E-2	1.4E-1	1.1E-1	4.2E-1	Not Reached
	(Train Unavailability) 3.4E-3	1.0E-2	1.1E-1	5.4E-1	Not Reached
Residual Heat Removal (SPC)	(Train Unreliability[2])2.7E-2	4.7E-2	3.9E-2	1.0E-1	3.3E-1
	(Train Unavailability) 1.0E-2	3.0E-2	5.9E-2	3.5E-1	Not Reached[3]
AOVs	Component Class Unreliability	N/A	Increase 1.3X	Increase 4.1X	Increase 33X
MOVs	Component Class Unreliability	N/A	Increase 1.9X	Increase 5.8X	Increase 17X
MDPs	Component Class Unreliability	N/A	Increase 3.9X	Increase 21X	Increase 80X

1. Calendar year is defined as 7000 critical hours.
2. Total unreliability, includes test and maintenance (TM) contribution.
3. This threshold can be reached if the T&M outages associated with this system are not confined to TECH SPEC allowable combinations.

Table A.2.4-14 BWR 5/6 Plant 8 Mitigating Systems Threshold Summary

BWR 5/6 Plant 8 SPAR 3i (3.7E-9/hr, 2.6E-5/calendar year[1]) BWR 5/6 Plant 8 RBPI Baseline (7.9E-9/hr, 5.6E-5/calendar year[1])					
System/Component	Baseline Train Unavailability and Unreliability	Green/White Threshold 95th%ile	Green/White Threshold (ΔCDF =1E-6/year)	White/Yellow Threshold (ΔCDF =1E-5/year)	Yellow/Red Threshold (ΔCDF =1E-4/year)
Emergency AC Power (EPS)	(Train Unreliability[2]) 4.1E-2	9.6E-2	4.7E-2	9.3E-2	3.3E-1
	(Train Unavailability) 9.7E-3	6.1E-2	6.6E-2	2.8E-1	Not Reached[3]
Reactor Core Isolation Cooling (RCI)	(Train Unreliability[2]) 7.9E-2	NOTE 6	1.0E-1	2.8E-1	Not Reached
	(Train Unavailability) 1.3E-2	2.6E-2	1.0E-1	3.1E-1	Not Reached
Essential/Emergency Service Water (SWS to EDGs)	(Train Unreliability[2]) 7.0E-3	1.5E-2	7.2E-3	9.2E-3	2.6E-2
	(Standby Train Unavail) 2.0E-3	7.0E-3	8.6E-3	2.2E-2	1.5E-1
HPCS (HCS)	(Train Unreliability[2]) 7.1E-2	NOTE 6	7.2E-2	8.7E-2	2.3E-1
	(Train Unavailability) 3.4E-3	7.5E-3	7.2E-2	8.7E-2	2.3E-1
Residual Heat Removal (SPC)	(Train Unreliability[2])2.8E-2	NOTE 7	4.0E-2	8.2E-2	3.4E-1
	(Train Unavailability) 1.0E-2	8.5E-3	4.1E-2	9.9E-2	6.8E-1
AOVs	Component Class Unreliability	N/A	Increase 2.4X	Increase 14X	Increase 109X
MOVs	Component Class Unreliability	N/A	Increase 1.0X	Increase 1.4X	Increase 4.4X
MDPs	Component Class Unreliability	N/A	Increase 3.4X	Increase 4.2X	Increase 10X

1. Calendar year is defined as 7000 critical hours.
2. Total unreliability, includes test and maintenance (TM) contribution.
3. This threshold can be reached if the T&M outages associated with this system are not confined to TECH SPEC allowable combinations.

Table A.2.4-15 B&W Plant 3 Mitigating Systems Threshold Summary

B&W Plant 3 SPAR 3i (2.2E-9/hr, 1.6E-5/calendar year) B&W Plant 3 RBPI Baseline (2.3E-9/hr, 1.6E-5/calendar year)					
System	Baseline Train Unavailability and Unreliability	Green/White Threshold 95^{th} %ile	Green/White Threshold (ΔCDF =1E-6/year)	White/Yellow Threshold (ΔCDF =1E-5/year)	Yellow/Red Threshold (ΔCDF =1E-4/year)
Auxiliary Feedwater	(MDP Train Unreliability[1]) 1.8E-2	3.8E-2	2.3E-2	6.7E-2	4.6E-1
	(TDP Train Unreliability[1]) 1.9E-1	3.4E-1	2.2E-1	3.5E-1	Not Reached
	(MDP Train Unavailability) 1.1E-3	2.7E-2	2.3E-2	6.9E-2	Not Reached
	(TDP Train Unavailability) 4.6E-3	2.9E-2	2.2E-1	4.1E-1	Not Reached
Component Cooling Water	(Train Unreliability[1]) 1.6E-2	4.8E-2	2.3E-2	6.6E-2	2.3E-1
	(Standby Train Unavail.[4]) 1.1E-2	4.3E-2	4.9E-2	3.4E-1	Not Reached
Emergency AC Power	(Train Unreliability[1]) 4.1E-2	9.7E-2	5.2E-2	1.2E-1	3.1E-1
	(Train Unavailability) 9.7E-3	6.5E-2	7.3E-2	3.6E-1	Not Reached[5]
Chemical and Volume Control System	(SI Train Unreliability[1]) 2.2E-1	2.8E-1	5.6E-1	Not Reached	Not Reached
	(Train Unavailability) 2.0E-1	2.6E-1	Not Reached	Not Reached	Not Reached
High Pressure Injection	(SI Train Unreliability[1]) 1.3E-2	1.3E-2	3.5E-2	1.6E-1	5.3E-1
	(SI Train Unavailability) 4.2E-3	3.2E-2	5.4E-2	4.5E-1	Not Reached
Power Operated Relief Valves	(System Unreliability) 6.4E-3	2.3E-2	1.7E-2	1.1E-1	Not Reached
	(Train Unavailability) N/A[3]	N/A[3]	N/A[3]	N/A[3]	N/A[3]
Residual/Decay Heat Removal	(Train Unreliability[1]) 1.2E-2	3.0E-2	1.8E-2	6.9E-2	3.6E-1
	(Train Unavailability) 7.3E-3	2.1E-2	7.6E-2	6.6E-1	Not Reached[5]
Service Water[4]	(Train Unreliability[1]) 3.3E-2	9.6E-2	4.7E-2	1.6E-1	5.0E-1
	(Train Unavailability) 2.7E-2	8.0E-2	Not Reached[5]	Not Reached[5]	Not Reached[5]
AOVs[2]	Component Class Unreliability	N/A	Increase 21X	Increase 125X	Increase 495X
MOVs	Component Class Unreliability	N/A	Increase 2.2X	Increase 11X	Increase 47X
MDPs	Component Class Unreliability	N/A	Increase 3.4X	Increase 10X	Increase 36X

1. Total unreliability, includes T&M
2. AOV component class does not include failure to re-close of the reliefs.
3. N/A, T&M events not included in SPAR logic.
4. The primary SWS load is the RHR HTX.
5. This threshold can be reached if the T&M outages associated with this system are not confined to TECH SPEC allowable combinations.

Table A.2.4-16 B&W Plant 4/5/6 Mitigating Systems Threshold Summary

B&W Plant 4/5/6 SPAR 3i (2.1E-9/hr, 1.5E-5/calendar year[1]) B&W Plant 4/5/6 RBPI Baseline (2.5E-9/hr, 1.7E-5/calendar year[1])					
System/Component	Baseline Train Unavailability and Unreliability	Green/White Threshold 95^{th} %ile	Green/White Threshold (ΔCDF =1E-6/year)	White/Yellow Threshold (ΔCDF =1E-5/year)	Yellow/Red Threshold (ΔCDF =1E-4/year)
Emergency Feedwater	(MDP Train Unreliability[3]) 1.0E-2	2.3E-2	1.6E-2	5.2E-2	1.9E-1
	(TDP Train Unreliability[3]) 2.0E-1	3.5E-1	2.8E-1	8.9E-1	Not Reached
	(MDP Train Unavailability) 1.1E-3	2.5E-3	3.0E-2	2.8E-1	Not Reached
	(TDP Train Unavailability) 4.6E-3	1.8E-2	1.0E-1	9.9E-1	Not Reached
Component Cooling System (CCS)	(Train Unreliability[3]) 1.6E-2	5.1E-2	Not Reached	Not Reached	Not Reached
	(Train Unavailability) 1.1E-2	4.4E-2	Not Reached	Not Reached	Not Reached
Emergency AC Power	(Hydro Train Unreliability[3]) 1.1E-3	1.6E-3	2.4E-3	1.4E-2	1.3E-1
	(Hydro Train Unavailability) 1.4E-2	TBD[5]	1.4E-1	Not Reached[4]	Not Reached[4]
High Pressure Injection (HPI)	(Train Unreliability[3]) 1.3E-2	2.5E-2	6.0E-2	1.8E-1	5.1E-1
	(Train Unavailability) 4.2E-3	1.6E-2	5.5E-1	Not Reached	Not Reached
Decay Heat Removal (DHR)	(Train Unreliability[3]) 2.2E-2	5.5E-2	3.0E-2	1.1E-1	4.9E-1
	(Train Unavailability) 7.3E-3	2.4E-2	3.9E-1	Not Reached[4]	Not Reached[5]
Low Pressure Service Water (LSW)	(Train Unreliability[3]) 3.2E-2	9.2E-2	5.6E-2	1.5E-1	4.5E-1
	(Standby Train Unavailability) 2.7E-2	1.0E-1	5.4E-1	Not Reached	Not Reached
AOVs[2]	Component Class Unreliability	N/A	Increase 4.5X	Increase 25X	Increase 110X
MOVs	Component Class Unreliability	N/A	Increase 2.0X	Increase 8.7X	Increase 30X
MDPs	Component Class Unreliability	N/A	Increase 1.8X	Increase 6.7X	Increase 26X

1. Calendar year is defined as 7000 critical hours.
2. AOVs does not include failure to re-close of the reliefs.
3. Total unreliability, includes T&M
4. This threshold can be reached if the T&M outages associated with this system are not confined to TECH SPEC allowable combinations.
5. The corresponding unavailability event in the SPAR model does not include a probability distribution.

Table A.2.4-17 B&W Plant 7 Mitigating Systems Threshold Summary

B&W Plant 7 SPAR 3i (1.9E-9/hr, 1.4E-5/calendar year) B&W Plant 7 RBPI Baseline (1.8E-9/hr, 1.2E-5/calendar year)					
System	Baseline Train Unavailability and Unreliability	Green/White Threshold 95^{th}%ile	Green/White Threshold (ΔCDF =1E-6/year)	White/Yellow Threshold (ΔCDF =1E-5/year)	Yellow/Red Threshold (ΔCDF =1E-4/year)
Emergency Feedwater	(MDP Train Unreliability[1]) 8.6E-3	2.0E-2	2.7E-2	1.2E-1	4.7E-1
	(TDP Train Unreliability[1]) 1.9E-1	3.5E-1	2.5E-1	7.2E-1	Not Reached
	(MDP Train Unavailability) 1.1E-3	2.6E-3	4.6E-2	3.8E-1	Not Reached
	(TDP Train Unavailability) 4.6E-3	3.0E-2	2.4E-1	6.6E-1	Not Reached
Decay Heat Removal Closed Cooling	(Train Unreliability) 1.6E-2	4.9E-2	3.3E-2	1.3E-1	4.8E-1
	(Train Unavailability) 1.1E-2	4.0E-2	6.8E-2	5.5E-1	Not Reached[4]
Decay Heat River Water	(Train Unreliability[1]) 1.9E-2	5.1E-2	1.1E-1	4.4E-1	Not Reached
	(Train Unavailability) 1.1E-2	3.6E-2	4.3E-1	Not Reached	Not Reached
Emergency AC Power	(Train Unreliability[1]) 4.2E-2	1.0E-1	4.8E-2	8.6E-2	2.2E-1
	(Train Unavailability) 9.7E-3	6.3E-2	7.2E-2	3.4E-1	Not Reached[4]
High Pressure Injection	(SI Train Unreliability[1]) 9.3E-3	2.0E-2	1.2E-1	2.8E-1	5.7E-1
	(SI Train Unavailability) 4.2E-3	3.9E-2	7.4E-1	Not Reached	Not Reached
Nuclear Service Closed Cooling Water	(Train Unreliability[1]) 3.2E-2	9.2E-2	3.4E-1	7.7E-1	Not Reached
	(Train Unavailability) 2.7E-2	8.7E-2	Not Reached[4]	Not Reached[4]	Not Reached[4]
Nuclear Service River Water	(Train Unreliability[1]) 3.5E-2	8.8E-2	1.2E-1	5.7E-1	Not Reached
	(Train Unavailability) 2.7E-2	8.4E-2	Not Reached	Not Reached	Not Reached
Power Operated Relief Valves	(System Unreliability[1]) 6.4E-3	6.3E-3	3.2E-2	2.5E-1	Not Reached
	(Train Unavailability) N/A[3]	N/A[3]	N/A[3]	N/A[3]	N/A[3]
Residual Heat Removal (DHR)	(Train Unreliability[1]) 1.2E-2	8.1E-2	1.8E-2	5.9E-2	2.9E-1
	(Train Unavailability) 7.3E-3	2.8E-2	6.5E-2	5.4E-1	Not Reached[4]
AOVs[2]	Component Class Unreliability	N/A	Increase 72X	Increase 237X	Increase 497X
MOVs	Component Class Unreliability	N/A	Increase 1.7X	Increase 6.6X	Increase 27X
MDPs	Component Class Unreliability	N/A	Increase 2.2X	Increase 9.1X	Increase 34X

1. Total unreliability, includes T&M
2. AOV component class does not include failure to re-close of the reliefs.
3. N/A, T&M events not included in SPAR logic.
4. This threshold can be reached if the T&M outages associated with this system are not confined to TECH SPEC allowable combinations.

Table A.2.4-18 CE Plant 1 Mitigating Systems Threshold Summary

CE Plant 1 SPAR 3i (4.2E-9/hr, 3.0E-5/calendar year) CE Plant 1 RBPI Baseline (4.2E-9/hr, 3.0E-5/calendar year)					
System	Baseline Train Unavailability and Unreliability	Green/White Threshold 95^{th} %ile	Green/White Threshold (ΔCDF = 1E-6/year)	White/Yellow Threshold (ΔCDF = 1E-5/year)	Yellow/Red Threshold (ΔCDF = 1E-4/year)
Auxiliary Feedwater[6,8]	(MDP Train Unreliability[1]) 1.8E-2	5.2E-2	2.1E-2	4.7E-2	2.9E-1
	(MDP Train Unavailability) 1.0E-2	4.0E-2	2.1E-2	4.9E-2	3.2E-1
Emergency Feedwater	(MDP Train Unreliability[1]) 8.3E-3	2.1E-2	1.1E-2	3.3E-2	2.3E-1
	(TDP Train Unreliability[1]) 1.9E-1	3.5E-1	2.0E-1	2.8E-1	Not Reached
	(MDP Train Unavailability) 1.1E-3	2.5E-3	1.1E-2	4.0E-2	3.2E-1
	(TDP Train Unavailability) 4.6E-3	2.7E-2	2.0E-1	2.9E-1	Not Reached
Component Cooling Water[7]	(Train Unreliability) NA[7]	NA[7]	NA[7]	NA[7]	NA[7]
	(Standby Train Unavailability) NA[7]	NA[7]	NA[7]	NA[7]	NA[7]
Emergency AC Power	(Train Unreliability[1]) 4.2E-2	1.2E-2	4.5E-2	6.9E-2	1.6E-1
	(Train Unavailability) 9.7E-3	6.2E-2	5.0E-2	1.2E-1	8.0E-1
High Pressure Injection	(SI Train Unreliability[1]) 9.3E-3	2.0E-2	1.7E-2	7.5E-2	2.5E-1
	(SI Train Unavailability) 4.2E-3	4.5E-2	2.9E-2	2.1E-1	Not Reached
Power Operated Relief Valves	(System Unreliability[1]) 4.3E-2	4.7E-2	4.9E-2	1.0E-1	5.4E-1
	(Train Unavailability) N/A[3]	N/A[3]	N/A[3]	N/A[3]	N/A[3]
Residual Heat Removal (SDC)	(Train Unreliability[1]) 1.3E-2	2.8E-2	1.6E-2	4.8E-2	3.4E-1
	(Train Unavailability) 7.3E-3	3.4E-2	Not Reached[5]	Not Reached[5]	Not Reached[5]
Service Water System[4]	(Train Unreliability[1]) 3.3E-2	9.7E-2	4.3E-2	1.1E-1	3.5E-1
	(Train Unavailability) 2.7E-2	9.2E-2	Not Reached[5]	Not Reached[5]	Not Reached[5]
AOVs[2]	Component Class Unreliability	N/A	Increase 203X	Increase 630X	Not Reached
MOVs	Component Class Unreliability	N/A	Increase 1.4X	Increase 4.4X	Increase 21X
MDPs	Component Class Unreliability	N/A	Increase 1.2X	Increase 3.3X	Increase 21X

1. Total unreliability, includes T&M
2. AOV component class does not include failure to re-close of the reliefs.
3. N/A, T&M events not included in SPAR logic.
4. The primary ESW load is the RHR HTX and the diesel generators.
5. This threshold can be reached if the T&M outages associated with this system are not confined to TECH SPEC allowable combinations.
6. Aux. Feedwater MDP is a start-up pump.
7. CCW does not support RHR or DG's, flagsets on fault trees RCPSL-CCW/SWS result in system failure.
8. SG discharge MOV's from both MDP and TDP trains are included in unreliability calculations due to CCF, see basic event EFW-MOV-CF-SGS.

Table A.2.4-19 CE Plant 2/3 Mitigating Systems Threshold Summary

CE Plant 2/3 SPAR 3i (2.6E-9/hr, 1.8E-5/calendar year[1]) Ce Plant 2/3 RBPI Baseline (2.1E-9/hr, 1.4E-5/calendar year[1])					
System/Component	Baseline Train Unavailability and Unreliability	Green/White Threshold 95th %ile	Green/White Threshold (ΔCDF =1E-6/year)	White/Yellow Threshold (ΔCDF =1E-5/year)	Yellow/Red Threshold (ΔCDF =1E-4/year)
Emergency Feedwater	(MDP Train Unreliability[4]) 8.0E-3	2.0E-2	1.7E-2	8.8E-2	3.5E-1
	(TDP Train Unreliability[4]) 1.9E-1	3.5E-1	2.0E-1	3.0E-1	8.0E-1
	(MDP Train Unavailability) 1.1E-3	2.5E-3	3.4E-2	1.9E-1	4.6-1
	(TDP Train Unavailability) 4.6E-3	1.8E-2	2.8E-2	2.3E-1	Not reached[5]
Component Cooling Water	(Train Unreliability[4]) 1.6E-2	5.0E-2	1.6E-1	4.3E-1	9.8E-1
	(Standby Train Unavailability) 1.1E-2	4.4E-2	Not Reached	Not Reached	Not Reached
Emergency AC Power	(Train Unreliability[4]) 4.2E-2	1.0E-1	5.1E-2	9.2E-2	2.0E-1
	(Train Unavailability) 9.7E-3	1.9E-2	2.4E-2	1.1E-1	4.7E-1
High Pressure Injection	(Train Unreliability[4]) 9.2E-3	2.0E-2	1.8E-2	8.9E-2	3.8E-1
	(Train Unavailability) 4.2E-3	1.6E-2	1.5E-2	1.1E-1	Not reached[5]
Power Operated Relief Valves	(System Unreliability) 4.4E-2	5.0E-2	5.4E-2	1.4E-1	7.0E-1
	(Train Unavailability) N/A[3]	NA[3]	NA[3]	NA[3]	NA[3]
Salt Water System	(System Unreliability[4]) 5.1E-2	1.2E-1	8.1E-2	2.8E-1	8.6E-1
	(Standby Train Unavailability) 2.7E-2	1.0E-1	Not reached[5]	Not reached[5]	Not reached[5]
Shutdown Cooling /RHR/LPI	(Train Unreliability[4]) 1.3E-2	3.4E-2	2.8E-2	1.4E-1	Not Reached
	(Train Unavailability) 7.3E-3	2.4E-2	4.4E-1	Not reached[5]	Not Reached
AOVs[2]	Component Class Unreliability	N/A	Increase 5.0X	Increase 33X	Increase 155X
MOVs	Component Class Unreliability	N/A	Increase 2.4X	Increase 13X	Increase 60X
MDPs	Component Class Unreliability	N/A	Increase 2.7X	Increase 14X	Increase 47X

1. Calendar year is defined as 7000 critical hours.
2. AOVs does not include failure to re-close of the reliefs.
3. N/A, T&M events not included in SPAR logic.
4. Total unreliability, includes T&M
5. This threshold can be reached if the T&M outages associated with this system are not confined to TECH SPEC allowable combinations.

Table A.2.4-20 CE Plant 4 Mitigating Systems Threshold Summary

CE Plant 4 SPAR 3i (2.6E-9/hr, 1.8E-5/calendar year[1]) CE Plant 4 RBPI Baseline (2.2E-9/hr, 1.6E-5/calendar year[1])					
System/Component	Baseline Train Unavailability and Unreliability	Green/White Threshold 95th %ile	Green/White Threshold (ΔCDF =1E-6/year)	White/Yellow Threshold (ΔCDF =1E-5/year)	Yellow/Red Threshold (ΔCDF =1E-4/year)
Emergency Feedwater	(DDP Train Unreliability[2]) 8.3E-2	1.5E-1	9.8E-2	2.2E-1	Not Reached
	(MDP Train Unreliability[2]) 8.1E-3	2.0E-2	1.3E-2	4.0E-2	5.4E-1
	(TDP Train Unreliability[2]) 1.9E-1	3.5E-1	2.3E-1	5.2E-1	Not Reached
	(DDP Train Unavailability) 1.5E-2	3.6E-2	3.4E-2	1.9E-1	Not Reached
	(MDP Train Unavailability) 1.1E-3	2.5E-3	8.8E-3	8.0E-2	7.8E-1
	(TDP Train Unavailability) 4.6E-3	1.8E-2	5.1E-2	4.6E-1	Not Reached
Component Cooling Water	(Train Unreliability) 1.6E-2	5.2E-2	Not Reached	Not Reached	Not Reached
	(Train Unavailability) 1.1E-2	4.4E-2	Not Reached	Not Reached	Not Reached
Emergency AC Power	(Train Unreliability[2]) 4.1E-2	9.8E-2	4.6E-2	7.8E-2	2.3E-1
	(Train Unavailability) 9.7E-3	1.9E-2	1.7E-2	8.7E-2	7.8E-1
High Pressure Injection	(Train Unreliability[2]) 9.2E-3	2.1E-2	2.4E-2	1.4E-1	5.1E-1
	(Train Unavailability)4.2E-3	1.6E-2	2.6E-2	2.3E-1	Not Reached
Power Operated Relief Valves	(Train Unreliability) 2.2E-3	5.0E-3	3.0E-2	2.1E-1	Not Reached
	(Train Unavailability[2]) N/A[4]	N/A[4]	N/A[4]	N/A[4]	Not Reached
Raw Water System (RWS)	(Train Unreliability[2]) 3.2E-2	1.1E-1	1.6E-1	5.1E-1	9.8E-1
	(Standby Train Unavailability) 2.7E-2	8.0E-2	Not Reached	Not Reached	Not Reached
Shutdown Cooling/RHR	(Train Unreliability[2]) 6.7E-3	1.4E-2	1.5E-2	8.4E-2	5.6E-1
	(Train Unavailability) 7.3E-3	2.4E-2	9.5E-1	Not Reached[6]	Not Reached[6]
AOVs[3]	Component Class Unreliability	N/A	Increase 3.9X	Increase 25X	Increase 120X
MOVs	Component Class Unreliability	N/A	Increase 3.2X	Increase 22X	Increase 115X
MDPs	Component Class Unreliability	N/A	Increase 1.7X	Increase 8.0X	Increase 55X

1. Calendar year is defined as 7000 critical hours.
2. Total unreliability, includes T&M
3. AOVs does not include failure of the reliefs to re-close.
4. N/A – T&M events not included in SPAR logic.
5. Multiplier used in determining the associated system thresholds.
6. This threshold can be reached if the T&M outages associated with this system are not confined to TECH SPEC allowable combinations.

Table A.2.4-21 CE Plant 5 Mitigating Systems Threshold Summary

CE Plant 5 SPAR 3i (4.0E-9/hr, 2.8E-5/calendar year[1]) CE Plant 5 RBPI Baseline (2.6E-9/hr, 1.8E-5/calendar year[1])					
System/Component	Baseline Train Unavailability and Unreliability	Green/White Threshold 95th %ile	Green/White Threshold (ΔCDF =1E-6/year)	White/Yellow Threshold (ΔCDF =1E-5/year)	Yellow/Red Threshold (ΔCDF =1E-4/year)
Emergency Feedwater	(MDP Train Unreliability[4]) 1.6E-2	2.9E-2	1.7E-2	2.5E-2	8.5E-2
	(TDP Train Unreliability) 2.0E-1	3.4E-1	2.2E-1	4.1E-1	Not Reached
	(MDP Train Unavailability[4]) 1.1E-3	2.5E-3	3.2E-3	2.2E-2	2.2E-1
	(TDP Train Unavailability[4]) 4.6E-3	1.8E-2	3.2E-2	2.8E-1	Not Reached
Component Cooling Water	(Train Unreliability[4]) 4.4E-2	7.8E-2	3.2E-1	7.8E-1	1.0
	(Train Unavailability) 1.1E-2	4.4E-2	Not Reached	Not Reached	Not Reached
Emergency AC Power	(Train Unreliability[4]) 5.1E-2	1.2E-1	6.0E-2	1.1E-1	2.7E-1
	(Train Unavailability) 9.7E-3	1.9E-2	5.8E-2	4.9E-1	Not Reached[6]
High Pressure Injection	(Train Unreliability[4]) 4.3E-2	5.9E-2	8.8E-2	3.7E-1	9.1E-1
	(Train Unavailability)4.2E-3	1.6E-2	2.6E-2	2.2E-1	Not Reached[6]
Power Operated Relief Valves	(System Unreliability) 1.3E-1	1.4E-1	1.5E-1	3.7E-1	Not Reached
	(Train Unavailability[4]) N/A[3]	NA[3]	NA[3]	NA[3]	NA[3]
Service Water System	(Train Unreliability[4]) 6.0E-2	1.2E-1	8.4E-2	1.8E-1	3.7E-1
	(Standby Train Unavail.) 2.7E-2	8.5E-2	Not Reached	Not Reached	Not Reached
Shutdown Cooling / Residual Heat Removal	(Train Unreliability[4]) 2.7E-2	5.2E-2	3.5E-2	1.0E-1	Not Reached
	(Train Unavailability) 7.3E-3	2.4E-2	5.8E-1	Not Reached[6]	Not Reached
AOVs[2]	Component Class Unreliability	N/A	Increase 1.9X	Increase 9.0X	Increase 43X
MOVs	Component Class Unreliability	N/A	Increase 3.1X	Increase 20X	Increase 91X
MDPs	Component Class Unreliability	N/A	Increase 1.2X	Increase 2.4X	Increase 10X

1. Calendar year is defined as 7000 critical hours.
2. AOVS does not include failure to re-close the reliefs.
3. N/A, T&M events not included in SPAR logic.
4. Total unreliability, includes T&M
5. Multiplier used to determine the associated system threshold.
6. This threshold can be reached if the T&M outages associated with this system are not confined to TECH SPEC allowable combinations.

Table A.2.4-22 CE Plant 10/11 Mitigating Systems Threshold Summary

CE Plant 10/11 SPAR 3i (7.4E-9/hr, 5.1E-5/calendar year) CE Plant 10/11 RBPI Baseline (8.6E-9/hr, 6.0E-5/calendar year)					
System	Baseline Train Unavailability and Unreliability	Green/White Threshold 95th %ile	Green/White Threshold (ΔCDF =1E-6/year)	White/Yellow Threshold (ΔCDF =1E-5/year)	Yellow/Red Threshold (ΔCDF =1E-4/year)
Auxiliary Feedwater	(MDP Train Unreliability[1]) 1.1E-2	2.3E-2	1.1E-2	1.5E-2	4.9E-2
	(TDP Train Unreliability[1]) 1.9E-1	3.4E-1	2.0E-1	2.4E-1	6.4E-1
	(MDP Train Unavailability) 1.1E-3	2.5E-3	1.2E-2	2.1E-2	1.2E-1
	(TDP Train Unavailability) 4.6E-3	2.6E-2	2.0E-1	2.5E-1	7.5E-1
Component Cooling Water	(Train Unreliability[1]) 3.6E-2	7.2E-2	7.3E-2	2.6E-1	6.5E-1
	(Stby Train Unavailability) 1.1E-2	4.1E-2	Not Reached[4]	Not Reached[4]	Not Reached[4]
Emergency AC Power	(Train Unreliability[1]) 5.5E-2	1.3E-1	5.6E-2	6.4E-2	1.2E-1
	(Train Unavailability) 9.7E-3	6.0E-2	5.7E-2	6.9E-2	1.8E-1
High Pressure Injection	(SI Train Unreliability[1]) 9.7E-3	2.2E-2	2.0E-2	5.9E-2	1.8E-1
	(SI Train Unavailability) 4.2E-3	3.6E-2	8.5E-2	7.8E-1	Not Reached[4]
Residual/Decay Heat Removal	(Train Unreliability[1]) 1.3E-2	3.0E-2	3.1E-2	8.2E-2	Not Reached
	(Train Unavailability) 7.3E-3	7.2E-3	6.2E-1	Not Reached[4]	Not Reached[4]
Service Water[3]	(Train Unreliability[1]) 5.1E-2	1.2E-1	8.4E-2	2.8E-1	6.6E-1
	(Train Unavailability) 2.7E-2	8.8E-2	Not Reached[4]	Not Reached[4]	Not Reached[4]
AOVs[2]	Component Class Unreliability	N/A	Increase 8.0X	Increase 100X	Increase 310X
MOVs	Component Class Unreliability	N/A	Increase 3.3X	Increase 8.9X	Increase 14X
MDPs	Component Class Unreliability	N/A	Increase 1.05X	Increase 1.5X	Increase 5.0X

1. Total unreliability, includes T&M
2. AOV component class does not include failure to re-close of the reliefs.
3. The primary SWS load is the RHR HTX. The EDG's are self cooled.
4. This threshold can be reached if the T&M outages associated with this system are not confined to TECH SPEC allowable combinations.

Table A.2.4-23 CE Plant 12 Mitigating Systems Threshold Summary

CE Plant 12 SPAR 3i (4.0E-9/hr, 2.8E-5/calendar year[1]) CE Plant 12 RBPI Baseline (2.7E-9/hr, 1.9E-5/calendar year[1])					
System/Component	Baseline Train Unavailability and Unreliability	Green/White Threshold 95th %ile	Green/White Threshold (ΔCDF =1E-6/year)	White/Yellow Threshold (ΔCDF =1E-5/year)	Yellow/Red Threshold (ΔCDF =1E-4/year)
Emergency Feedwater	(MDP Train Unreliability[4]) 8.1E-3	2.0E-2	9.2E-3	1.8E-2	8.8E-2
	(TDP Train Unreliability[4]) 2.0E-1	3.5E-1	2.1E-1	3.4E-1	Not Reached
	(MDP Train Unavailability) 1.1E-3	2.5E-3	3.3E-3	2.3E-2	2.2E-1
	(TDP Train Unavailability) 4.6E-3	1.8E-2	3.0E-2	2.5E-1	Not Reached
Component Cooling Water	(Train Unreliability[4]) 6.0E-2	9.5E-2	1.4E-1	5.0E-1	1.0
	(Train Unavailability) 4.4E-2	8.8E-2	1.0E-1	6.2E-1	Not Reached[5]
Emergency AC Power	(Train Unreliability[4]) 3.8E-2	1.4E-1	4.2E-2	7.3E-2	2.2E-1
	(Train Unavailability) 9.7E-3	1.9E-2	1.8E-2	9.3E-2	8.6E-1
High Pressure Injection	(Train Unreliability[4]) 1.3E-2	2.5E-2	2.7E-2	1.1E-1	4.5E-1
	(Train Unavailability) 4.2E-3	1.6E-2	7.1E-2	6.8E-1	Not Reached[5]
Power Operated Relief Valves	(System Unreliability) 4.4E-2	4.5E-2	6.7E-2	2.5E-1	Not Reached
	(Train Unavailability) N/A[3]	NA[3]	NA[3]	NA[3]	NA[3]
Shutdown Cooling / Residual Heat Removal	(Train Unreliability[4]) 2.5E-2	4.7E-2	6.5E-2	2.3E-1	6.7E-1
	(Train Unavailability) 7.3E-3	2.4E-2	3.3E-1	Not Reached[5]	Not Reached[5]
AOVs[2]	Component Class Unreliability	N/A	Increase 7.5X	Increase 65X	Not Reached
MOVs	Component Class Unreliability	N/A	Increase 1.4X	Increase 4.5X	Increase 22X
MDPs	Component Class Unreliability	N/A	Increase 1.2X	Increase 2.9X	Increase 14X

1. Calendar year is defined as 7000 critical hours.
2. AOVs does not include failure to re-close of the reliefs.
3. N/A, T&M events not included in SPAR logic.
4. Total unreliability, includes T&M.
5. This threshold can be reached if the T&M outages associated with this system are not confined to TECH SPEC allowable combinations.

Table A.2.4-24 WE 2-Lp Plant 5/6 Mitigating Systems Threshold Summary

WE 2-Lp Plant 5/6 SPAR 3i (2.1E-9/hr, 1.4E-5/calendar year[1]) WE 2-Lp Plant 5/6 RBPI Baseline (2.1E-9/hr, 1.5E-5/calendar year[1])					
System/Component	Baseline Train Unavailability and Unreliability	Green/White Threshold 95th %ile	Green/White Threshold (ΔCDF =1E-6/year)	White/Yellow Threshold (ΔCDF =1E-5/year)	Yellow/Red Threshold (ΔCDF =1E-4/year)
Auxiliary Feedwater	(MDP Train Unreliability[5]) 8.7E-3	2.1E-2	1.2E-2	3.4E-2	5.4E-1
	(TDP Train Unreliability[5]) 1.9E-1	3.5E-1	2.5E-1	7.6E-1	Not Reached
	(MDP Train Unavailability) 1.1E-3	2.5E-3	8.0E-3	6.6E-2	6.6E-1
	(TDP Train Unavailability) 4.6E-3	1.8E-2	7.6E-2	7.0E-1	Not Reached
Component Cooling Water	(Unreliability[5]) 6.4E-2	1.2E-1	9.7E-2	2.7E-1	8.6E-1
	(Unavailability) 1.1E-2	4.4E-2	1.1E-1	9.9E-1	Not Reached
Emergency AC Power	(Unreliability[5]) 4.0E-2	9.8E-2	5.5E-2	1.3E-1	2.9E-1
	(Unavailability) 9.7E-3	1.9E-2	1.3E-1	Not Reached[4]	Not Reached[4]
High Pressure Injection	(Unreliability[5]) 9.3E-3	2.8E-2	2.6E-2	1.1E-1	4.6E-1
	(Unavailability) 4.2E-3	1.6E-2	8.8E-2	8.4E-1	Not Reached[4]
Power Operated Relief Valves	(System Unreliability) 3.3E-2	3.4E-2	5.6E-2	2.4E-1	9.9E-1
	(Unavailability) N/A[3]	N/A[3]	N/A[3]	N/A[3]	N/A[3]
Residual/Decay Heat Removal	(Unreliability[5]) 2.4E-2	4.7E-2	3.2E-2	8.7E-2	3.8E-1
	(Unavailability) 7.3E-3	2.4E-2	6.6E-2	5.8E-1	Not Reached
Service Water	(MDP Train Unreliability[5]) 3.2E-2	9.2E-2	5.2E-1	1.0	Not Reached
	(DDP Train Unreliability[5]) 7.6E-2	2.0E-1	3.0E-1	9.6E-1	Not Reached
	(MDP Train Unavailability) 2.7E-2	9.0E-2	Not Reached	Not Reached	Not Reached
	(DDP Train Unavailability) 5.5E-2	1.7E-1	Not Reached[4]	Not Reached[4]	Not Reached[4]
AOVs[2]	Component Class Unreliability	N/A	Increase 32X	Increase 185X	Not Reached
MOVs	Component Class Unreliability	N/A	Increase 1.9X	Increase 7.4X	Increase 27X
MDPs	Component Class Unreliability	N/A	Increase 1.4X	Increase 4.0X	Increase 16X

1. Calendar year is defined as 7000 critical hours. (ΔCDF is calculated in calendar years).
2. AOVs does not include failure to re-close of the reliefs.
3. N/A, T&M events not included in SPAR logic.
4. This threshold can be reached if the T&M outages associated with this system are not confined to TECH SPEC allowable combinations.
5. Total unreliability, includes T&M.

Table A.2.4-25 WE 3-LP Plant 5 Mitigating Systems Threshold Summary

WE 3-LP Plant 5 SPAR 3i (6.3E-9/hr, 4.4E-5/calendar year[6]) WE 3-LP Plant 5 RBPI Baseline (6.7E-9/hr, 4.7E-5/calendar year[6])					
System	Baseline Train Unavailability and Unreliability	Green/White Threshold 95th %ile	Green/White Threshold (ΔCDF =1E-6/year)	White/Yellow Threshold (ΔCDF =1E-5/year)	Yellow/Red Threshold (ΔCDF =1E-4/year)
Auxiliary Feedwater	(MDP Train Unreliability[1]) 9.7E-3	2.2E-2	1.1E-2	2.4E-2	1.2E-1
	(TDP Train Unreliability[1]) 2.0E-1	3.5E-1	2.0E-1	2.8E-1	Not Reached
	(MDP Train Unavailability) 1.1E-3	2.5E-3	1.2E-2	3.2E-2	2.3E-1
	(TDP Train Unavailability) 4.6E-3	6.7E-2	2.0E-1	2.8E-1	Not Reached
Component Cooling Water	(Train Unreliability[1]) 6.1E-2	1.3E-1	9.2E-2	2.6E-1	8.3E-1
	(Standby Train Unavail.[4]) 1.1E-2	8.9E-2	1.8E-1	Not Reached	Not Reached
Emergency AC Power	(Train Unreliability[1]) 4.1E-2	1.0E-1	4.3E-2	5.2E-2	1.1E-1
	(Train Unavailability) 9.7E-3	6.2E-2	4.4E-2	6.4E-2	2.6E-1
High Pressure Injection	(SI Train Unreliability[1]) 9.5E-2	2.2E-1	1.2E-1	2.8E-1	7.2E-1
	(SI Train Unavailability) 4.2E-3	3.8E-2	2.9E-1	8.8E-1	Not Reached
Power Operated Relief Valves	(System Unreliability[1]) 3.9E-2	3.9E-2	6.5E-2	2.8E-1	2.0E-1
	(Train Unavailability) N/A[3]	N/A[3]	N/A[3]	N/A[3]	N/A[3]
Residual/Decay Heat Removal	(Train Unreliability[1]) 1.9E-2	5.2E-2	2.9E-2	8.7E-2	3.1E-1
	(Train Unavailability) 7.3E-3	2.5E-2	7.1E-2	5.3E-1	Not Reached[5]
Emergency Service Water[4]	(Train Unreliability[1]) 2.5E-2	8.7E-2	2.6E-2	3.4E-2	9.0E-2
	(Train Unavailability) 2.0E-2	7.5E-2	3.0E-2	7.0E-2	4.7E-1
AOVs[2]	Component Class Unreliability	N/A	Increase 20X	Increase 62X	Increase 177X
MOVs	Component Class Unreliability	N/A	Increase 1.8X	Increase 7.5X	Increase 28X
MDPs	Component Class Unreliability	N/A	Increase 1.1X	Increase 2.2X	Increase 11X

1. Total unreliability, includes T&M
2. AOV component class does not include failure to re-close of the reliefs.
3. N/A, T&M events not included in SPAR logic.
4. The primary ESW load is the RHR HTX and the diesel generators.
5. This threshold can be reached if the T&M outages associated with this system are not confined to TECH SPEC allowable combinations.
6. Calendar year is defined as 7000 critical hours.

Table A.2.4-26 WE 3-LP Plant 10/11 Mitigating Systems Threshold Summary

WE 3-LP Plant 10/11 SPAR 3i (3.2E-9/hr, 2.3E-5/calendar year[6]) WE 3-LP Plant 10/11 RBPI Baseline (2.9E-9/hr, 2.1E-5/calendar year[6])					
System	Baseline Train Unavailability and Unreliability	Green/White Threshold 95th %ile	Green/White Threshold (ΔCDF =1E-6/year)	White/Yellow Threshold (ΔCDF =1E-5/year)	Yellow/Red Threshold (ΔCDF =1E-4/year)
Auxiliary Feedwater	(MDP Train Unreliability[1]) 8.6E-3	2.1E-2	1.0E-2	2.1E-2	1.1E-1
	(TDP Train Unreliability[1]) 1.9E-1	3.4E-1	2.1E-1	3.7E-1	Not Reached
	(MDP Train Unavailability) 1.1E-3	2.4E-3	1.1E-2	2.9E-2	2.1E-1
	(TDP Train Unavailability) 4.6E-3	2.9E-2	2.1E-1	3.8E-1	Not Reached
Component Cooling Water	(Train Unreliability[1]) 1.6E-2	4.7E-2	2.5E-1	7.4E-1	Not Reached
	(Standby Train Unavail.) 1.1E-2	4.6E-2	Not Reached[5]	Not Reached[5]	Not Reached[5]
Emergency AC Power	(Train Unreliability[1]) 4.1E-2	9.7E-2	4.4E-2	7.0E-2	1.7E-1
	(Train Unavailability) 9.7E-3	6.4E-2	4.8E-2	1.2E-1	7.8E-1
High Pressure Injection	(SI Train Unreliability[1]) 1.1E-2	TBD	8.2E-2	3.3E-1	8.1E-1
	(SI Train Unavailability) 4.2E-3	3.2E-3	1.2E-1	9.9E-1	Not Reached
Power Operated Relief Valves	(System Unreliability) 3.4E-2	6.8E-2	5.8E-2	2.5E-1	Not Reached
	(Train Unavailability) N/A[3]	N/A	N/A	N/A	N/A
Residual/Decay Heat Removal	(Train Unreliability[1]) 1.2E-2	3.5E-2	9.9E-2	3.5E-1	Not Reached
	(Train Unavailability) 7.3E-3	4.6E-2	Not Reached[5]	Not Reached[5]	Not Reached[5]
Service Water[4]	(Train Unreliability[1]) 1.0E-1	2.3E-1	Not Reached	Not Reached	Not Reached
	(Train Unavailability) 5.5E-2	1.8E-1	Not Reached	Not Reached	Not Reached
AOVs[2]	Component Class Unreliability	N/A	Increase 4.7X	Increase 100X	Not Reached
MOVs	Component Class Unreliability	N/A	Increase 2.0X	Increase 8.2X	Increase 34X
MDPs	Component Class Unreliability	N/A	Increase 1.2X	Increase 2.7X	Increase 11X

1. Total unreliability, includes T&M
2. AOV component class does not include failure to re-close of the reliefs.
3. N/A, T&M events not included in SPAR logic.
4. The primary SWS load is the RHR HTX. SWS is a backup to circ water. The EDG's are self cooled.
5. This threshold can be reached if the T&M outages associated with this system are not confined to TECH SPEC allowable combinations.
6. Calendar year is defined as 7000 critical hours.

Table A.2.4-27 WE 4-Lp Plant 1/2 Mitigating Systems Threshold Summary

WE 4-Lp Plant 1/2 SPAR 3i (1.0E-8/hr, 7.2E-5/calendar year[1]) WE 4-Lp Plant 1/2 RBPI Baseline (1.4E-8/hr, 9.8E-5/calendar year[1])					
System/Component	Baseline Train Unavailability and Unreliability	Green/White Threshold 95th %ile	Green/White Threshold (ΔCDF =1E-6/year)	White/Yellow Threshold (ΔCDF =1E-5/year)	Yellow/Red Threshold (ΔCDF =1E-4/year)
Auxiliary Feedwater	(MDP Train Unreliability[2]) 8.5E-3	2.0E-2	1.1E-2	3.5E-2	2.7E-1
	(DDP Train Unreliability[2]) 2.4E-2	6.2E-2	2.4E-2	2.8E-2	7.3E-2
	(MDP Train Unavailability[1]) 1.1E-3	2.5E-3	7.4E-3	6.3E-2	6.0E-1
	(DDP Train Unavailability) 3.0E-3	TBD[5]	3.5E-3	8.0E-3	5.4E-2
			1.5E-1	4.0E-1	8.6E-1
Component Cooling Water	(Train Unreliability[2]) 2.7E-2	3.1E-2	Not Reached[6]	Not Reached[6]	Not Reached
	(Train Unavailability[4]) 2.2E-2	TBD[5]	4.6E-2	7.3E-2	2.1E-1
Emergency AC Power (Train)	(Train Unreliability[2]) 4.3E-2	2.2E-1	1.6E-2	7.3E-2	6.5E-1
	(Train Unavailability) 9.7E-3	1.9E-2	2.6E-2	1.6E-1	Not reached
High Pressure Injection (Includes CVCS trains)	(SI Train Unreliability[2]) 9.6E-3	2.2E-2	1.0	Not reached	Not reached
	(CVCS Train Unreliability[2]) 7.7E-3	1.3E-2	3.0E-2	2.5E-1	Not reached
	(SI Train Unavailability) 4.2E-3	1.6E-2	Not reached	Not reached	Not reached
	(CVCS Train Unavail.) 2.4E-3	TBD[5]	4.0E-2	1.0E-1	6.2E-1
Power Operated Relief Valves	(System Unreliability) 3.2E-2	6.8E-2	N/A[4]	N/A[4]	N/A[4]
	(Train Unavailability) N/A[4]	N/A[4]	1.1E-2	8.4E-2	2.8E-1
Residual/Decay Heat Removal	(Train Unreliability[2]) 1.9E-2	4.9E-2	8.3E-2	4.0E-1	Not Reached[6]
	(Train Unavailability) 7.3E-3	2.4E-2	1.5E-2	4.8E-2	2.1E-1
Essential Service Water	(Train Unreliability[2]) 1.1E-2	1.6E-2	1.5E-1	Not Reached[6]	Not Reached
	(Train Unavailability) 5.9E-3	TBD[5]	Increase 100X	Increase 235X	Not Reached
AOVs[3]	Component Class Unreliability	N/A	Increase 2.7X	Increase 11X	Increase 36X
MOVs	Component Class Unreliability	N/A	Increase 1.3X	Increase 3.6X	Increase 23X
MDPs	Component Class Unreliability	N/A			

1. Calendar year is defined as 7000 critical hours.
2. Total unreliability, includes T&M.
3. AOVs component class does not include failure to re-close the reliefs.
4. N/A, T&M events not included in SPAR logic.
5. The corresponding unavailability event in the SPAR model does not include a probability distribution.
6. This threshold can be reached if the T&M outages associated with this system are not confined to TECH SPEC allowable combinations.

Table A.2.4-28 WE 4-LP Plant 10/11 Mitigating Systems Threshold Summary

WE 4-LP Plant 10/11 SPAR 3i (3.5E-9/hr, 2.5E-5/calendar year[6]) WE 4-LP Plant 10/11 RBPI Baseline (3.6E-9/hr, 2.5E-5/calendar year[6])					
System	Baseline Train Unavailability and Unreliability	Green/White Threshold 95th%ile	Green/White Threshold (ΔCDF =1E-6/year)	White/Yellow Threshold (ΔCDF =1E-5/year)	Yellow/Red Threshold (ΔCDF =1E-4/year)
Auxiliary Feedwater	(MDP Train Unreliability[1]) 8.7E-3	4.6E-4	1.3E-2	4.0E-2	1.6E-1
	(TDP Train Unreliability[1]) 1.9E-1	3.5E-1	2.3E-1	5.0E-1	Not Reached
	(MDP Train Unavailability) 1.1E-3	2.4E-3	2.1E-2	1.3E-1	Not Reached
	(TDP Train Unavailability) 4.6E-3	2.9E-2	2.3E-1	5.1E-1	Not Reached
Component Cooling Water	(Train Unreliability[1]) 1.6E-2	4.8E-2	1.2E-1	3.4E-1	7.9E-1
	(Standby Train Unavail.[4]) 1.1E-2	3.3E-2	Not Reached[5]	Not Reached[5]	Not Reached[5]
Emergency AC Power	(Train Unreliability[1]) 3.9E-2	2.0E-1	4.1E-2	5.9E-2	1.6E-1
	(Train Unavailability) 9.7E-3	6.2E-2	4.4E-2	8.6E-2	5.0E-1
Chemical and Volume Control System	(CVCS Train Unreliability[1]) 2.0E-1	3.0E-1	Not Reached	Not Reached	Not Reached
	(CVCS Train Unavailability) 2.0E-1	2.8E-1	Not Reached	Not Reached	Not Reached
High Pressure Injection	(SI Train Unreliability[1]) 1.6E-2	3.1E-2	3.7E-2	2.1E-1	8.7E-1
	(SI Train Unavailability) 4.2E-3	2.8E-2	5.3E-2	4.2E-1	Not Reached
Power Operated Relief Valves	(System Unreliability[1]) 3.8E-2	3.8E-2	6.3E-2	2.6E-1	Not Reached
	(Train Unavailability) N/A[3]	N/A[3]	N/A[3]	N/A[3]	N/A[3]
Residual/Decay Heat Removal	(Train Unreliability[1]) 2.6E-2	5.8E-2	2.5E-2	7.3E-2	3.6E-1
	(Train Unavailability) 7.3E-3	3.0E-2	5.6E-2	3.9E-1	Not Reached
Emergency Service Water[4]	(Train Unreliability[1]) 1.8E-2	4.7E-2	5.9E-2	1.9E-1	4.8E-1
	(Train Unavailability) 9.9E-3	3.6E-2	Not Reached[5]	Not Reached[5]	Not Reached[5]
AOVs[2]	Component Class Unreliability	N/A	Increase 123X	Increase 560X	Not Reached
MOVs	Component Class Unreliability	N/A	Increase 1.6X	Increase 6.6X	Increase 50X
MDPs	Component Class Unreliability	N/A	Increase 1.4X	Increase 4.7X	Increase 19X

1. Total unreliability, includes T&M
2. AOV component class does not include failure to re-close of the reliefs.
3. N/A, T&M events not included in SPAR logic.
4. The primary ESW load is the RHR HTX.
5. This threshold can be reached if the T&M outages associated with this system are not confined to TECH SPEC allowable combinations.
6. Calendar year is defined as 7000 critical hours.

Table A.2.4-29 WE 4-Lp Plant 22/23 Mitigating Systems Threshold Summary

WE 4-Lp Plant 22/23 SPAR 3i (4.7E-9/hr, 3.3E-5/calendar year[1]) WE 4-Lp Plant 22/23 RBPI Baseline (4.9E-9/hr, 3.4E-5/calendar year[1])					
System	Baseline Train Unavailability and Unreliability	Green/White Threshold 95th %ile	Green/White Threshold (ΔCDF =1E-6/year)	White/Yellow Threshold (ΔCDF =1E-5/year)	Yellow/Red Threshold (ΔCDF =1E-4/year)
Auxiliary Feedwater	(MDP Train Unreliability[5]) 8.7E-3	2.1E-2	9.8E-3	1.8E-2	5.4E-2
	(TDP Train Unreliability[5])1.9E-1	3.4E-1	2.0E-1	2.9E-1	Not Reached
	(MDP Train Unavailability) 1.1E-3	2.5E-3	3.7E-3	2.8E-2	2.5E-1
	(TDP Train Unavailability) 4.6E-3	1.8E-2	2.1E-2	1.7E-1	Not Reached
Component Cooling Water	(Train Unreliability[5]) 1.6E-2	4.7E-2	2.0E-1	6.5E-1	Not Reached
	(Standby Train Unavail.) 1.1E-2	4.4E-2	7.8E-1[4]	Not Reached	Not Reached
Emergency AC Power	(Train Unreliability[5]) 4.2E-2	1.0E-1	4.3E-2	5.5E-2	1.3E-1
	(Train Unavailability) 9.7E-3	1.9E-2	1.3E-2	3.9E-2	3.0E-1
High Pressure Injection (Includes CVCS trains)	(SI Train Unreliability[5]) 9.7E-3	2.1E-2	8.8E-1	Not Reached	Not Reached
	(SI Train Unavailability) 4.2E-3	1.6E-2	Not Reached[6]	Not Reached[6]	Not Reached[6]
	(CVCS Train Unreliability[5]) 5.9E-2	1.9E-1	4.3E-1	Not Reached	Not Reached
	(CVCS Train Unavailability) 5.4E-2	1.7E-1	Not Reached[6]	Not reached[6]	Not Reached[6]
Power Operated Relief Valves	(System Unreliability) 3.2E-2	6.8E-2	5.7E-2	2.6E-1	Not Reached
	(Train Unavailability) N/A[3]	N/A	N/A[3]	N/A[3]	N/A[3]
Residual/Decay Heat Removal	(Train Unreliability[5]) 1.7E-2	3.8E-2	3.8E-2	1.3E-1	4.7E-1
	(Train Unavailability) 7.3E-3	2.4E-2	9.3E-2	8.8E-1	Not Reached[6]
Service Water	(Train Unreliability[5]) 3.2E-2	9.4E-2	1.3E-1	2.1E-1	3.2E-1
	(Standby Train Unavail.) 2.7E-2	9.0E-2	Not Reached[6]	Not Reached[6]	Not Reached[6]
AOVs[2]	Component Class Unreliability	N/A	Increase 2.2X	Increase 13X	Increase 106X
MOVs	Component Class Unreliability	N/A	Increase 2.4X	Increase 11X	Increase 39X
MDPs	Component Class Unreliability	N/A	Increase 1.2X	Increase 3.2X	Increase 16X

1. Calendar year is defined as 7000 critical hours.
2. AOV component class does not include failure to re-close of the reliefs.
3. N/A, T&M events not included in SPAR logic.
4. Two normally running CCW trains with one train in standby.
5. Total unreliability, includes T&M.
6. This threshold can be reached if the T&M outages associated with this system are not confined to TECH SPEC allowable combinations.

Table A.2.4-30 WE 4-LP Plant 28 Mitigating Systems Threshold Summary

WE 4-LP Plant 28 SPAR 3i (5.0E-9/hr, 3.5E-5/calendar year[6]) WE 4-LP Plant 28 RBPI Baseline (3.8E-9/hr, 2.7E-5/calendar year[6])					
System	Baseline Train Unavailability and Unreliability	Green/White Threshold 95th %ile	Green/White Threshold (ΔCDF =1E-6/year)	White/Yellow Threshold (ΔCDF =1E-5/year)	Yellow/Red Threshold (ΔCDF =1E-4/year)
Auxiliary Feedwater	(MDP Train Unreliability[1]) 9.7E-3	2.2E-2	1.5E-2	4.7E-2	1.8E-1
	(TDP Train Unreliability[1]) 1.9E-1	3.5E-1	2.1E-1	3.5E-1	Not Reached
	(MDP Train Unavailability) 1.1E-3	2.5E-3	2.4E-2	1.5E-1	Not Reached
	(TDP Train Unavailability) 4.6E-3	1.6E-2	2.1E-1	3.6E-1	Not Reached
Component Cooling Water	(Train Unreliability[1]) 1.6E-2	4.8E-2	5.6E-2	3.4E-1	Not Reached
	(Standby Train Unavail.[4]) 1.1E-2	4.6E-2	1.3E-1	Not Reached[5]	Not Reached[5]
Emergency AC Power	(Train Unreliability[1]) 4.1E-2	1.0E-1	4.3E-2	5.5E-2	1.3E-1
	(Train Unavailability) 9.7E-3	6.3E-2	4.4E-2	7.0E-2	3.3E-1
Chemical and Volume Control System	(CVCS Train Unreliability[1]) 5.9E-2	1.8E-1	7.4E-1	Not Reached	Not Reached
	(CVCS Train Unavailability) 5.4E-2	1.6E-1	Not Reached	Not Reached	Not Reached
High Pressure Injection	(SI Train Unreliability[1]) 9.4E-3	2.1E-2	4.8E-1	Not Reached	Not Reached
	(SI Train Unavailability) 4.2E-3	2.6E-2	Not Reached	Not Reached	Not Reached
Power Operated Relief Valves	(System Unreliability[1]) 2.0E-2	2.1E-2	2.8E-2	1.6E-1	Not Reached
	(Train Unavailability) N/A[3]	N/A[3]	N/A[3]	N/A[3]	N/A[3]
Residual/Decay Heat Removal	(Train Unreliability[1]) 1.9E-2	5.1E-2	2.6E-2	7.6E-2	3.7E-1
	(Train Unavailability) 7.3E-3	2.8E-2	8.4E-2	6.5E-1	Not Reached[5]
Essential Raw Cooling Water[4]	(Train Unreliability[1]) 5.1E-2	1.1E-1	8.0E-2	1.6E-1	2.9E-1
	(Train Unavailability) 2.7E-2	9.4E-2	5.3E-1	Not Reached[5]	Not Reached[5]
AOVs[2]	Component Class Unreliability	N/A	Increase 4.2X	Increase 24X	Increase 105X
MOVs	Component Class Unreliability	N/A	Increase 2.3X	Increase 11X	Increase 41X
MDPs	Component Class Unreliability	N/A	Increase 1.6X	Increase 5.8X	Increase 23X

1. Total unreliability, includes T&M
2. AOV component class does not include failure to re-close of the reliefs.
3. N/A, T&M events not included in SPAR logic.
4. The primary ESW load is the RHR HTX and the diesel generators.
5. This threshold can be reached if the T&M outages associated with this system are not confined to TECH SPEC allowable combinations.
6. Calendar year is defined as 7000 critical hours.

Table A.2.5-1 Summary of Inspection Areas Impacted by New RBPIs for Mitigating System Cornerstone

RBPI	Attribute	Inspection Area
Full Power: Mitigating Systems (UR)	Equipment Performance	71111.04, Equipment Alignment 71111.12, Maintenance Rule Implementation 71111.15, Operability Evaluations 71111.22, Surveillance Testing 71152, Identification and Resolution of Problems
Mitigating Systems (UA)	Equipment Performance	71111.12, Maintenance Rule Implementation
	Human Performance (Pre-Event)	71111.14, Personnel Performance During Nonroutine Evolutions 71152, Identification and Resolution of Problems
	Configuration Control	71111.04, Equipment Alignment 71111.12, Maintenance Rule Implementation 71111.13, Maintenance Risk Assessments and Emergent Work Evaluation 71111.23, Temporary Plant Modifications 71152, Identification and Resolution of Problems

- Type A elements are those that have an effect on LERF, at least partly because they have an effect on CDF. For example, the change in CDF associated with degradation of a mitigation system that plays a role in key accident sequences (transients and SBLOCAs involving high RCS pressure) could carry over directly to a change in LERF.
- Type B elements are those that have an effect on LERF, but largely independent of CDF. An example of this is the containment isolation function, the degradation of which usually has no modeled effect on CDF.

Given a model that comprehensively addresses a Type A element's impact on CDF, its impact on LERF can be assessed using a relationship of the form:

$$\Delta\text{LERF} = (\text{Factor}) \times (\Delta\text{CDF affecting LERF sequences}),$$

where "Factor," a number less than 1, is adjusted to reflect the effects on containment failure probability of relevant accident sequence characteristics such as RCS pressure.

For some Type A mitigating systems or initiating events, it may be found that LERF is more limiting than CDF for purposes of determining the performance thresholds of RBPIs associated with these elements. This has not been assessed so far, due to the lack of integrated CDF/LERF models available to this project.

The present emphasis under the Containment Integrity portion of the Barrier Integrity cornerstone is on Type B elements. In ongoing work, Type A elements will be more comprehensively assessed, based if necessary on the approximate treatment developed for the SDP.

A.3.1 Assess the Potential Risk Impact of Degraded Performance

Unlike the analyses that address initiating events and mitigating systems, there are no functioning SPAR models for directly calculating changes to LERF resulting from element performance changes. This is a major limitation on proceeding with the RBPI development process. Nevertheless, a scoping assessment has been made, based on the Individual Plant Examination (IPE) submittals and the associated IPE Database (Ref. 5), supplemented by the NUREG-1150 (Ref. 20) assessments, the review of the IPEs in NUREG-1560 (Ref. 21), and other containment analyses.

General insights were obtained from Refs. 20 and 21. The general assessment of the LERF-significance of Type B elements is summarized for the five containment types in Table A.3.1-1.

Table A.3.1-1 Assessment of Elements of LERF-Significant Containment Barrier Attributes for PWRs with Large-Dry Containments

Plant Type	Element	LERF Significance
PWRs with Large Dry Containments	Containment Isolation	Major
PWRs with Ice Condenser Containments	Containment Isolation Ice Condenser Function Hydrogen Ignitors	Major Major Major
BWRs with Mark I Containments	Suppression pool bypass Isolation condenser (for some Mark I's) Drywell spray Operator actions (drywell spray) EOPs	Intermediate Intermediate Intermediate Intermediate Not Modeled but "Intermediate"
BWRs with Mark I Containments	Suppression pool bypass Drywell Spray EOPs	Intermediate Intermediate Not Modeled but "Intermediate"
BWRs with Mark III Containments	Containment Isolation Suppression pool bypass Hydrogen Ignitors Drywell Spray	Intermediate Intermediate Intermediate Intermediate

Although containment heat removal is not generally considered to be highly significant to LERF, it is modeled at some PWRs as playing a role in core damage prevention and in prevention of large early releases. This, too, is a Type A function, and needs to be examined within an integrated CDF/LERF perspective. Ongoing work (examining all remaining plants) will establish whether this function is risk-significant at enough plants to warrant RBPI treatment.

An examination of selected IPE results tabulated in Ref. 5 has been carried out by determining the containment-barrier-related elements affecting particular plant damage states, and using this information to try to infer how plant damage state frequency (and therefore release frequency) would change if element performance changed. Where this can be quantified, it is presented as a worst-case estimate of the change in LERF, i.e., it assumes complete degradation of element performance. In some cases, information presented in the IPE Database is not sufficiently detailed and explicit to confirm the importance of elements that were described above as important to LERF; this is indicated in the tables as an insufficiency of information presented. The IPE Database presents results, but was not intended to support requantification exercises of this kind. This approach is therefore limited. Nevertheless, some results are obtainable; they are provided in Tables A.3.1-2 to A.3.1-6.

Table A.3.1-2 Assessment of Potential Changes in LERF Due to Element Performance Changes for PWRS with Large-Dry Containments (Including Sub-Atmospheric)

Plant	Containment Failure Mode and Frequency	Element	Maximum ΔLERF†
ANO-2	Large Bypass 3.78E-7 Large Isolation 7.82E-7 Large Early 1.53E-6 Small Early 1.39E-6 Late 4.73E-6 None 2.48E-5	Containment isolation	3.01E-5
		RCS depressurization systems	0*
Braidwood 1&2	Large Bypass 0 Large Isolation 0 Large Early 0 Small Early 0 Late 2.73E-6 None 2.39E-5	Containment isolation	2.66E-5
Wolf Creek	Large Bypass 7.31E-8 Large Isolation 4.95E-8 Large Early 3.83E-8 Small Early 0 Late 1.37E-6 None 4.05E-5	Containment isolation	4.18E-5

* For cases involving LERF, RCS depressurization is either unavailable, or occurs due to a hot leg or surge line break (ref. ANO-2 IPE, Section 4.6.5, and CETs)

† Maximum ΔLERF is the change in LERF that would be caused by total failure of the element in question.

Table A.3.1-3 Assessment of Some Potential Changes in LERF Due to Element Performance Changes for PWRs with Ice Condenser Containments

Plant	Containment Failure Mode and Frequency	Element	Maximum ΔLERF
Catawba 1&2	Large Bypass 1.03E-7 Large Isolation 2.9E-8 Large Early 3.53E-8 Small Early 0 Late 2.72E-5 None 3.05E-5	Containment isolation	5.77E-5
		Ice condenser*	(Insufficient Information)
D.C. Cook 1&2	Large Bypass 7.11E-6 Large Isolation 6.26E-9 Large Early 9.25E-7 Small Early 1.58E-9 Late 1.13E-6 None 5.4E-5	Containment isolation	5.51E-5
		Ice condenser*	(Insufficient Information)

Table A.3.1-3 (Continued)

Plant	Containment Failure Mode and Frequency	Element	Maximum ΔLERF
McGuire 1&2	Large Bypass 9.48E-7 Large Isolation 1.28E-7 Large Early 0 Small Early 8.10E-7 Late 1.62E-5 None 2.17E-5	Containment Isolation	3.86E-5
		Ice condenser*	(Insufficient Information)
Sequoyah 1&2	Large Bypass 7.99E-6 Large Isolation 0 Large Early 2.73E-6 Small Early 0 Late 8.32E-5 None 7.60E-5	Containment Isolation	1.59E-4
		Ice condenser*	(Insufficient Information)
Watts Bar	Large Bypass 2.46E-5 Large Isolation 8.91E-6 Large Early 8.14E-6 Small Early 0 Late 7.1E-5 None 2.18E-4	Containment Isolation	2.8E-4
		Ice condenser*	(Insufficient Information)

* IPE insufficient to determine ΔLERF

Table A.3.1-4 Assessment of Potential Changes in LERF Due to Element Performance Changes for BWRs with Mark I Containments

Plant	Containment Failure Mode and Frequency	Element	Maximum ΔLERF
Peach Bottom 2&3	Large Bypass 6.64E-9 Large Isolation 0 Large Early 2.57E-7 Small Early 0 Late 1.40E-6 None 2.57E-6	Drywell spray/ flooding systems	0[1]
		Suppression pool bypass	0[2]
Quad Cities 1&2	Large Bypass 6E-10 Large Isolation 0 Large Early 1.38E-7 Small Early 3.52E-8 Late 6.62E-7 None 2.53E-7	Drywell spray/ flooding systems	Insufficiently Modeled
		Suppression pool bypass	Insufficiently Modeled
Vermont Yankee	Large Bypass 4.3E-8 Large Isolation 0 Large Early 1.11E-6 Small Early 0 Late 9.89E-7 None 1.16E-6	Drywell spray/ flooding systems	Insufficiently Modeled
		2. Suppression pool bypass	Insufficiently Modeled

Table A.3.1-4 (Continued)

Note: For some BWR Mark I plants, isolation condenser performance may be a key element, but is not part of containment barrier performance
[1]Coolant injection to drywell or initiation of containment flooding is important for PDSs where there is low vessel pressure
[2]For some PDSs, suppression pool bypass typically results in late releases; therefore, ΔLERF will not increase. For other PDSs, if the suppression pool is not bypassed, the release is small early; however, when the suppression pool is bypassed, the release is large early. Therefore, for these PDSs, the ΔLERF can increase by some fraction of the small early release when the suppression pool was not bypassed

Table A.3.1-5 Assessment of Potential Changes in LERF Due to Element Performance Changes for BWRs with Mark II Containments

Plant	Containment Failure Mode and Frequency	Element	Maximum ΔLERF
Nine Mile Point 2	Large Bypass 2.79E-8 Large Isolation 0 Large Early 1.58E-6 Small Early 1.08E-6 Late 2.04E-5 None 8.30E-6	Suppression pool bypass	(Insufficiently modeled)
WNP 2	Large Bypass 2.98E-8 Large Isolation 2.26E-7 Large Early 4.89E-6 Small Early 0 Late 5.30E-6 None 6.83E-6	Suppression pool bypass	(Insufficiently modeled)

Table A.3.1-6 Assessment of Potential Changes in LERF Due to Element Performance Changes for BWRs with Mark III Containments

Plant	Containment Failure Mode and Frequency	Element	Maximum ΔLERF
Grand Gulf 1	Large Bypass 0 Large Isolation 0 Large Early 5.97E-6 Small Early 1.36E-6 Late 5.66E-6 None 3.51E-6	Containment isolation	1.05E-6
		Suppression pool bypass	(Insufficiently modeled)
River Bend	Large Bypass 0 Large Isolation 4.12E-7 Large Early 0 Small Early 4.13E-6 Late 2.14E-6 None 8.98E-6	Containment isolation	1.53E-5
		Suppression pool bypass	(Insufficiently modeled)
Perry 1	Large Bypass 0 Large Isolation 3.96E-9 Large Early 2.14E-6 Small Early 9.12E-7 Late 4.76E-6 None 5.30E-6	Containment isolation	1.1E-5
		Suppression pool bypass*	(Insufficiently modeled)

*At Perry, suppression pool bypass events (other than those that involve drywell failure) have a frequency of 0.

Based on the information summarized in the above tables, it is concluded that performance degradation in the following equipment-related elements could cause significant changes in LERF:

- For Large-Dry PWRs, containment isolation
- For Ice-Condenser PWRs, in addition to containment isolation, the Ice-Condenser function and the Hydrogen Control (ignitor) function (not in table)
- For Mark-1 BWRs, suppression pool bypass, drywell spray, and isolation condenser (some Mark-1s)
- For Mark-2 BWRs, suppression pool bypass
- For Mark-3 BWRs, suppression pool bypass and containment isolation

These areas are frequently modeled, but are not reflected in models currently available to this project.

A.3.2 Obtain Performance Data for Risk-Significant, Equipment-Related Elements

Plant-specific performance data for these elements are not currently available to this project.

A.3.3 Identify Indicators Capable of Detecting Performance Changes in a Timely Manner

Although data for the above elements are not available to this project, based on Appendix F and on work presented in Sections A.1 and A.2, it is judged that essentially passive elements or elements that are only infrequently challenged are not amenable to RBPI development. Based on this, the following elements have been identified from the LERF-significant elements listed above as having the potential to be RBPIs. Each possible indicator is applicable to different containment types:

- Unreliability / unavailability of drywell spray (Mark I BWRs, Mark II BWRs, Mark III BWRs)
- Unreliability / unavailability of large containment isolation valves (PWRs, Mark III BWRs) (valves isolating paths that connect the containment atmosphere directly to the outside atmosphere)
- Unreliability / unavailability of hydrogen ignitors (Ice Condenser PWRs, Mark III BWRs)

A.3.4 Identify Performance Thresholds Consistent with a Graded Approach to Performance Evaluation from SECY 99-007

Although the RBPI development process has established the risk significance of the functions identified above in Section A.3.3, models and data available are not sufficient to establish baseline performance values and to quantify thresholds. LERF models exist for some PWR large dry containments and PWR ice condenser containments, as well as BWR Mark I containments. Therefore, BWR Mark II and Mark III containments, as well as some PWR containments, do not have LERF models developed for establishing threshold values. In addition, the existing models can only link with the older and less complete Revision 2QA SPAR models. Therefore, some accident sequences that could affect LERF cannot be propagated through the LERF models because they are not included in the Revision 2QA SPAR models.

Moreover, drywell spray is closely identified with Type A functionality (low pressure injection and suppression pool cooling). This means that RBPIs and thresholds for certain mitigating systems and certain containment-related systems need to be evaluated together within an integrated CDF/LERF perspective. Similarly, although containment heat removal is not generally an important contributor to LERF, in some PWRs it has a role in core damage prevention and in prevention of large early releases. This, too, is a Type A function, and needs to be examined within an integrated CDF/LERF perspective.

When applicable models and data are obtained, RBPI development will be completed for these potential RBPIs. In addition, RBPIs previously analyzed under the initiating events and mitigating systems cornerstones will also be re-examined to determine whether LERF considerations alter the findings of Sections A.1 and A.2.

A.3.5 Inspection Areas Covered by New RBPIs

These RBPIs are not among the performance indicators in the ROP. The inspection areas that could be impacted by these RBPIs were determined. The results are summarized in Table A.3.5-1.

Table A.3.5-1 Summary of Inspection Areas Impacted by Potential RBPIs for Containment Portion of Barrier Integrity Cornerstone

RBPI	Attribute	Inspection Area
CIV (UR&UA) and Drywell Spray (UR&UA)	Design Control	71111.02, Evaluation of Changes, Tests, or Experiments 71111.17, Permanent Plant Modifications 71111.23, Temporary Plant Modifications 71152, Identification and Resolution of Problems
	Barrier Performance	71111.12, Maintenance Rule Implementation 71111.15, Operability Evaluations 71111.20, Refueling and Outage Activities 71111.22, Surveillance Testing

A.3.6 LERF as the Figure of Merit for Containment Barrier Performance

A.3.6.1 The Definition of LERF

Regulatory Guide 1.174 defines LERF as follows:

"In this context, LERF is being used as a surrogate for the early fatality QHO. It is defined as the frequency of those accidents leading to significant, unmitigated releases from containment in a time frame prior to effective evacuation of the close-in population such that there is a potential for early health effects. Such accidents generally include unscrubbed releases associated with early [containment failure at or shortly after vessel breach, containment bypass events, and loss of containment] isolation. This definition is consistent with accident analyses used in the safety goal screening criteria discussed in the Commission's regulatory analysis guidelines. An NRC contractor's report (Ref. 22) describes a simple screening approach for calculating LERF."

Definition used in the RBPI Program:

A number of requirements and constraints peculiar to the RBPI Program contribute to a (slight) reformulation of the definition of LERF from that in Regulatory Guide 1.174. These are:

- Since quantitative determinations will be made as part of the RBPI process, it is necessary to assume a quantitative value for "large." The large release threshold is defined by volatile/semi-volatile fission product releases greater than 2.5% (i.e., the release of iodine, cesium, or tellurium greater than 2.5% is considered large). The reason for this choice is three-fold: (1) releases at or above this level have been shown to result in early fatalities (Ref. 23), thus maintaining consistency with the qualitative definition in Regulatory Guide 1.174; (2) this definition is one of three considered (Ref. 22) (the other two are "... greater than 10%" and "all releases, regardless of release magnitude); and (3) this definition allows for effective use of the IPE database to determine large-early release sequences and LERF. This large release threshold, of course, can be changed if warranted.

- The definition of "early" in the IPEs and the IPE Database is generally consistent with the definition of "early" in Regulatory Guide 1.174, that is, "... in a time frame prior to effective evacuation of the close-in population such that there is a potential for early health effects." In the absence of health effect and evacuation analysis in the IPEs, this definition has been translated into a containment failure definition, based on the occurrence of the first radiological release from the containment (containment failure) relative to time of failure of the reactor vessel. "Early Release," then, is any release before, at, or shortly after (usually a few hours) vessel failure. Although the IPEs vary in the demarcation from early to late, that is, the specific number of hours after vessel failure, they are sufficiently consistent for the purposes here. "Early" as used here is no different from "early" in the IPE Database.

A.3.6.2 The Justification for Using LERF as a Containment Barrier Metric

The issue arises as to why LERF is used alone, rather than (or in combination with) a metric that includes "late" large releases. In this report, LERF has been used based on its role in risk-informed regulation as described in Regulatory Guide 1.174. It can be argued that the E-4/yr core damage frequency (CDF) objective is more limiting than the late release frequency criterion that one would derive from the latent fatality QHO, and this argument has been used to justify a focus on LERF.

However, focusing exclusively on LERF as a metric for the containment barrier does not assign risk significance to those elements of containment barrier performance discussed in SECY 99-007 that do not affect either CDF or LERF significantly, although they affect late release frequency or other post-accident considerations such as worker dose. Moreover, if performance bands for large late release frequency were derived from the latent fatality QHO in the same way that performance bands for LERF are derived from the early fatality QHO, then performance thresholds for many of these significant elements would be implied, where currently they are not. The possibility of considering late releases in near-term RBPI development will be discussed with stakeholders.

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APPENDIX B

RBPI DETERMINATION FOR SHUTDOWN MODES

Contents

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Preface

This appendix develops one specific type of possible RBPI, which is based on total time spent in risk-significant configurations at shutdown. As seen in the following subsections, there are significant difficulties associated with this formulation, deriving partly from the nature of the risk contributions at shutdown and partly from the information needs of this particular formulation. Based on this, and as a result of internal and external stakeholder comments, it has been decided to transfer the development presented in this appendix for mitigating system RBPIs to NRR for possible use in the Significance Determination Process for shutdown-related findings.

The remainder of this preface summarizes key issues affecting the development.

RBPI Development for Shutdown vs. RBPI Development for Full Power

The following conditions make RBPI determination for shutdown significantly different from RBPI determination for full power operation.

- At shutdown, the risk is strongly dependent on the RCS condition and on the operability of mitigating systems. Risk metrics plotted as a function of time exhibit pronounced increases and decreases as RCS conditions change and accident mitigating systems are removed from service and returned to service.
- Human-induced initiating events are relatively more frequent during shutdown than during power operations.
- The risk is strongly dependent on operator response to initiating events.
- Configuration management is a more significant factor in shutdown safety than in full power safety.
- Shutdown occupies a much smaller fraction of the year than does full power operation, so shutdown-specific reliability, availability, and frequency metrics would accumulate failure data much more slowly than do comparable metrics for full power.
- Relatively few models for shutdown CDF and LERF are available compared to model availability for full power.

Model Availability

Because of lack of plant-specific shutdown PRA models, the RBPI determination process has to rely on risk insights gained from the representative models available to this project. Only two quantifiable shutdown PRA models were available to this project.

- A draft version of the Sequoyah SPAR model (which is based on the Surry LPSD PRA model) (Ref. 1)
- A generic Westinghouse 4-loop shutdown model developed for use in the Safety Monitor Version 2.0 software. (Ref. 2)

The Grand Gulf LPSD PRA model (Ref. 3) was selected as the reference model for BWR plants. The results of this PRA were used to develop thresholds for PIs. This project did not have access to a working version of the Grand Gulf PRA model.

Shutdown PRA Model Insights

Based on the results of the shutdown PRAs for Surry and Grand Gulf, the following factors dominate the risk of shutdown.

Early phase of cold shutdown at PWRs:

- High decay heat
- Overpressurization of RHR causing a rupture

Mid-loop at PWRs

- High decay heat
- RCS loops isolated (no steam generator cooling capability)
- High maintenance unavailabilities and human error probabilities (e.g., over-draining)

Cold shutdown at PWRs

- RCS loops isolated
- High maintenance unavailabilities and human error probabilities
- Failure of thimble tube seals

Startup at PWRs

- Rapid boron dilution (French reactivity scenario)

Cold shutdown at BWRs

- LOCA/diversions
- Unavailability of safety relief valves (alternative means of core cooling) when the vessel head is on
- High maintenance unavailabilities
- High human error probabilities when decay heat is high
- Failure of makeup from the suppression pool for LOCAs

Difficulties Associated with Defining a Baseline CDF

Baseline values are a special problem at shutdown compared to full power. At full power, the overall configuration is constrained by technical specifications, and with a few typical PRA assumptions (technical specifications are not violated, all legal configurations occur with probabilities determined by the products of the unavailability of individual elements, decay heat is always computed as if the reactor were at the end of a cycle, ...), baseline performance can be characterized in a straightforward manner. At shutdown, the plant configuration is much more discretionary, and determining baseline risk is therefore less straightforward than at full power. Shutdowns vary widely in risk, according to what kinds of operating states are entered, the respective dwell times, and what configurations within those states are realized. Early PRAs (e.g., Surry and Grand Gulf LPSD studies), in generating average risk values, effectively averaged over a broad range of configurational possibilities consistent with operating practices that were current at that time. In principle, these studies could be used to assess baseline performance, but operating practices have changed significantly since those studies were performed, and adopting those risk values as baselines in the current program would not serve the aim of maintaining risk at current levels.

Modern shutdown PRAs (on-line risk monitors) essentially require the input of a specific outage schedule (configurations and dwell times), so that outage-specific risk figures of merit can be obtained. Determining baseline values from such a model logically requires that either a particular outage schedule be designated as "baseline," or a set of outage schedules be taken in the aggregate to define "baseline." In this report, a representative outage schedule and a representative annual frequency of outage have been assumed (it is assumed that the baseline annual risk predicted by the reference PRA model approximates the risk of shutdown for plants belonging to the class).

Baseline Annual CDF for PWRs at Shutdown

The core damage frequency during a typical outage can vary by several orders of magnitude. The cumulative risk caused by the entry into risk-significant configurations (those with relatively high conditional CDF or conditional LERF) represents a significant portion of the total average risk. The entry into certain RCS vulnerable conditions (e.g., mid-loop operation in PWRs) is unavoidable due to the nature of the outage. Also, many equipment maintenance and testing activities are scheduled during shutdown conditions. Because the threshold values can only be developed after a realistic baseline yearly CDF is established, an attempt was made to arrive at a baseline CDF by surveying shutdown PRAs. The results for PWR plants are shown in Table B-1 and in Figure B-1. The CDF values reported for PWRs are generally between 1.0E-5 and 1.0E-4. The following clarifications are noted:

In items 1-10, the reported CDFs

- are predicted using IPE-like PRA models,
- reflect past shutdown practices (pre-NUMARC initiative, Ref. 10), and
- are underestimated in some cases because of the scope of the models.

The CDF reported in items 11-13

- is either the actual cumulative risk or target risk associated with a recent outage in a PWR plant, and
- reflects present shutdown practices (post-NUMARC initiatives).

Table B-1 Summary of PRA Results for PWRs

	PRA Study (PWR)	CDF (per calendar year)
1	NSAC-84 (1981) (Quoted in Ref. 4)	1.8E-05
2	NUREG/CR-5015 (Quoted in Ref. 4)	5.2E-05
3	Seabrook (Quoted in Ref. 4)	4.5E-05
4	Sequoyah (upper bound for LOCAs only) (Quoted in Ref. 4)	7.5E-05
5	Safety Monitor™ model for a generic Westinghouse plant (zero maintenance assumption) (Ref. 2) (Assuming two outages every 18 months and 30 days per outage)	3.1E-05

Table B-1 (Continued)

	PRA Study (PWR)	CDF (per calendar year)
6	NUREG/CR-6144 (midloop only) (Ref. 5)	5.0E-06
7	NUREG/CR-6616 (zero maintenance assumption) (Ref. 6) (Assuming two outages every 18 months and 30 days per outage)	1.2E-05
8	Sequoyah SPAR model (Ref. 1) (Assuming two outages every 18 months and 30 days per outage)	1.0E-04
9	Surry (RES study; cold shutdown only; zero maintenance) (Ref. 7)	3.2E-06
10	Surry (RES study; cold shutdown only; with maintenance) (Ref. 7)	4.4E-05
11	STP (1RE08; projected) (Ref. 8) (Assuming two outages every 18 months)	5.6E-05
12	STP (2RE06) (Ref. 8) (Assuming two outages every 18 months)	5.3E-05
13	STP (1RE07) (Ref. 8) (Assuming two outages every 18 months)	5.3E-05
14	IN 2000-13, Review of Refueling Outage Risk (Table 1, Ref. 9) (Assuming two outages every 18 months)	1.7E-4 range: [2.8E-6, 8.9E-4]

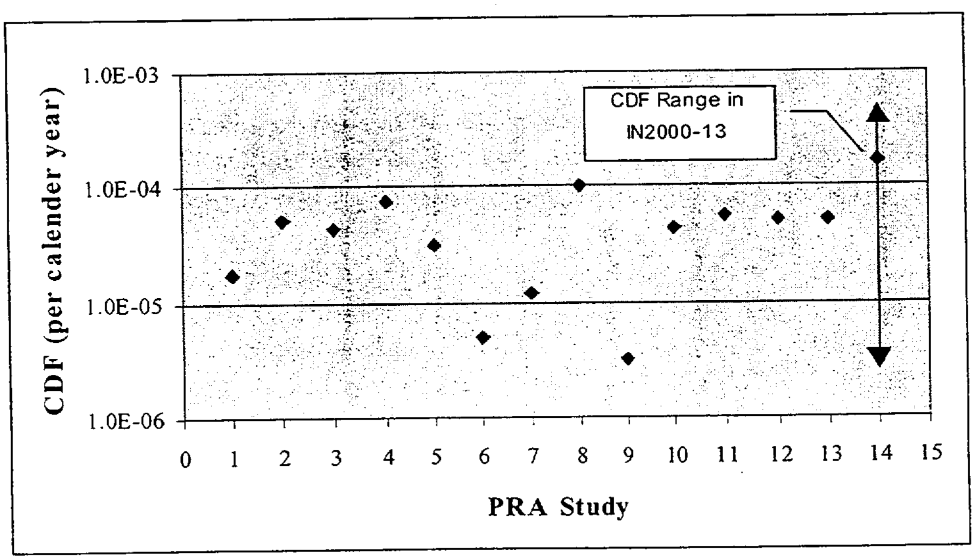

Figure B-1 Summary of PRA Results for PWRs

Item 14, derived from IN 2000-13 (Ref. 9), represents a more recent survey of outage risk experience. The following PWR shutdown risk information is provided in IN 2000-13:

> With respect to the cumulative risk data, (both predicted and actual) an extremely wide range of values were observed with respect to the outage risk. When pooled, the data

(associated with the actual risk) for the PWRs showed a cumulative mean core damage probability (CDP) of approximately 1.2E-04 for the outage. However, the values ranged from a low of 1.5E-06 to a high of 6.6E-04 with a standard deviation of 2.0E-04. (Twelve data points were used in the analysis.) These same wide ranges of values were observed with respect to the data associated with the predicted cumulative risk. The mean value for the PWR peak risk (in units of cdp/hr) was 1.6E-06/hr. As with the cumulative risk data, a wide range of values were observed with a high of 5.0E-06/hr, a low of 2.0E-08/hr and a standard deviation of 2.1E-06/hr.

Elsewhere in IN 2000-13, it is noted that some of the reported variation in the numbers is due to differences in assumptions and methods used in different evaluations. However, it is further stated that another major source of variation in the risk numbers is variation in the outages themselves. A significant factor in PWR outage risk is reduced-inventory operation. According to IN 2000-13:

> The majority of the PWR outages which were assessed employed an early "hot" midloop or reduced inventory configuration. This was almost exclusively an economic consideration in that the early midloop allowed for earlier entry into the steam generators to perform the required inspection activities. In order to eliminate the midloop, licensees would have been required to delay the steam generator entry until after the reactor vessel was defueled. This would have had the net effect of making the steam generator inspections "critical path" (i.e., the driving factor for the outage duration) in many instances thereby increasing the overall length of the outage. Even with the implementation of the early midloop, the steam generator inspection activities constituted the critical path for many of the refueling outages which were assessed. For the vast majority of the PWR outages, either the steam generator inspections or the actual refueling activities themselves constituted the critical path for the outage.

Midloop configurations contribute significantly to the total CDF, especially those occurring before 5 days after reactor shutdown. RBPI development needs to reflect this.

<u>Baseline Annual CDF for BWRs at Shutdown:</u>

Relatively little published information is available for BWR shutdown risk. The following results are provided in IN 2000-13.

> The data for the BWR plants included only three observations. Additionally, one of the BWR units experienced unexpected complications due to fuel integrity issues which significantly extended the duration of the outage. Similar to the PWR data, a wide range of values existed in the cumulative and peak risk estimates associated with the BWR outage observations. Notwithstanding these issues related to data quality, the mean actual risk was estimated to be approximately 8.6E-07 with a high and low of 1.7E-06 and 2.0E-08 respectively. The peak risk was estimated at about 1.2E-08/hr with a range of 3.3E-10/hr to 3.1E-08/hr.

Among the few published studies for BWR shutdown risk is the Grand Gulf study (Ref. 3). The annualized CDF indicated by that study is 4E-6 per calendar year. This is approximately a factor of two higher than the "high" value quoted above from IN 2000-13. This difference could be due to the difference in average CDF as a result of dwell times rather than differences in CCDF. The risk information from Ref. 3 will be used to define the BWR baseline for this development.

B.1 Initiating Events Cornerstone

B.1.1 Assess the Potential Risk Impact of Degraded Performance

Many events have the potential to challenge the shutdown cooling function. Examples of undesirable and potentially risk-significant events include:

- any unintentional, uncontrolled, undesired, and unexpected reduction of water level[1] in the reactor vessel of greater than 1 foot (a few inches during mid-loop operation in PWRs)
- flow diversions from the reactor vessel,
- inadvertent drain downs,
- uncontrolled level perturbations in the reactor vessel.

The more significant events are those that drain water down to a level close to or below the top of the core. But any undesired, uncontrolled, or unexpected draindown is an instance of poor performance. The same is true of violations of mode temperature or reactivity parameters. Reactor Mode is defined by the technical specifications in terms of temperature and reactivity bounds. Mode 6, for example, is k_{eff} less than .95 and RCS temperature less than 140^{o}F. Mode 5 is k_{eff} less than .99 and RCS temperature less than 200^{o}F. If flow diversions, reactivity changes, or other events occur causing a heat production/heat removal mismatch, exceeding one of these mode parameters may be the first indication of performance problems.

However, many such events do not qualify as IEs in PRA space because they do not actually lead to the loss of RHR. Shutdown PRAs typically do not develop these events logically below the level of the initiating event itself. RBPI development is therefore limited to consideration of the initiating events for RBPI potential. The statistics that are used to quantify these high level nodes may contain information on the causal factors that led to an initiating event, but, in general, these lower level factors are not modeled. Therefore, RBPIs cannot be based on events of this kind.

The following are modeled initiating events that lead to the actual loss of the shutdown cooling function and are therefore potentially risk-significant:

- Loss of the decay heat removal (LDH) system (loss of RHR or a critical support system)
- Loss or diversion of inventory (LDI) sufficient to cause loss of RHR
- Loss of level control (LLC) when going to mid-loop (PWRs only) sufficient to cause loss of RHR
- Loss of offsite power (LOP) causing at least momentary loss of RHR

[1]Excluding normal water level fluctuations.

Based on the representative studies cited above, the risk significance of these events has been assessed as described below. The risk significance of these events with respect to the CDF metric is determined by their frequencies and their conditional core damage probabilities (CCDPs). For the above initiating events, the CCDPs were assessed as follows. Results presented below establish that all of these initiating events are risk-significant in at least some configurations.

Assessment of Initiating Event CCDPs

- A 35-day refueling outage each 18 months of operation was assumed. It was further assumed that analyzing the time during which the decay heat is removed by the RHR system (during mode 4, 5, and 6) could capture the more risk-significant portions of a refueling outage. This corresponds to approximately 85% of the assumed outage time (29-30 days).
- Non-refueling outages consist of both scheduled outages and unscheduled outages. These outages share one characteristic - they vary widely from a few hours in hot standby to many days of cold shutdown. The latter may or may not include extended periods with the containment and the RCS open, and may sometimes include extended mid-loop operation in PWRs. For purposes of the PI analysis, it was judged that the risk of non-refueling outage operation could be estimated by assuming that the refueling outage results could be applied to non-refueling outages. An additional 35 days every 18 months is assumed for non-refueling outages.
- The assumed refueling outage and maintenance outage times of 35 days every 18 months, yields a power operation fraction of 87%.
- The shutdown SPAR model for Sequoyah (the reference model for PWR plants) uses the concept of POS groups/time windows to account for the variability in RCS conditions and decay heat level. The approximate correspondence between POS groups and operating modes of a typical PWR are as follows:

  - Pressurized cooldown -- Mode 4: hot shutdown (cooldown with RHR to 200°F); Mode 5: cold shutdown (cooldown to ambient temperature); Mode 4: hot shutdown (RCS heat-up)
  - Depressurized RHR cooling with normal inventory -- Mode 5: cold shutdown (reactor inventory is at normal level and RCS is depressurized); Mode 6: refueling (draining RCS to midloop before and after refueling)
  - Depressurized RHR cooling with reduced inventory -- Mode 5: (mid-loop operation and reduced inventory)
  - Depressurized RHR cooling with refueling cavity filled -- Mode 6: (refueling)

- The Grand Gulf shutdown PRA model (the reference model for BWR plants) also uses the concept of POS groups/time windows to account for the variability in RCS conditions and decay heat level. This model is however limited to the analysis of cold shutdown only.
- The differences in decay heat level are accommodated by introducing the time windows shown below in Table B.1-1.

Table B.1-1 Time Window Definitions

	Time Window 1	Time Window 2	Time Window 3	Time Window 4
Time After Shutdown (TAS) in hours	<75	75<TAS<240	240<TAS<768	>768

- In Phase 2 of the Grand Gulf study, the annual CDF associated POS 5 (consisting mainly of cold shutdown operating condition) is estimated to be 2.1E-6 per reactor year. Based on the Phase 1 study, approximately 60% of the CDF occurs in POS 5. To account for the risk of the unanalyzed portion of the outage, the CDF of POS 5 was extrapolated linearly. This provided an estimate of a total baseline aggregate CDF of 3.5E-6 (2.1E-6/0.6). To obtain average conditional core damage probabilities (CCDPs), the hourly rate of each class of initiating events was converted to a calendar base rate (using the outage schedule defined above). The results are shown in Table B.1-2.

Table B.1-2 Calculation of Weighted CCDPs for BWR Shutdown Initiators

IE	POS 5 CDF (Based on Grand Gulf Study)	Approximate Aggregate Yearly CDF (adjusted to account for unanalyzed POSs)	IE Frequency (per hour)	IE Frequency (per year)	Average Baseline CCDP
LDH	9.9E-08	1.65E-07	6.16E-05	5.72E-02	2.88E-06
LDI	1.3E-06	2.17E-06	8.74E-06	8.11E-03	2.67E-04
LOP	7.0E-07	1.17E-06	1.50E-05	1.39E-02	8.38E-05
Total	2.10E-06	3.50E-06	8.54E-05	7.92E-02	4.41E-05

- The PWR SPAR model provided the estimates of the initiating event frequencies on a per hour basis (see Table B.1-3), and the conditional core damage probability (CCDP) for various combinations of IEs and time windows (see Table B.1-4). The data in Table B.1-4 along with the assumed refueling outage schedule are used to generate a weighted baseline CCDP for each initiator (Table B.1-5 below). The third and fourth columns of Table B.1-5 provide the timing of entry into a POS group in terms of days after shutdown (DAS) and the residence time (RT) in the POS group.

Table B.1-3 Generic Initiating Event Frequency Estimates for PWRs

	Frequency	
IE	**Per Reactor Hour**	**Per Calendar Year[2]**
LDH	8.38E-05	7.78E-02
LDI	7.20E-05	6.68E-02
LOP	1.63E-05	1.51E-02
Total (Time- Related Initiating Events)	1.72E-04	1.60E-01

[2]The frequency value accounts for the average amount of time that a plant is in the shutdown condition during a typical calendar year.

Table B.1-3 (Continued)

	Frequency	
	Per Demand	**Per Calendar Year** (assuming 2 drain downs per year)
LLC	1.81E-02	2.41E-02

Table B.1-4 Estimates of CCDPs for Various POS Groups and Time Windows (SPAR Generated data; Applicable to PWRs)

General Mode Characteristic				Time Window	Baseline CCDP			
Mode	POS Group	Reactor Inventory	RCS Boundary		LDH	LDI	LLC	LOP
Mode 4/5	Pressurized RHR cooldown	Normal	Intact	window 1	1.24E-03	1.63E-03	NA	5.17E-03
				window 2	1.04E-03	1.52E-03		5.00E-03
				window 3	1.01E-03	1.13E-03		4.92E-03
				window 4	1.04E-04	2.20E-04		2.78E-04
Mode 5	Depressurized RHR cooling with normal inventory	Normal	Intact or vented	window 1	5.21E-04	7.55E-04	NA	2.43E-03
				window 2	3.34E-04	6.52E-04		1.24E-03
				window 3	4.62E-05	2.10E-04		1.12E-03
				window 4	4.90E-05	1.92E-04		4.31E-04
Mode 5	Depressurized RHR cooling with reduced inventory	Reduced	Intact or vented	window 1	9.92E-04	7.64E-04	7.64E-04	2.12E-03
				window 2	9.69E-04	6.63E-04	6.63E-04	1.26E-03
				window 3	3.32E-05	1.26E-04	1.26E-04	6.12E-04
				window 4	2.23E-05	1.08E-04	1.08E-04	1.99E-04
Mode 6	Refueling cavity filled	Gravity full	Vessel head off	window 1	cannot realistically reach this state			
				window 2				
				window 3	time to core uncovery > 48 hours	5.60E-04[3]	NA	time to core uncovery > 48 hours
				window 4		5.60E-04		

[3]The reference SPAR model does not handle this POS group. The value for the CCDP is obtained from a generic Westinghouse 4-loop shutdown model developed for use in the Safety Monitor Version 2.0 software.

Table B.1-5 Calculation of Weighted CCDP for PWR Shutdown Initiators [Weighted by Residence Time (RT) in Each POS Group] – Applicable to PWR Plants

Mode	POS Group	DAS	RT Day	Fraction of Time in State	LDH		LDI		LLC	LOP	
					CCDP	Duration Weighted CCDP	CCDP	Duration Weighted CCDP	CCDP	CCDP	Duration Weighted CCDP
Mode 2/3	Low power cooldown with SGs	0-1	Not analyzed								
Mode 4 hot/cold shutdown	Pressurized RHR cooldown	1-2	1	0.03	1.24E-03	4.28E-05	1.63E-03	5.62E-05		5.17E-03	1.78E-04
Mode 5 cold shutdown	Pressurized RHR cooldown	2-3	1	0.03	1.24E-03	4.28E-05	1.63E-03	5.62E-05		5.17E-03	1.78E-04
Mode 5 cold shutdown	Depressurized RHR cooling with normal inventory	3-5	2	0.07	3.34E-04	2.30E-05	6.52E-04	4.50E-05		1.24E-03	8.55E-05
Mode 5 cold shutdown	Depressurized RHR cooling with reduced inventory	5-7	2	0.07	9.69E-04	6.68E-05	6.63E-04	4.57E-05	6.63E-04	1.26E-03	8.69E-05
Mode 6 refueling	Depressurized RHR cooling with normal inventory	7-9	2	0.07	3.34E-04	2.30E-05	6.52E-04	4.50E-05		1.24E-03	8.55E-05
Mode 6 refueling	Refueling cavity filled	9-19	10	0.34	0.00E+00	0.00E+00	5.60E-04	1.93E-04		0.00E+00	0.00E+00
Mode 6 refueling	Depressurized RHR cooling with normal inventory	19-20	1	0.03	4.62E-05	1.59E-06	2.10E-04	7.24E-06		1.12E-03	3.86E-05
Mode 5 cold shutdown	Depressurized RHR cooling with reduced inventory	20-22	2	0.07	3.32E-05	2.29E-06	1.26E-04	8.69E-06	1.26E-04	6.12E-04	4.22E-05
Mode 5 cold shutdown	Depressurized RHR cooling with normal inventory	22-27	5	0.17	4.62E-05	7.97E-06	2.10E-04	3.62E-05		1.12E-03	1.93E-04
Mode 4 hot shutdown	RCS heat-up (similar to pressurized RHR cooldown)	27-30	3	0.10	1.01E-03	1.04E-04	1.13E-03	1.17E-04		4.92E-03	5.09E-04
Mode 3/2	RCS heat-up	30-35	Not analyzed								
Σ			29	1.00		3.15E-04		6.10E-04	7.89E-04		1.40E-03

B.1.2 Obtain Performance Data for Risk-Significant, Equipment-Related Elements

Previous work has led to the values of initiating event frequency tabulated above. Data for these initiating events are forthcoming because their associated reporting requirements are governed by the LER rule.

B.1.3 Identify Indicators Capable of Detecting Performance Changes in a Timely Manner

The initiating event frequencies tabulated above are too low to indicate plant-specific performance changes in a timely manner. Therefore, there are no shutdown-related RBPIs for the initiating events cornerstone.

However, these events occur at an observable rate in the operating fleet. Therefore, these initiating events are referred for industry trending.

B.1.4 Identify Performance Thresholds Consistent with a Graded Approach Outlined in SECY 99-007

No RBPIs were identified, so no performance thresholds were determined.

B.2 Mitigating Systems Cornerstone

B.2.1 Assess the Potential Risk Impact of Degraded Performance

Some equipment that is important at shutdown is also used at full power and is covered by RBPIs developed to cover full power operation. In principle, performance thresholds for these items should be determined based on change in total CDF resulting from performance degradation, and not just the change in full-power, internal-events CDF resulting from performance degradation.

The following discussion focuses on licensee management of plant configuration during shutdown. Most licensees manage shutdown risk in accordance with Generic Letter 88-17 and the NUMARC-91-06 (Ref. 10) directives. These directives are designed to give the licensee guidance in maintaining adequate defense in depth during shutdown operations for controlling risk. From a risk point of view, defense in depth is maintained if, through configuration control, the licensee maintains an adequate mitigating capability consistent with the risk significance of the POS. Because technical specifications are relaxed at shutdown, there is a potential for entering into vulnerable RCS conditions (e.g., mid-loop in PWRs) without adequate mitigating capability. The shutdown PRA models surveyed have identified the unavailability of equipment due to maintenance as the dominant cause of loss of mitigation capability. If the duration and frequency of risk-significant configurations (configurations in which CCDF is relatively high; defined more explicitly below) are not controlled, the accumulated risk (core damage probability) can be significant.

Maintenance unavailabilities of mitigating systems and human performance responding to the initiating event are especially risk-significant elements that are modeled. Accident sequences include contributions from conjunctions of train unavailabilities. These conjunctions of unavailabilities are the elements of risk-significant configurations. There are many risk-significant configurations that are not covered by TS in Modes 5 and 6. It is possible for a plant to be in a risk-significant configuration for significant portions of an outage.

Equipment performance is also important, but as noted above, most of the equipment involved in the mitigation of accidents during shutdown falls within the scope of the RBPIs developed for full power operation. Moreover, the reliability characteristics of the mitigating systems are not likely to change significantly during shutdown, because the duration of a shutdown is typically much shorter than the duration of full power operations. Much of the equipment used only at shutdown is not modeled in typical PRAs. Shutdown-specific system-level RBPIs therefore have limited potential.

B.2.2 Obtain Performance Data for Risk-Significant, Equipment-Related Elements

Performance data for the configuration element consists of a statement of plant configuration (availability of mitigating system trains) as a function of time. For a given shutdown, an outage plan is a statement of the licensee's intent. The actual configurational data will reflect not only equipment trains being taken out for maintenance deliberately, but also trains being unavailable due to failure, error, or unplanned over-running of allotted maintenance time. Calculations presented below were based on outage schedules considered representative.

Routine characterization of actual plant configuration would require information collection beyond current reporting requirements.

B.2.3 Identify Indicators Capable of Detecting Performance Changes in a Timely Manner

A key element of configuration control than can be monitored is the accumulated time spent in risk-significant configurations during the observation period. Performance indicators are formulated below based on this metric. These performance indicators directly measure the time the plant spent in risk-significant configurations (combinations of unavailabilities and plant conditions with respect to decay heat and RCS inventory).

As a result of internal and external stakeholder comments, it has been decided to transfer this development to NRR for possible use in the SDP for shutdown-related findings. This development is presented here for purposes of illustration.

Characterization of Risk-Significant Configurations

In order to quantify PWR CDF conditional on plant configuration (i.e., CCDF), a generic Westinghouse safety monitor model was quantified under different configurations that have the potential to occur during a typical refueling outage. The results of the evaluation of the risk impact of the different preventive maintenance schedules (NUREG/CR-6166, Ref. 11) contain the estimates of CCDFs for various shutdown configurations at BWRs. The results are as shown in Table B.2-1 for PWRs and Table B.2-2 for BWRs. The "zero-maintenance CDF" values shown in each table represent the core damage frequency per day assuming all mitigating systems are available. The following observations are made:

PWRs:

- The most vulnerable RCS condition is when RCS water level is low and secondary cooling is unavailable within the first two weeks following shutdown. The daily CCDF for this configuration (with zero maintenance assumption) is on the order of 1E-4 per day up to 4½ days after shutdown, and 1E-5 per day 5 or more days after shutdown. Several days of residence in this state can incur significant core damage probability.

BWRs:

- The baseline CDF is highest in POS 5 when the decay heat is still high and the vessel head is on.
- The highest daily CCDF calculated was about 5E-5/day. This is about 2E-6/hr. This corresponds to two conditions:
    - When the suppression pool is drained in POS 4 or 5. The suppression pool provides the suction source for ECCS pumps and acts as a heat sink for the removal of decay heat from the core. This condition should definitely be captured as a risk-significant configuration.
    - When all safety relief valves are removed from service in POS 4 or 5. The SRVs are required for water solid closed loop core cooling following the loss of shutdown cooling.

Definition of the RBPI

The RBPIs formulated below reflect excess time spent in risk-significant configurations during the observation period. Four categories of configurations are defined in terms of conditional core damage frequency (CCDF) and, in the case of "Early Reduced-Inventory," operational conditions. The baseline for each category (the typical time spent in configurations associated with that category) has been determined by examination of representative outage profiles, as discussed in the Preface to this appendix. Spending time over and above the baseline duration in configurations having relatively high CCDF results in core damage probability above the baseline value.

The configuration category definitions are as follows:

Negligible	CCDF << 1E-6/day
Low	CCDF ~ 1E-6/day
Medium	CCDF ~ 1E-5/day
Early Reduced-Inventory	CCDF ~ 1E-4/day
High	CCDF ~ 1E-4/day

Based on these definitions, realizable configurations can be assigned to these categories based on the CCDF and operational conditions associated with the configuration. This is done in Table B.2-3 and Table B.2-4 for PWRs and BWRs respectively.

The BWR results (Tables B.2-2 and B.2-4) are extracted from published results, and details of the assumptions underlying those results are not available. For the PWR case, the results presented were performed using a generic Westinghouse 4-loop shutdown model acquired by the USNRC from SCIENTECH, Inc. This model was developed for use in Safety Monitor Version 2.0 software. The assumptions used in calculating CCDF for PWR configurations are presented below.

Detailed Assumptions Underlying Calculation of CCDF for PWR Configurations

Pressurized Cooldown (Mode 4)

- This mode is hot shutdown.
- The RCS temperature is below 275° F, and the RCS is pressurized.
- There is a bubble in the pressurizer. The Safety Monitor model assumes that the reactor is normally cooled by SG heat removal in this POS, with SG's supplied by AFW or condensate. Although RHR shutdown cooling is possible in this POS, the model does not have a Loss of RHR initiating event for this POS.
- All SGs are supplied with secondary makeup and removing decay heat.
- RHR shutdown cooling is available if SG heat removal fails, but may not be the preferred option for the operators. If the accident goes too long without restoration of DHR, the primary will heat up and pressurize beyond the point at which RHR shutdown cooling can be established.
- Both RHR loops are operable.
- Both DG's are operable.
- Both PORV's are operable with block valves open.
- 3 AFW pumps are operable, with one MD pump operating.
- All SI signals are disabled.
- The SI pump breakers are racked out.
- One charging pump is providing charging flow. The other two charging pumps are racked out, but available.
- All operator errors are set to nominal probabilities.

Depressurized RHR Cooling with Normal Inventory (Mode 5)

- The RCS temperature is less than 200° F, and the RCS is at atmospheric pressure.
- The RCS is not vented.
- There is a bubble in the pressurizer.
- RHR shutdown cooling is operating.

Table B.2-1 PWR Shutdown Configuration Conditional CDF (Based on a Generic Westinghouse 4-Loop Shutdown PRA Model)

Representative Configurations Occurring in a Typical Outage																			
POS				Configuration CDF															
				No Maintenance	Backup RHR Train Unavailable	Electrical Power Trains Unavailable				Support Cooling Trains Unavailable		Secondary Cooling Trains Unavailable	Emergency Injection Trains Unavailable				Other Trains Unavailable		
Group	Mode	RCS Boundary	Days After Shutdown	Unavailability	RHR	One EDG	Two EDG	One Safety-Related AC Bus	One Safety-Related DC Bus	One Train of ESW	One Train of CCW	(All SGs)	RWST	Two SI	Both Sumps	Two PORV	All SG and PORV	All SG and RWST	All SG and Both Sumps
Pressurized Cooldown	Mode 4 Hot shutdown	Intact	4	7.7E-08	1.5E-07	8.5E-07	2.4E-05	1.2E-06	3.0E-07	7.4E-07	2.7E-07		7.7E-06	7.9E-08	1.3E-06	3.0E-07			
Depressurized RHR Cooldown with Normal Inventory	Mode 5 Cold shutdown	Intact	8	1.9E-08	2.3E-08	4.1E-08	6.8E-07	7.1E-07	4.3E-08	3.8E-07	3.8E-07	1.2E-06	8.8E-07	1.9E-08	1.4E-06	1.6E-06	1.1E-04	1.1E-04	1.1E-04
Depressurized RHR Cooling with Reduced Inventory	Mode 5 Cold shutdown	Intact or isolatable	12	3.8E-07	6.8E-07	4.1E-07	1.2E-06	1.9E-06	4.1E-07	1.3E-06	1.3E-06	1.2E-05	1.7E-06	3.8E-07	9.2E-06	1.9E-06	1.1E-04	1.1E-04	1.1E-04
Depressurized RHR Cooling with Reduced Inventory	Mode 5 Cold shutdown	vented	13	1.0E-05	1.5E-05	1.1E-05	1.7E-05	5.8E-05	1.0E-05	1.6E-05	1.6E-05		1.1E-04	1.0E-05	1.7E-05				
Refueling Cavity Filled	Mode 6	vented	14	3.3E-08	2.2E-07	3.8E-08	2.5E-07	2.7E-07	3.8E-08	2.2E-07	2.2E-07		3.8E-08	3.3E-08	1.2E-05				
Depressurized RHR Cooling with Reduced Inventory	Mode 5 Cold shutdown	vented	24	3.7E-06	5.3E-6	4.0E-06	9.6E-06	1.1E-06	4.0E-06	4.0E-06	5.6E-06		1.1E-4	3.7E-06	9.8E-06				
Low Inventory Configurations Occurring Early in a Typical Outage																			
Depressurized RHR Cooling with Reduced Inventory	Mode 5 Cold shutdown	Intact or isolatable	2	3.0E-06	5.1E-06	3.0E-06	3.6E-06	3.6E-06	3.5E-06	3.0E-06	5.0E-06	1.1E-04	3.0E-06	3.0E-06	2.8E-05	3.0E-06	1.1E-04	1.1E-04	1.3E-04
Depressurized RHR Cooling with Reduced Inventory	Mode 5 Cold shutdown	vented	2.5	1.1E-04	1.6E-04	1.1E-04	1.2E-04	1.6E-04	1.1E-04	1.1E-04	1.6E-04		1.1E-04	1.1E-04	1.3E-04				
Depressurized RHR Cooling with Reduced Inventory	Mode 5 Cold shutdown	vented	7	2.4E-05	3.5E-05	2.4E-05	3.3E-05	6.9E-05	2.2E-05	2.4E-05	3.5E-05		1.1E-04	2.4E-05	3.6E-05				

Table B.2-2 BWR Shutdown Configuration Conditional CDF (Based on NUREG/CR-6166 Results)

Description	POS 4 Hot Shutdown	POS 5 Cold Shutdown (vessel head is on)	POS 6 Refueling (with vessel head off and level raised to steam lines)	POS 7 Refueling (with vessel head off and upper pool filled)
Zero Maintenance, Baseline	1.3E-07	3.4E-07	~1E-8	~1E-8
Emergency Diesel Generator III (dedicated to HPCS)	2.6E-07	4.8E-07	~1E-8	~1E-8
Condensate System (CDS)	1.1E-07	3.4E-07	~1E-8	~1E-8
Control Rod Drive Train B	1.2E-07	3.5E-07	~1E-8	~1E-8
Emergency Diesel Generator (EDG) I	4.6E-07	8.5E-07	~1E-8	~1E-8
Emergency Diesel Generator II	4.6E-07	8.5E-07	~1E-8	~1E-8
Standby Service Water Train C (dedicated support system to HPCS)	9.6E-07	1.3E-06	~1E-7	~1E-8
Suppression Pool (empty)	2.3E-05	5.5E-05	5.8E-06	1.3E-06
Residual Heat Removal System Train (RHR) A	1.2E-07	3.5E-07	~1E-7	~1E-8
Residual Heat Removal System Train C	1.2E-07	3.5E-07	~1E-8	~1E-8
Standby Service Water (SSW) Train A	4.9E-07	1.1E-06	1.2E-07	~1E-7
All Safety Relief Valves (SRV)	4.5E-05	5.1E-05	N/A	N/A
Division I Battery	4.6E-07	8.5E-07	~1E-8	~1E-8
Division II Battery	4.6E-07	8.5E-07	~1E-8	~1E-8
Division III Battery	2.6E-07	4.8E-07	~1E-8	~1E-8
High Pressure Core Spray (HPCS)	9.3E-07	1.2E-06	~1E-7	~1E-8
Low Pressure Core Spray (LPCS)	1.1E-07	3.7E-07	~1E-8	~1E-8
SSW Train A and HPCS	6.3E-06	9.7E-06	1.2E-07	1.1E-06
SSW Train A and CDS	4.9E-07	1.1E-06	1.3E-07	~1E-7
Firewater System (all three pump trains)	1.1E-07	3.6E-07	2.9E-07	~1E-7
Firewater Diesel-Driven Pumps	1.1E-07	3.4E-07	~1E-8	~1E-8
EDGs I and II	6.0E-06	9.1E-06	~1E-7	~1E-8
EDGs I and III	1.9E-06	2.1E-06	~1E-7	~1E-8
RHR System Train A and all SRVs	6.8E-05	7.4E-05	N/A	N/A
Divisions I and II Batteries	6.8E-05	6.9E-05	~1E-7	~1E-8
Shutdown Cooling Train A and the Suppression Pool	2.4E-05	5.8E-05	6.4E-06	1.3E-06
LPCS and HPCS	1.2E-06	1.6E-06	~1E-8	1.2E-07
LPCS and RHR Train A	1.7E-07	7.3E-07	~1E-7	~1E-7
SSW Train A and SSW Train C	6.3E-06	9.7E-06	1.4E-07	1.1E-06
RHR Train A and RHR Train C	3.6E-07	3.8E-07	~1E-7	~1E-8

Table B.2-3 PWR Shutdown Configurations Risk Classification (Based on a Generic Westinghouse 4-Loop Shutdown PRA Model)

POS				Configuration Risk Classification															
				No Maintenance Unavailability	Backup RHR Train Unavailable	Electrical Power Trains Unavailable				Support Cooling Trains Unavailable		Secondary Cooling Trains Unavailable	Emergency Injection Trains Unavailable				Other Trains Unavailable		
Group	Mode	RCS Boundary	Days After Shutdown		RHR	One EDG	Two EDG	One Safety-Related AC Bus	One Safety-Related DC Bus	One Train of ESW	One Train of CCW	(All SGs)	RWST	Two SI[a]	Both Sumps	Two PORV	All SG and PORV	All SG and RWST	All SG and Both Sumps
Low Inventory Configurations Occurring Very Early (within the first 5 days) in an Outage																			
Depressurized RHR Cooling with Reduced Inventory	Mode 5 Cold shutdown	Intact or isolatable	2	Low	Med	Low	Low	Low	Low	Low	Med	High	Low	Low	Med	Low	High	High	High
Depressurized RHR Cooling with Reduced Inventory	Mode 5 Cold shutdown	vented	< 5	ERI-V[b]	ERI-V[b]								ERI-V[b]						
Representative Configurations Occurring in a Typical Outage																			
Pressurized Cooldown	Mode 4 Hot shutdown	Intact	4			Low	Med	Low		Low			Med		Low				
Depressurized RHR Cooldown with Normal Inventory	Mode 5 Cold shutdown	Intact	8				Low	Low				Low	Low		Low	Low	High	High	High
Depressurized RHR Cooling with Reduced Inventory	Mode 5 Cold shutdown	Intact or isolatable	12		Low		Low	Low		Low	Low	Med	Low		Med	Low	High	High	High
Depressurized RHR Cooling with Reduced Inventory	Mode 5 Cold shutdown	vented	7	Med	Med	Med	Med	High	Med	Med	Med		High	Med	Med				
Depressurized RHR Cooling with Reduced Inventory	Mode 5 Cold shutdown	vented	13	Med	Med	Med	Med	High	Med	Med	Med		High	Med	Med				
Refueling Cavity Filled	Mode 6	vented	14												Med				
Low Inventory Configurations Occurring Late in a Typical Outage																			
Depressurized RHR Cooling with Reduced Inventory	Mode 5 Cold shutdown	Vented	24	Low	Med	Low	Med	Low	Low	Low	Med		High	Low	Med				

Notes:

a. In this configuration it is assumed that a makeup pump is available.
b. This configuration category assumes that measures are taken to compensate for the risk associated with early reduced-inventory operations. If compensatory measures are not taken, these configurations are assigned to the "High" configuration category.

- Shaded cells correspond to combinations of POS and configuration that are not analyzed, either because the configuration violates the POS definition, or because the systems involved play no role in the POS. They include:

  – Mode 4 configurations related to complete unavailability of the secondary cooling systems. This is because it is assumed that in Mode 4 (hot shutdown) the heat removal function is performed by the SGs.
  – Mode 5 configurations related to complete/partial unavailability of the secondary cooling systems when the RCS is vented. This is because under vented RCS condition, secondary cooling is not possible.
  – Mode 6 configurations related to complete/partial unavailability of the secondary cooling systems. In this mode, secondary cooling is not possible.
  – Mode 5/6 configurations related to unavailability of PORVs when the RCS is vented. PORV operability is inconsequential when the RCS is vented.

- Blank cells represent configurations whose CCDF < 1.0E-6 per day. The low CCDF for specific cells is explained below.

  – Cell <one RHR train is OOS in Mode 4 >: the decay heat removal function is performed by AFW. Cooling by the operable RHR train and feed and bleed are credited.
  – Cell <one RHR train is OOS in Mode 5 <u>and</u> RCS is intact or isolatable>: SG heat removal is credited as recovery after RCS heats up. Feed and bleed is also credited.
  – Cell <one EDG is OOS in Mode 5 <u>and</u> RCS is intact or isolatable>: SG heat removal is credited as recovery after RCS heats up.
  – Cell <one ESW/CCW train is OOS in Mode 5 <u>and</u> RCS is intact or isolatable>: The equipment OOS affects RHR. SG heat removal, which is unaffected by the CCW/ESW outage, is credited as recovery after RCS heats up.
  – Cell <specified equipment OOS in Mode 6 <u>and</u> refueling cavity is flooded>: Continual boiling with water addition to vessel is credited.
  – Cell <2 sumps OOS in Mode 6 <u>and</u> refueling cavity is flooded>: This is a risk-significant configuration (medium ranking) because the long term inventory control function is lost following a LOCA.
  – Cell <two SI trains are OOS in Mode 4/5 <u>and</u> RCS is intact or isolatable>: Credit is taken for the isolation of the leak, initial injection by the make-up pumps, and secondary cooling via SGs. When the RCS is vented, secondary cooling cannot be established.
  – Cell <PORVs are OOS in Mode 5 <u>and</u> RCS is intact or isolatable and RCS inventory low>: a bleed path cannot be established to support cooling by a "feed and spill" method. The worth of the "feed and spill" success path is greater under reduced inventory conditions than under normal-inventory conditions.

<u>**Key:**</u>

Low	Low Risk Configuration	AFW	Auxiliary feed water	RHR	Residual heat removal
Med	Medium Risk Configuration	CCW	Component cooling water	SG	Steam generator
High	High Risk Configuration	DC	Direct Current power division	RWST	Refueling water storage tank
ERI-V	Early Reduced-Inventory (vented)	EDG	Emergency diesel generator	SI	Safety injection
AC	Alternating Current power division	ESW	Emergency service water	PORV	Power-operated relief valve

Table B.2-4 BWR Shutdown Configurations Risk Classification (Based on NUREG/CR-6166 Results)

POS			Configuration Risk Classification																
			No Maintenance Unavailability	Emergency AC/DC Trains Unavailable					Support Cooling Trains Unavailable			Emergency Cooling Trains Unavailable				Other Trains Unavailable			
Group	Mode	RCS Boundary		EDG I or II	EDG I and II	EDG I and III	One BAT division	Two BAT divisions	SSW A	SSW C	SSW A and C	HPCS	LPCS and HPCS	SP empty	SRVs all	SSW A and HPCS	SSW A and CDS	RHR A and all SRVs	SDC A and SP
POS 4	Hot shutdown	Intact		Low	Med	Low		High		Low	Med	Low	Low	Med	Med	Med		High	Med
POS 5	Cold shutdown	Vessel head on		Low	Med	Low	Low	High	Low	Low	Med	Low	Low	High	High	Med	Low	High	High
POS 6	Refueling	Vessel head off (level raised to steam line)												Med					Med
POS 7	Refueling	Upper pool filled									Low			Low		Low			Low

Note: Blank cells represent configurations whose CCDF < 1.0E-6 per day.

Key:

Low	Low Risk Configuration	HPCS	High pressure core spray
Med	Medium Risk Configuration	LPCS	Low pressure core spray
High	High Risk Configuration	SP	Suppression pool
EDG	Emergency diesel generator	SRV	Safety relief valve
BAT	Battery	CDS	Condensate system
SSW	Standby service water	SDC	Shutdown cooling

- If RHR fails, SG heat removal, using AFW / condensate, is available. The SG secondary sides contain normal inventory.
- Both RHR loops are operable.
- Both DG's are operable.
- Both PORV's are operable with block valves open.
- Two motor driven AFW pumps are operable.
- All SI signals are disabled.
- The SI pump breakers and 1 charging pump breaker are racked out.
- One charging pump breaker is racked in, but the charging pump is in standby.
- SI and charging are "available" with operator action if required.
- No RCP cooling is required.
- All operator errors are set to nominal probabilities.
- Two trains of AC are operable.
- The RCS is at atmospheric pressure.
- Pipe break LOCA frequencies are reduced from those that pertain to power operation.
- SG tube rupture and steam line break are not postulated.
- Inventory diversion from the RCS (in containment) is postulated.
- Interfacing LOCA (due to human error) is postulated.

Depressurized RHR cooling with Reduced inventory (Non-vented RCS in Mode 5)

- The RCS temperature is less than 200° F, and the RCS is at atmospheric pressure.
- The RCS is not vented.
- The pressurizer is drained. The water level is at midloop of the cold leg. RHR shutdown cooling is operating.
- If RHR fails, SG heat removal is available through reflux cooling
- Both RHR loops are operable.
- Both DGs are operable.
- Both PORVs are operable with block valves open.
- All ESF signals are disabled.
- The SI pump breakers and one charging pump breaker are racked out.
- One Charging pump breaker is racked in, but the charging pump is in standby.
- SI and charging are "available" with operator action if required.
- No RCP cooling is required.
- Two trains of AC are operable.
- Pipe break LOCA frequencies are reduced from those that pertain to power operation.
- Inventory diversion from the RCS (in containment) is postulated.
- Interfacing LOCA (due to human error) is postulated.

Depressurized RHR cooling with Reduced inventory (Vented RCS in Mode 5)

- The RCS temperature is less than 200° F, and the RCS is at atmospheric pressure.
- The RCS is vented.
- The pressurizer is drained. The water level in the RCS is at midloop.
- RHR shutdown cooling is operating.
- If RHR fails, SG heat removal is not available because the RCS is vented.
- Both RHR loops are operable.
- Both DG's are operable.
- The PORV status is inconsequential because the RCS is vented.
- Gravity feed from the RWST is available.
- All SI signals are disabled.
- SI pump and charging pump breakers are racked out.
- SI and charging pumps are "available" with operator action if required.
- All operator errors are set to nominal probabilities.
- Two trains of AC are operable.
- The RCS pressure is atmospheric.
- Pipe break LOCA frequencies are reduced from those that pertain to power operation.
- Inventory diversion from the RCS (in containment) is postulated.
- Interfacing LOCA (due to human error) is postulated.

Refueling (Mode 6)

- The RCS temperature is less than 140° F, and the RCS is at atmospheric pressure.
- The head is off.
- The refueling cavity is full.
- RHR shutdown cooling is operating.
- One RHR loop is operable and operating.
- Both DG's are operable.
- Gravity feed from the RWST is available.
- All SI signals are disabled.
- SI pump and charging pump breakers are racked out.
- SI and charging pumps are "available" with operator action if required.
- All operator errors are set to nominal
- 2 trains of AC are operable.
- Loss of RHR cooling in this state can not lead to core damage within 24 hrs. Time to boiling after loss of RHR is about 15 hours. Time to core damage is greater than 48 hrs.
- The RCS pressure is atmospheric.
- Pipe break LOCA frequencies are reduced from those that pertain to power operation.
- Inventory diversion from the RCS (in containment) is postulated.
- Interfacing LOCA (due to human error) is postulated.

Interpretation of the Risk Significance of Shutdown Configurations

The POS group in which the accident is postulated to occur determines what systems can be credited for mitigation. The potential success paths are determined by the operability of mitigating systems at the time of an accident, and by whether the challenge to the RHR function is caused by a LOCA or a non-LOCA condition. The following success paths are potentially available:

If the RHR function is lost as a result of a LOCA or a flow diversion:

1. Leak termination prior to loss of RHR cooling
2. (Makeup to RCS) *AND* (Spill if needed) *AND* (Long term re-circulation)

If the RHR function is lost as a result of a non-LOCA condition:

1. RHR restoration – either by repair of the lost train or alignment of the standby train
2. Secondary cooling
3. (Forced feed to RCS) *AND* (Spill) *AND* (Long term re-circulation)
4. Gravity feed of the RWST through the RCS if conditions allow.

The key characteristics of the POS group are the following:

Water Level

The water level in the vessel is one of the key attributes of a POS definition. In a Westinghouse PWR, the water level can range from mid-loop to 23' above the vessel flange. In mid-loop, the time to boil after a loss of RHR cooling can be as short as 10 to 30 minutes. Time to core uncovery can be as short as 2 hours. In this configuration, the loss of RHR is a significant safety challenge. During refueling, when the refueling cavity is flooded, the time to boil can be 15 to 30 hours. The time to core uncovery after a loss of RHR is 2 to 3 days. In this configuration, the loss of RHR is a less significant safety challenge.

RCS Pressure Boundary

The status of the RCS pressure boundary affects the methods available for decay heat removal. During Modes 4, 5, and 6 the RCS can be intact (with operable relief valves), vented, or have the head removed. Heat removal through the steam generators and reflux cooling is only available when the RCS is isolatable or intact. RHR shutdown cooling is available in all modes. Gravity feed of the RWST (through the RHR lines) is only available under certain conditions when a large vent exists. Feed and bleed is available when the RCS is intact or when sufficient vent area exists. Avoidance of Low Temperature Overpressure (LTOP) is required when the RCS boundary is intact and the

RCS temperature is less than 275° F. Charging pumps and SI pumps are usually racked out in Mode 4 and 5 if the RCS is not vented. This complicates operator response to lowering water level in response to a LOCA, and operator initiation of feed and bleed cooling in response to a loss of RHR.

Decay Heat Level

Decay heat level is important to accident sequence modeling during shutdown, because it determines the time available for mitigation, prior to inventory boil-off. This time affects the probability of successful operator action. The decay heat varies as a function of time from shutdown, and it depends on whether the reactor contains old fuel waiting to be off-loaded, or new fuel waiting for start-up. Over a complete refueling operation, decay heat levels vary by a factor of 6 from 2 days after shutdown to 30 days after shutdown with new fuel. Decay heat levels determine the success criteria, and the time for operator action. Thus the time at which an accident occurs impacts the effectiveness of mitigating functions.

Based on the above, the CCDF associated with reduced inventory operations soon after shutdown is potentially high (> 1E-4 per day). These configurations are nevertheless entered, but typically with compensatory measures in place that serve to reduce the CCDF. This is explained in IN 2000-13:

> With respect to the time of entry into the midloop configurations, data were collected relative to the scheduled as well as the actual time after shutdown before midloop conditions were achieved. Additionally, information associated with the estimated time-to-boil while at midloop was collected. As shown in Table 1 [of the IN], the average scheduled time after shutdown before entering midloop was about 84 hours with the actual value being closer to 93 hours. (The most aggressive schedule planned a midloop configuration 68 hours after shutdown.) The average estimated time-to-boil for the reduced inventory/midloop configurations was about 15 minutes (assuming a loss of shutdown cooling or inventory control) with a high and low estimate of 24 minutes and 9 minutes respectively.
>
> ...
>
> Of the PWR outages employing a midloop or reduced inventory configuration, 9 of the 15 outages did so with a concurrent unavailability of either an emergency diesel generator or the performance of significant switchyard maintenance. At least one outage employed a midloop configuration with concurrent switchyard and emergency diesel maintenance. However, each of the outages prescribed a number of contingencies and other strict controls during midloop activities. These controls generally followed the NUMARC guidance with respect to protecting trains of equipment, comprehensive pre-evolution briefings, establishment of diverse means of level indications, and in some cases, the addition of temporary emergency power supplies.

The calculations presented in Table B.2-3 are based on a model that reflects the impact of decay heat, reduced inventory, and most aspects of equipment configuration, but not the compensatory measures described above.

B.2.4 Identify Performance Thresholds Consistent with a Graded Approach Outlined in SECY 99-007

The thresholds for time spent in risk-significant configurations can be developed once the baseline risk values are established. The baseline values need to reflect typical times spent in risk-significant configurations. As stated in the Preface to this appendix, the baseline at shutdown is a strong function of the outage plan, and assignment of a baseline for purposes of this indicator requires the characterization of a characteristic shutdown risk profile.

Some insight into the expected incidence of configurations other than negligible or low can be obtained from IN-00013, which examined recent outage experience at selected plants. Where licensee risk models were available to analyze the outages, this IN reports the peak risk per hour in the outages examined. Based on the reported peak risk per hour, of the three BWR outages examined, two entered no non-negligible configurations, and the other entered only "Negligible" or "Low." Of the 16 PWR outages examined, 12 were evaluated using quantitative risk models. Of these 12, 8 entered configurations that would be considered either "Medium," "High," or "ERI-V" within the present classification scheme. According to the IN, most of these PWR outages employed an early midloop or reduced-inventory configuration (ERI-V), and many did so with a concurrent unavailability of either an emergency diesel generator or the performance of significant switchyard maintenance. However, each of the outages prescribed a number of contingencies and other strict controls during midloop activities, including such things as comprehensive pre-evolution briefings, establishment of diverse means of level indications, and addition of temporary emergency power supplies.

In addition, a BWR outage schedule was and it was found that no risk-significant configuration was entered during that outage.

Accordingly, based on available risk insights, the following assumptions are made:

<u>PWRs:</u>

- A baseline of 20 days is assigned to "Low" risk configurations. This accounts for a total contribution from this category on the order of 2E-5.
- A baseline of 2 days is assigned to "Medium" risk configurations. This corresponds to a contribution from this category of approximately 2E-5. An important sub-set of this category is mid-loop operations that take place early in the shutdown, but 5 or more days after reactor shutdown occurs.
- A baseline of 1 day is assigned to "ERI-V" configurations. These are reduced-inventory configurations with the RCS vented, taking place less than 5 days after reactor shutdown occurs when decay heat is still relatively high. This baseline corresponds to a contribution

from this category that could be as high as 1E-4, if compensatory measures are not in place.

- A baseline of 0 is assigned to "High" risk configurations. A PWR plant will not deliberately enter into any "High" risk configurations, although it may enter ERI-V configurations if compensatory measures are in place.

BWRs:

- A BWR plant does not enter into any high risk category configurations (daily CCDF 1E-4).
- On average, 50% of the annual CDF of 4E-6 is incurred while in medium risk category configurations (CCDF of 1E-5) that typically last less than 6 hours.
- The remaining CDF (2E-6) is incurred while operating in low risk category configurations (daily CCDF of 1E-6). This corresponds to 2 days of stay in low risk category configurations.

Using the assumptions listed above, the threshold values for time spent in each risk category configuration are calculated. The results are shown in Table B.2-5 and Table B.2-6.

The thresholds calculated for "ERI-V" configurations are quantified as if the associated CCDF were on the order of 1E-4 per day. These thresholds may be somewhat conservative if the compensatory measures taken upon entry into ERI-V are highly effective. However, no quantitative model available to this project takes credit for those compensatory measures. This is discussed further in Sections 3.2 and 6.6. The possible conservatism in the thresholds has been offset to some extent by the choice of 1 day as a baseline for ERI-V configurations.

Table B.2-5 Baseline and Thresholds for Time in Risk-Significant Configurations Indicators - PWRs

Configuration Category	Baseline	G/W Threshold	W/Y Threshold	Y/R Threshold
Low	20 days	21 days	30 days	120 days
Medium	2 days	2 days + .08 day (2 hrs)	3 days	12 days
Early Reduced-Inventory (vented)[a]	1 day	1 day	1.08 days (1 day + 2 hrs)	2 days
High	0	0	.08 day (2 hrs)	1 day

a. This configuration category assumes that measures are taken to compensate for the risk associated with early reduced-inventory operations. If compensatory measures are not taken, these configurations are assigned to the "High" configuration category.

Table B.2-6 Baseline and Thresholds for Time in Risk-Significant Configurations Indicators - BWRs

Configuration Category	Baseline	G/W Threshold	W/Y Threshold	Y/R Threshold
Low	2 days	3 days	12 days	102 days
Medium	0.20 day (5 hrs)	0.29 day (7 hrs)	1 day	10 days
High	0	0	.08 day (2 hrs)	1 day

B.2.5 Inspection Areas Covered by New RBPIs

The potential RBPIs developed above for shutdown are not currently in the ROP. The inspection areas that could be impacted by the new initiating event RBPIs were determined. The results are summarized below in Table B.2.5-1.

Table B.2.5-1 Summary of Inspection Areas Impacted by Potential Shutdown RBPIs for Mitigating Systems Cornerstone

RBPI	Attribute	Inspection Area
Time in High/Medium/Low Risk-Significant Configurations	Configuration Control	71111.04, Equipment Alignment 71111.13, Maintenance Risk Assessments and Emergent Work Evaluation 71111.20, Refueling and Outage Activities 71111.23, Temporary Plant Modifications

B.3 Barrier Integrity Cornerstone

No quantifiable models of LERF at shutdown were available to this project to support application of the full flowchart process presented in Section 2 of the main report. The following discussion is based on risk insights summarized below.

Containment performance at shutdown is affected by one issue that does not enter into consideration of full-power RBPIs, namely, that containment may be open during shutdown, and needs to be reclosed expeditiously under certain conditions. The situation for specific plant types is as follows:

PWRs:

Analysis performed in NUREG-1449 shows that timely closure of PWR containment prevents large early release in core damage scenarios initiated at shutdown.

BWRs with Mark-I and Mark-II Containments:

Analysis performed in NUREG-1449 shows that BWR secondary containment alone is not expected to prevent large early release in core damage scenarios. This means that a change in BWR Mark-I and -II shutdown CDF equates to a change in LERF if primary containment is open. This circumstance is offset by generally lower shutdown CDFs for BWRs.

BWRs with Mark-III Containments:

Analysis performed in NUREG/CR-6143 shows that timely closure of these BWR containments prevents large early release in core damage scenarios initiated at shutdown.

This suggests possible containment RBPIs analogous to the possible time-in-risk-significant-configurations RBPIs defined above in Section B.2.2. These would be defined for the risk-significant configuration categories introduced for the RBPIs defined for mitigating systems as follows.

Potential RBPI for PWRs and Mark-III BWRs:

Time spent in risk-significant configurations with containment not closed and preparations for timely closure not complete (timely: before boiling, if RCS is vented)

Potential RBPI for Mark-I and Mark-II BWRs:

Time spent in risk-significant configurations with primary containment not closed and not capable of timely closure.

An increase in time spent in a particular configuration with containment not capable of timely closure implies an increase in LERF equal to the CDF associated with that configuration. Configurations with negligible conditional CDF are therefore associated with negligible changes in LERF (except for changes in CDF that exceed 1.0E-7, which would not be considered negligible changes in LERF). However, risk-significant configurations contribute directly and significantly to LERF if containment is open and timely closure is not provided for. Configurations in which only a short time is available to respond to initiating events are also generally those in which only a short time is available to effect containment closure.

Data and models are not presently available to quantify these indicators. Therefore, neither baselines nor thresholds can be quantified. Quantification of these indicators would require the following:

- the time spent in risk-significant configurations defined in Section B.2.3,
- the time spent with containment in the indicated state during those risk-significant configurations, and
- extension of the treatment in Section B.2.3 to assessment of configurations in which the CDF change exceeds 1.0E-7.

B.4 References

1. Thatcher, T. A., et al., "Accident Sequence Precursor Extension to Low Power, Shutdown, and External Events Feasibility Study (Sequoyah Model)," Idaho National Engineering and Environmental Laboratory, Lockheed Idaho Technologies Company, January 1996.

2. A Generic Westinghouse 4-Loop Shutdown Model Developed for Use in the Safety Monitor Version 2.0 Software, acquired by USNRC from SCIENTECH, Inc.

3. "Evaluation of Potential Severe Accidents During Low Power and Shutdown Operations at Grand Gulf, Unit 1," NUREG/CR-6143, SAND93-2440, Vol. 1, U.S. NRC, July 1995.

4. "Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States," NUREG-1449, U.S. NRC, 1993.

5. "Evaluation of Potential Severe Accidents during Low Power and Shutdown Operations at Surry Unit-1," NUREG/CR-6144, U.S. NRC, October 1994.

6. "Risk Comparison of Scheduling Preventive Maintenance During Shutdown vs. During Power Operation for PWRs," NUREG/CR-6616, BNL-NUREG-52549, U.S. NRC, December 1998.

7. T. L. Chu, Private Communication, based on work done for NUREG/CR-6144 (Ref. 5).

8. "Perspective on Shutdown Issues at STP," Presented to the "Use of Low Power and Shutdown Risk in Regulatory Activities Public Workshop," April 27, 1998.

9. "Review of Refueling Outage Risk," IN 2000-13, U.S. NRC, September 27, 2000.

10. "Guidelines for Industry Actions to Assess Shutdown Management," NUMARC 91-06, NUMARC, December 1991.

11. "Risk Impact of BWR Technical Specifications Requirements During Shutdown," NUREG/CR-6166, SAND93-3998, Sandia National Laboratories, U.S. NRC, October 1994.

APPENDIX C

RBPI DETERMINATION FOR EXTERNAL EVENTS ACCIDENT RISK

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Tables

Appendix C: RBPI Determination for External Events Accident Risk

This appendix provides preliminary RBPI results for fire. Other external events, such as seismic and flood, are not included in the scope of Phase 1 RBPI development.

The results from the Individual Plant Examinations for External Events (IPEEE's) were used to assess the risk-significant performance attributes in accordance with the RBPI development process shown in Figure 2.1. In addition, the Fire Protection Risk Significance Screening Methodology, used in the current fire significance determination process (Ref. 1), was reviewed to provide additional insights to the use of IPEEE information. The IPEEE results are not collated in as comprehensive a way as was done for the IPE program, although draft NUREG-1742 (Ref. 2) does provide a comprehensive summary of the perspectives gleaned from the technical reviews of the IPEEE submittals. These studies indicate that fire CDF varies significantly among plants. However, fire CDF is generally high enough that some elements of fire scenarios are risk-significant compared to risks associated with full power internal events or shutdown risk. Specifically, NUREG-1742 states "... the CDFs from accidents initiated by fires are of the same order of magnitude as those from other random internal events for the industry taken as a whole."

The following IPEEE reports were reviewed (Refs. 3 through 17):

Browns Ferry 2	Fort Calhoun	Prairie Island
Clinton	H.B. Robinson 2	Quad Cities 1&2
Davis-Besse	Millstone 2	Sequoyah 1&2
Dresden 2&3	Monticello	Waterford 3
Duane Arnold	North Anna 1&2	Washington Nuclear 2

Table C-1 below shows a comparison of fire CDF to internal events CDF for the above plants.

Table C-1 Significance of Fire CDF Relative to Internal Events CDF

Plant	Fire CDF	Internal Events CDF	Fire/Internal Events Ratio
Browns Ferry 2	6.73E-06	4.80E-05	14%
Clinton	3.26E-06	2.66E-05	12%
Davis-Besse	1.72E-05	6.60E-05	26%
Dresden 2	2.04E-04	1.85E-05	1103%
Dresden 3	2.53E-04	1.85E-05	1368%
Duane Arnold	1.05E-05	7.84E-06	128%
Fort Calhoun	2.78E-05	1.36E-05	204%
H.B. Robinson 2	2.23E-04	3.20E-04	70%
Millstone 2	6.30E-06	3.42E-05	18%
Monticello	8.37E-06	2.60E-05	32%
North Anna 1&2	3.99E-06	7.16E-05	6%

Table C-1 (Continued)

Plant	Fire CDF	Internal Events CDF	Fire/Internal Events Ratio
Prairie Island	6.32E-05	5.00E-05	126%
Quad Cities 1	6.60E-05	1.20E-06	5500%
Quad Cities 2	7.13E-05	1.20E-06	5942%
Sequoyah 1&2	1.56E-06	1.70E-04	9%
Waterford 3	7.04E-06	1.70E-05	41%
Washington Nuclear 2	1.76E-05	1.75E-05	100%

NUREG-1742 states that the IPEEE results appear to confirm the general perception that fire risk is more a function of spatial phenomena than it is a function of plant systems design. That is, there were no clear patterns relating to fire-induced CDF that could be attributed to differences in plant system design features. Therefore, grouping of plants as was done for internal events and shutdown is not feasible.

C.1 Initiating Events Cornerstone

For the purposes of this analysis, a fire initiating event is defined as the occurrence of a potentially significant fire, regardless of its duration or significance, and regardless of whether a given event actually causes a plant trip. (By definition, a potentially significant fire has the potential to cause a plant trip, if not suppressed.) Detection and suppression are addressed as part of the mitigating systems cornerstone.

C.1.1 Assess the Potential Risk Impact of Degraded Performance

"Elements" correspond to items that appear in accident sequence descriptions. Under the initiating events cornerstone, the only elements appearing in typical models are the initiating events themselves. Fire accident sequences are defined by fire areas. In fact, then, "fire" is not the initiating event definition: rather, fire *in a specific area* is the initiating event of a fire CDF sequence. Because different areas are associated with different degrees of vulnerability to fire, associating thresholds with generic fires would be a poor approximation.

The risk-significant fire areas vary from plant to plant. However, the following fire areas are the most common among the list of risk-significant fire areas based on the accident sequences identified in the IPEEE for each plant:

- Switchgear Room
- Control Room
- Cable Spreading Room
- Auxiliary Building (PWR)/Reactor Building (BWR)
- Turbine Building
- Battery Room
- Cable Vault/Tunnel/Chase Zones
- Diesel Generator Rooms

The complete list of risk-significant fire areas was created for each IPEEE reviewed and is provided in the tables below. A fire area was considered risk-significant if the contribution to the total fire CDF was two percent or greater.

Table C.1.1-1 Significant Fire Areas for Browns Ferry

Fire Area	CDF	Percent of Total
Unit 2 Reactor Building, 621' and North Side of 639'	1.07E-06	15.9%
Turbine Building	7.30E-07	10.8%
Unit 2 Battery and Battery Board Room	5.53E-07	8.2%
4kV Shutdown Board Room B	4.97E-07	7.4%
Control Bay - 593' Elev	4.73E-07	7.0%
Intake Pump Station	4.72E-07	7.0%
4kV Shutdown Board Room C and 250V Battery Room	4.51E-07	6.7%
Cable Spreading Room	4.48E-07	6.7%
4kV Shutdown Board Room D	4.15E-07	6.2%
4kV Bus Tie Board Room	3.08E-07	4.6%
Unit 1 and 2 Diesel Generator Building	2.84E-07	4.2%
Unit 2 Reactor Building, South 593' Elev. And RHR Hx Rooms	2.78E-07	4.1%
4kV Shutdown Board Room A and 250V Battery Room	2.54E-07	3.8%
Total	6.73E-06	

Table C.1.1-2 Significant Fire Areas for Clinton

Fire Area	CDF	Percent of Total
Div 1, Div 2, & Div 3 Switchgear Rooms	1.45E-06	44.5%
Main Control Room	1.20E-06	36.8%
Screenhouse, General Access and Pipe Tunnel Areas	3.39E-07	10.4%
Total	3.26E-06	

Table C.1.1-3 Significant Fire Areas for Davis-Besse

Fire Area	CDF	Percent of Total
No. 1 Low Voltage Switchgear Rooms	5.90E-06	34.4%
High Voltage Switchgear Room B	5.18E-06	30.2%
Control Room	4.31E-06	25.1%
High Voltage Switchgear Room A	1.38E-06	8.0%
Total	1.72E-05	

Table C.1.1-4 Significant Fire Areas for Dresden 2

Fire Area	CDF	Percent of Total
Units 2 & 3 Control Room Backup HVAC	6.16E-05	30.2%
Units 2 & 3 SBGT & TBCCW Hx	5.87E-05	28.8%
Unit 2 Reactor Building Open Area 545 Elev.	2.34E-05	11.5%
Unit 2 North Trackway/Switchgear Area	1.57E-05	7.7%
Units 2 & 3 Turbine Corridor	1.32E-05	6.5%
Unit 2 Battery Room	1.04E-05	5.1%
Unit 2 Reactor Building Switchgear Area	9.11E-06	4.5%
Unit 2 Reactor Building Elev. 545	8.76E-06	4.3%
Total	2.04E-04	

Table C.1.1-5 Significant Fire Areas for Dresden 3

Fire Area	CDF	Percent of Total
Units 2 & 3 SBGT & TBCCW Hx	5.89E-05	23.3%
Unit 3 West Corridor and Trackway	5.27E-05	20.8%
Unit 3 Second Floor Reactor Building	5.06E-05	20.0%
Units 2 & 3 Turbine Corridor	2.15E-05	8.5%
Unit 3 Reactor Building Switchgear Area	1.78E-05	7.0%
Units 2 & 3 Cable Tunnel	1.38E-05	5.5%
Units 2 & 3 Aux. Electric Equipment Room	1.12E-05	4.4%
Unit 3 Reactor Building Ground Floor	7.39E-06	2.9%
Units 2 & 3 Mezzanine Floor	7.27E-06	2.9%
Units 2 & 3 Control Room Backup HVAC	5.54E-06	2.2%
Total	2.53E-04	

Table C.1.1-6 Significant Fire Areas for Duane Arnold

Fire Area	CDF	Percent of Total
Division I Switchgear Room	5.61E-06	53.3%
Division II Switchgear Room	4.92E-06	46.7%
Total	1.05E-05	

Table C.1.1-7 Significant Fire Areas for Fort Calhoun

Fire Area	CDF	Percent of Total
Control Room	7.90E-06	28.4%
Compressor Area	6.01E-06	21.6%
Turbine Building	3.97E-06	14.3%
Upper Electrical Penetration	3.26E-06	11.7%
Basement Level General Area	2.05E-06	7.4%
East Switchgear Area	7.84E-07	2.8%

Table C.1.1-7 (Continued)

Fire Area	CDF	Percent of Total
Transformer Yard Area	6.18E-07	2.2%
Intake Structure	5.96E-07	2.1%
Group 1 MCC Area	5.66E-07	2.0%
Total	2.78E-05	

Table C.1.1-8 Significant Fire Areas for H.B. Robinson 2

Fire Area	CDF	Percent of Total
Battery Room	7.76E-05	34.7%
Control Room	4.47E-05	20.0%
Transformer Yard	3.70E-05	16.6%
Electric Switchgear/Electrical Equipment Room	2.38E-05	10.7%
Unit 2 Cable Spreading Room	1.50E-05	6.7%
Aux. Bldg Hallway	1.24E-05	5.5%
SW Pump Area	4.38E-06	2.0%
Total	2.23E-04	

Table C.1.1-9 Significant Fire Areas for Millstone 2

Fire Area	CDF	Percent of Total
Auxiliary Building - Area A-1G	1.69E-06	26.8%
Turbine Building	1.63E-06	25.9%
Intake Structure - Area I-1A	9.66E-07	15.3%
Control Room - Main Control Board/ESAS Cabinets	6.57E-07	10.4%
Auxiliary Building - Area A-12A	5.50E-07	8.7%
Auxiliary Building - Area A-1B	5.21E-07	8.3%
Cable Vault - Area A-24	2.83E-07	4.5%
Total	6.30E-06	

Table C.1.1-10 Significant Fire Areas for Monticello

Fire Area	CDF	Percent of Total
Admin Building (Cable Spreading Room)	1.45E-06	17.3%
Admin Building (Control Room)	1.45E-06	17.3%
Turbine Building (MCC 142/143 TB Fire Area XII)	1.27E-06	15.2%
Turbine Building (MCC 133/Feedwater Pump Area)	1.20E-06	14.3%
Reactor Building (West Side)	5.56E-07	6.6%
Turbine Building (Lower 4KV Area)	5.03E-07	6.0%
Emergency Filtration Building (Div. II)	4.05E-07	4.8%
Admin Building (Battery Rooms 7A & 7B)	3.21E-07	3.8%

Table C.1.1-10 (Continued)

Fire Area	CDF	Percent of Total
Turbine Building (Upper 4 KV Area)	2.47E-07	2.9%
Reactor Building (NE Corner)	2.18E-07	2.6%
Total	8.37E-06	

Table C.1.1-11 Significant Fire Areas for North Anna 1&2

Fire Area	CDF	Percent of Total
Emergency Switch Gear Room - Instrument Rack Room	2.43E-06	60.8%
Cable and Vault Tunnel - Control Rod Drive Room	4.39E-07	11.0%
Emergency Switch Gear Room -1H Room	3.79E-07	9.5%
Emergency Switch Gear Room -1J Room	3.45E-07	8.6%
Auxiliary Building B Component Cooling Pumps	1.78E-07	4.5%
Total	3.99E-06	

Table C.1.1-12 Significant Fire Areas for Prairie Island

Fire Area	CDF	Percent of Total
Auxiliary Building Ground Floor Unit 1	2.78E-05	44.0%
408V Safeguards Switchgear Room (Bus 121)	8.90E-06	14.1%
Turbine Building Ground and Mezzanine Floor Unit 1	6.44E-06	10.2%
Relay and Cable Spreading Room Units 1 and 2	3.94E-06	6.2%
4KV Safeguards Switchgear Room (Bus 15)	3.67E-06	5.8%
480V Safeguards Switchgear Room (Bus 111)	2.93E-06	4.6%
"B" Train Hot Shutdown Panel & Air Comp/AFW Room	2.25E-06	3.6%
Control Room	1.97E-06	3.1%
"A" Train Hot Shutdown Panel & Air Comp/AFW Room	1.82E-06	2.9%
Total	6.32E-05	

Table C.1.1-13 Significant Fire Areas for Quad Cities 1

Fire Area	CDF	Percent of Total
Unit 1 Turbine Building Ground Floor (South)	1.98E-05	30.0%
Main Control Room	9.51E-06	14.4%
Unit 1 Mezzanine Floor (South)	3.72E-06	5.6%
Auxiliary Transformer 11	3.32E-06	5.0%
Reserve Auxiliary Transformer 12	3.32E-06	5.0%
Unit 1 Switchgear Area (North)	2.91E-06	4.4%
Unit 2 Turbine Building Ground Floor	2.64E-06	4.0%
Unit 1 Cable Tunnel	2.19E-06	3.3%
Unit 1/2 Mezzanine Floor (Central)	2.04E-06	3.1%

Table C.1.1-13 (Continued)

Fire Area	CDF	Percent of Total
Unit 2 Cable Tunnel	1.82E-06	2.8%
Auxiliary Electric Tunnel	1.78E-06	2.7%
Cable Spreading Room	1.52E-06	2.3%
Unit 1 DC Panel Room	1.45E-06	2.2%
Total	6.60E-05	

Table C.1.1-14 Significant Fire Areas for Sequoyah 1&2

Fire Area	CDF	Percent of Total
Aux Building	1.17E-05	74.8%
ERCW Pump Station	3.26E-06	20.8%
Turbine Building	6.78E-07	4.3%
Total	1.56E-05	

Table C.1.1-15 Significant Fire Areas for Waterford 3

Fire Area	CDF	Percent of Total
H&V Mechanical Room	1.95E-06	27.7%
Control Room	1.94E-06	27.6%
Switchgear Room	1.48E-06	21.0%
Emergency Diesel Generator B	5.90E-07	8.4%
Electrical Penetration Area A	4.30E-07	6.1%
Turbine Generator Building	3.17E-07	4.5%
Total	7.04E-06	

Table C.1.1-16 Significant Fire Areas for Washington Nuclear 2

Fire Area	CDF	Percent of Total
Control Room	8.40E-06	47.8%
Turbine Generator Corridor	2.91E-06	16.6%
Div 2 Battery Room	1.48E-06	8.4%
Div 1/Div 2 Elec/Battery Room Corridor	1.06E-06	6.0%
NW Reactor Building	7.77E-07	4.4%
Turbine Generator Building West	5.91E-07	3.4%
Div 1 Electrical Equipment Room	5.54E-07	3.2%
Div 2 Electrical Equipment Room	4.06E-07	2.3%
Equipment Hatch	3.77E-07	2.1%
Total	1.76E-05	

C.1.2 Obtain Performance Data for Risk-Significant, Equipment-Related Elements

Since fire initiating events are modeled at the event level, performance data are obtained for the initiating events themselves for indicator development. Much of the information on fire initiating events comes from an NRC study in 1997 of all fire events from 1986-1994, AEOD/S97-03 (Ref. 18).

C.1.3 Identify Indicators Capable of Detecting Performance Changes in a Timely Manner

Based on these data, the fire initiating event frequencies for these areas range from 6.9E-2 to 8.5E-4. These frequencies (once every 14 years or more on a plant-specific basis) do not allow for timely quantification of changes to the frequencies. Therefore, there were no fire frequency RBPIs. For transient combustible fires, lower-lying elements were considered, such as transient combustible control. However, modeling at this level is not typically detailed enough to support RBPI development. Moreover, data are not currently available to support quantification of indicators at this level.

Fire in a risk-significant area is considered an industry trending indicator.

C.1.4 Identify Performance Thresholds Consistent with a Graded Approach to Performance Evaluation from SECY 99-007

No RBPIs were identified, so no performance thresholds were identified.

C.1.5 Outputs of RBPI Development Process

The frequencies of occurrence of fires in the most commonly risk-significant fire areas listed above will be used for industry trending. There is no impact on inspection areas.

C.2 Mitigating Systems Cornerstone

Key performance areas for fire include fire detection and suppression systems, installed fire barriers, human response, and post-fire safe shutdown systems.

NUREG-1742 indicates that most IPEEE submittals concluded that multi-zone fire scenarios are not significant CDF contributors. However, a review of other available information indicates that the role of physical fire barriers is significant. Although barriers are identified in the IPEEE models, failure of barriers is not explicitly modeled by the IPEEE's. Physical failure of fire barriers may allow propagation of a fire beyond the initial fire area, but the risk significance of this potential, or of leaving fire doors open, is not practical to establish from the information available in the IPEEE submittals.

As defined in the Appendix R Analysis, fire areas are bounded by fire barriers that will withstand the fire hazards within the fire area and protect the equipment within the fire area from a fire outside the area.

C.2.1 Assess the Potential Risk Impact of Degraded Performance

Elements of fire-initiated core damage sequences include the following:

- Occurrence of Fire in Specific Fire Area
- Failure of Detection/Suppression (automatic and/or manual)
- Fire Damage to Plant Systems
- Failure of Post-Fire Safe Shutdown Systems (typically normal mitigation systems that are not affected by the fire scenario, covered in Section 3.1.2)
- Fire Barrier/Separation Effectiveness

It is noted for completeness that NUREG-1742 states that none of the IPEEE submittals evaluated the impact of fires on the reactor protection system. That is to say, none of the submittals discussed a CDF contribution from ATWS sequences.

As identified in the initiating events cornerstone for fire, the risk-significant accident sequences are defined by fire areas. For the mitigating systems cornerstone, the typical risk-significant fire areas are the same as those identified for the initiating events cornerstone, with the same high degree of variability from plant to plant.

The equipment-related elements are the following:

- Detection (automatic)
- Suppression (automatic)
- Safe shutdown systems (including human action)

It is important to note that the IPEEE's have included detection probabilities in the automatic suppression "system" unavailability when automatic suppression is credited in a fire area. Thus, it is not possible to separate detection and automatic suppression contributions to fire CDF, as modeled, in the IPEEE's.

C.2.2 Obtain Performance Data for Risk-Significant, Equipment-Related Elements

Very few data are available for detection and suppression. Generic values are typically used in the IPEEE's for these functions, and are the basis for the calculations below.

Data for post-fire safe shutdown systems are the same as the data used to evaluate those systems' performance in non-fire scenarios.

C.2.3 Identify Indicators Capable of Detecting Performance Changes in a Timely Manner

For generically significant post-fire safe shutdown systems, RBPIs are already developed to the extent practical, as a result of those systems' importance in non-fire scenarios. For detection and suppression equipment, the widely used generic data are "unavailability" data, and do not furnish the kind of event frequency information needed to establish the practicality of detecting performance changes in a timely manner. For purposes of this step, it is tentatively assumed that monitoring at the train or channel level (depending on the system) will turn out to be appropriate.

C.2.4 Identify Performance Thresholds Consistent with a Graded Approach to Performance Evaluation from SECY 99-007

Thresholds for the RBPIs for safe shutdown systems should be quantified in light of the impact of performance declines on fire CDF as well as internal events CDF. This is addressed as part of the development of internal events RBPIs.

For automatic suppression systems, performance data are not currently reported. In addition, although automatic suppression system reliability and availability may in fact be risk significant, development of thresholds based on the information in the IPEEE submittals is not an accurate representation of risk information. Credit for compensatory actions would strongly affect RBPI thresholds for fire detection and suppression systems. Unfortunately, this area is not modeled well enough in available models to address this point adequately within the RBPI program. Thus, development of an indicator for automatic suppression systems is not currently feasible. However, in the event that performance data and improved modeling of suppression systems do become available, typical RBPI thresholds were calculated for several plants based on the available information from the IPEEE submittals. These calculations are solely for demonstration purposes and should not be viewed as proposed thresholds for the reasons discussed above.

Table C.2.4-1 Potential Automatic Suppression System Thresholds for Mitigating Systems Cornerstone - External Events (Fire)

Plant	Automatic Suppression System	Baseline	Thresholds		
			White	Yellow	Red
Browns Ferry 2	N/A	N/A	No automatic suppression credited in significant sequences		
Davis-Besse	wet pipe	2.0E-02	7.88E-02	6.08E-01	-
Duane Arnold	N/A	N/A	No automatic suppression credited		
Fort Calhoun	halon	5.0E-02	5.90E-2	1.40E-1	9.47E-1
	wet pipe	2.0E-02	1.27E-1	8.20E-1	-
Millstone 2	halon	5.0E-02	8.05E-02	3.55E-01	-
	wet pipe	2.0E-02	9.06E-02	7.26E-01	-
Monticello	halon	5.0E-02	8.45E-02	3.95E-01	-
	wet pipe	2.0E-02	3.67E-02	1.87E-01	-
North Anna 1&2	N/A	N/A	No automatic suppression credited		
Prairie Island	CO_2	2.02E-02	5.02E-02	1.42E-01	-
	wet pipe	5.0E-02	2.52E-02	6.98E-02	5.17E-01
Quad Cities 1	wet pipe	2.0E-02	6.85E-02	5.05E-01	None

Table C.2.4-1 (Continued)

Plant	Automatic Suppression System	Baseline	Thresholds		
			White	Yellow	Red
Quad Cities 2	wet pipe	2.0E-02	2.35E-02	5.46E-02	3.66E-01
Sequoyah 1&2	preaction	5.0E-02	5.65E-02	1.15E-01	6.98E-01
Waterford	preaction	5.0E-02	6.96E-02	2.46E-01	-
	wet pipe	2.0E-02	9.29E-01	-	-
Washington Nuclear 2	wet pipe	2.5E-02	1.79E-01	-	-

Notes: A "-" indicates that the threshold is greater than 1.0. Also, the Clinton, Dresden 2&3 and H.B. Robinson 2 IPEEE's were reviewed and determine to credit automatic suppression systems, but insufficient information was contained in the IPEEE to calculate thresholds.

Systems credited by each IPEEE in prevention of core damage, given a fire, were identified for each risk-significant fire area whenever possible. Based on the information available in the IPEEEs, it was not possible to determine the exact contribution to the CCDP due to a given system. In fact, some IPEEEs did not even provide enough information to characterize the roles played by any post-fire safe shutdown systems. Many, however, did identify the "major" contributors to CCDP for each risk significant fire area. For some IPEEEs, enough information is presented to allow the use of an IPE or SPAR model, with appropriate fire-damaged equipment "removed," to determine the assumed contribution to CCDP of a given system. Currently, the information contained in the IPEEEs was only extracted to identify "significant" safe shutdown systems and compare these systems to the systems identified during the development of risk-based performance indicators for internal events. Table C.2.4-2 below lists the safe shutdown systems identified by each IPEEE. The systems are abbreviated using the IPE database standardized abbreviations. Table C.2.4-2 shows that the significant mitigating systems identified for post-fire scenarios that are not captured in the internal events indicators are systems that do not meet the criteria for development into RBPIs.

Table C.2.4-2 Significant Post-Fire Safe Shutdown Systems for Mitigating Systems Cornerstone - External Events (Fire)

Plant	Fire safe shutdown systems that ARE internal events indicators	Fire safe shutdown systems that ARE NOT internal events indicators
Davis-Besse	HPI	MFW, RPS
Dresden 2&3	ICS	
H.B. Robinson 2	CCW, MDAFW, PPORV, SDAFW, SW2	ACBU1, BI, DC, EDC
Millstone 2	MDAFW, SDAFW	RCPS, RPS
Monticello	EAC, HPCI/HPCS, RCIC, SPC	CRDS*, CS*, CTS, LPCI*, MFW, SRVS*, VENT (HPV)*
North Anna 1&2	CCW, CHPI, EAC, ESW, HPI, HPR, MDAFW, PPORV, SDAFW	ACC, AR1, CSI*, HVAC1*, LPI, LPR, MFW, PSRV, SGA

Table C.2.4-2 (Continued)

Plant	Fire safe shutdown systems that ARE internal events indicators	Fire safe shutdown systems that ARE NOT internal events indicators
Waterford 3	MDAFW, SDAFW	DC
WNP 2	SPC	

* Indicates systems that have significant potential as an indicator for internal events, but it is currently uncertain whether this will be an indicator for the particular plant in question.

C.2.5 Outputs of RBPI Development Process

No RBPIs were identified. Many of the systems relied upon to mitigate the effects of a fire are already addressed under internal events. In the event that performance data and improved models addressing suppression systems do become available, development of an appropriate RBPI will be pursued. The inspection areas that could be impacted by this potential RBPI were determined. The results are in Table C.2.5-1.

Table C.2.5-1 Summary of Inspection Areas Impacted by Potential External Event (Fire) RBPIs for Mitigating Systems Cornerstone

RBPI	Attribute	Inspection Area
Fire Suppression System (UR&UA)	Protection Against External Factors	71111.05, Fire Protection

C.3 Barrier Integrity Cornerstone: Containment Performance

According to NUREG-1742, the majority of licensees assessed containment performance by determining whether a fire can lead to containment bypass, isolation failure, or failure of containment heat removal. Only a few performed a more thorough Level 2 PRA assessment. Overall, those few licensees that performed a Level 2 fire PRA indicated that their assessments did not identify any unique containment failure modes or vulnerabilities to early containment failure. One plant identified a new plant damage state (PDS) related to fire-induced core damage, which resulted from fire scenarios that required control room evacuation and could result in spurious opening of containment isolation valves. Based on a review of approximately 25% of the remaining IPEEE submittals, NUREG-1742 determined that a single fire can neither completely destroy the ability to isolate the containment nor fail all of the containment heat removal systems. The majority of licensees for the plants reviewed for this report concluded that the impact a fire on the containment is within acceptable limits when compared to the impact of internal events. Thus, consideration of fire does not lead to any risk-significant LERF scenarios whose containment barrier attributes are not already being addressed under the internal events treatment of the containment barrier.

C.4 References

1. "Determining Potential Risk Significance of Fire Protection and Post-Fire Safe Shutdown Inspection Finding," NRC Inspection Manual, Chapter 0609, Appendix F, U.S. NRC, February 2001.

2. "Perspectives Gained from the Individual Plant Examination of External Events (IPEEE) Program," Draft NUREG-1742, U.S. NRC, April 2001.

3. "Browns Ferry Nuclear Plant Individual Plant Examination for External Events (IPEEE)," Tennessee Valley Authority, July 1995.

4. "Clinton Power Station Individual Plant Examination For External Events Final Report," Illinois Power, September 1995.

5. "Individual Plant Examination of External Events for the Davis-Besse Nuclear Power Station," The Toledo Edison Company, December 1996.

6. "Individual Plant Examination of External Events for Severe Accident Vulnerabilities Submittal Report, Dresden Nuclear Power Station Units 2 and 3," ComEd, December 30, 1997.

7. "Duane Arnold Energy Center Individual Plant Examination of External Events (IPEEE)," IES Utilities, November 1995.

8. "Individual Plant Examination of External Events for Fort Calhoun Station," June 1995.

9. "Individual Plant Examination of External Events for H.B. Robinson 2," Carolina Power & Light Company, June 1995.

10. "Millstone Unit 2 Individual Plant Examination of External Events," Northeast Utilities Services Co., December 1995.

11. "Monticello Individual Plant Examination of External Events (IPEEE)," Revision 1, Northern States Power Co., November 17, 1995.

12. "North Anna Units 1 and 2 Individual Plant Examination of External Events," April 1994.

13. "Prairie Island Individual Plant Examination of External Events (IPEEE)," Revision 1, Northern States Power Co., September 1998.

14. "Quad Cities Individual Plant Examination of External Events (IPEEE) Submittal Report," Revision 1, May 25, 1999.

15. "Sequoyah Nuclear Plant (SQNP) Individual Plant Examination of External Events (IPEEE)," Tennessee Valley Authority, June 1995.

16. "Waterford Individual Plant Examination of External Events," Entergy Operations, Inc., July 1995.

17. "Individual Plant Examination of External Events, Washington Nuclear Plant 2," Washington Public Power Supply System, June 1995.

18. "Special Study - Fire Events - Feedback of U.S. Operating Experience," AEOD/S97-03, U.S. NRC, June 1997.