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MEMORANDUM TO: Thomas L. King, Director
Division of Risk Analysis and Applications
Office of Nuclear Regulatory Research

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FROM: Mark Cunningham, Chief
Probabilistic Risk Analysis Branch
Division of Risk Analysis and Applications
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SUBJECT: TRANSMITTAL OF THE LOW-POWER SHUTDOWN PUBLIC WORKSHOP SUMMARY REPORT

Attached is a summary report of the public workshop on Low-Power and Shutdown Risk, held in Rockville Maryland, April 27, 1999. This report titled "Summary of Information Presented at an NRC-Sponsored Low-Power Shutdown Public Workshop, April 27, 1999, Rockville Maryland" was prepared by Sandia National Laboratories (SNL). It summarizes the presentations given during the workshop, and the discussions held during the general discussion session. It also includes the workshop's agenda, the attendance list, and the view graphs used by the NRC as well as the public.

SNL has forwarded copies of this report to the workshop participants. By copy of this memorandum, the report is being placed in the Public Document Room.

cc: A. Thadani
M. Federline
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J.T. Larkins

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

January 19, 2000

MEMORANDUM TO: Thomas L. King, Director
Division of Risk Analysis and Applications
Office of Nuclear Regulatory Research

FROM: Mark Cunningham, Chief
Probabilistic Risk Analysis Branch
Division of Risk Analysis and Applications
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Alan Rubin for MAC

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SANDIA REPORT

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Summary of Information Presented at an NRC-Sponsored Low-Power Shutdown Public Workshop, April 27, 1999, Rockville Maryland

Timothy A. Wheeler, Donnie W. Whitehead and Erasmia Lois

Prepared by
Sandia National Laboratories
Albuquerque, New Mexico 87185 and Livermore, California 94550

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Summary of Information Presented at an NRC-Sponsored Low-Power Shutdown Public Workshop April 27, 1999 Rockville, Maryland

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Abstract

This report summarizes a public workshop that was held on April 27, 1999, in Rockville, Maryland. The workshop was conducted as part of the United States Nuclear Regulatory Commission's (NRC) efforts to further develop its understanding of the risks associated with low power and shutdown operations at United States nuclear power plants. A sufficient understanding of such risks is required to support decision-making for risk-informed regulation, in particular Regulatory Guide 1.174, and the development of a consensus standard. During the workshop the NRC staff discussed and requested feedback from the public (including representatives of the nuclear industry, state governments, consultants, private industry, and the media) on the risk associated with low-power and shutdown operations.

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Division of Risk Analysis and Applications
Office of Nuclear Regulatory Research
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Washington, DC 20555
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Tsong-Lun Chu, Brookhaven National Laboratory
John Lehner, Brookhaven National Laboratory
Allen Camp, Sandia National Laboratories
Mary Drouin, Nuclear Regulatory Commission

CONTENTS

List of Acronyms and Initialisms

AOT	Allowed outage time
ACRS	Advisory Committee on Reactor Safety
BWR	Boiling water reactor
CDF	Core damage frequency
CRM	Configuration risk management
DG	Diesel generator
DHR	Decay heat removal
EOOS™	Equipment out of service
EPRI	Electric Power Research Institute
ERIN	Engineering Research, Inc.
FTS	fail-to-run
FTR	fail-to-start
HRA	human reliability analysis
IAEA	International Atomic Energy Agency
IE	initiating event
LERF	large early release frequency
LOCA	Loss-of-coolant accident
LPSD	low-power shutdown
NSAC	Nuclear Safety Analysis Center
NRC	Nuclear Regulatory Commission
ORAM™	Outage risk assessment management
POS	Plant operating state
PRA	Probabilistic risk assessment
PSA	Probabilistic safety assessment
PSSA	Probabilistic shutdown safety assessment
PWR	Pressurized water reactor
RCS	Reactor cooling system
RG	Regulatory guide
RHR	Residual heat removal
RPV	Reactor pressure vessel
STP	South Texas Project
WOG	Westinghouse Owners Group

1 INTRODUCTION

1.1 Background

The Office of Nuclear Regulatory Research of the United States Nuclear Regulatory Commission (NRC) has initiated a program on low-power and shutdown (LPSD) nuclear power plant operations. The objective is to provide (or develop, as necessary) an understanding of the risk associated with LPSD operations that is sufficient to support risk-informed regulatory decision-making. The development of this understanding involves a review of the lessons learned from NRC screening studies and from domestic and international work on LPSD risk. A public workshop was conducted on April 27, 1999, in Rockville Maryland, to support this information gathering NRC activity. The objectives of the workshop were to:

- solicit, gather, and share the results of previous and ongoing LPSD evaluations
- identify the LPSD information and methods required for risk-informed regulatory decision-making
- identify an acceptable approach and structure for an LPSD probabilistic risk assessment (PRA) consensus standard

This report summarizes the workshop.

1.2 Workshop Structure

The morning session consisted of presentations by the NRC and representatives of the nuclear power industry. The afternoon session consisted of a general discussion. The workshop was well attended and very successful in generating significant feedback from interested parties. Most of the feedback was given verbally during the general discussion session, but some written comments were submitted as well. This report summarizes the comments received in both forms.

1.3 Organization of the Report

The intent of this report is to capture the main point of the presentations and comments offered as well as those of the written comments. It is *not* intended to provide a verbatim transcript of the actual dialogue that occurred. Chapters 2 and 3 summarize the various presentations. Chapter 4 summarizes information gathered during the open discussion session and from written comments. Appendix A provides the workshop agenda. Appendix B contains the attendance list; Appendix C, copies of the viewgraphs used by the NRC; and Appendix D, copies of the view graphs used by representatives of the nuclear power industry.

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2. NRC PRESENTATION ON LOW-POWER SHUTDOWN RISK

The workshop opened with remarks by NRC Commissioner Nill J. Diaz; Ashok Thadani, NRC Director of the Office of Nuclear Regulatory Research; and Tom King, NRC Director of the Division of Risk Analysis and Applications. The presentation summarized below was given by Mary Drouin, Section Leader, Probabilistic Risk Analysis Branch. The viewgraphs are provided in Appendix C.

1. The risk associated with core damage frequency (CDF) for LPSD plant configurations is of the same order of numerical magnitude as the risk associated with full-power plant operations. This is supported by NRC, domestic, and international industry-sponsored Level 1 LPSD risk assessments. Level 2 and 3 risks for LPSD have not be thoroughly evaluated.

   A comparison of CDF, early fatality risk, and total latent cancer fatality risk results from the Grand Gulf and Surry NUREG-1150 and NUREG/CR-6143 and NUREG/CR-6144 studies indicates that LPSD risks may even be higher than full-power risks. The risks associated with LPSD plant configurations can be highly dependent on the specific plant operating states (POS) during LPSD activities. The instantaneous LPSD risk (per hour) can vary significantly throughout the time in which a plant is in LPSD configurations, and can be significantly higher than the instantaneous risk during full-power operations. Furthermore, important contributors to risk can be significantly different than for full-power operations.

2. Operational events presented (Wolf Creek, Cooper, Clinton, and Washington Nuclear Plant 2) indicate that LPSD risk should be examined.

3. The main differences between LPSD and full-power risks are:

   - The significance of human actions is greater than that during full-power operations.
   - There is a greater reliance on administrative procedures.
   - The vessel and containment may be open during LPSD.
   - Plant configurations change frequently throughout a shutdown.
   - Plant configuration transitions may be risk significant.

4. The objective of the NRC's LPSD research program is to develop an understanding of the risk associated with LPSD plant configurations that is sufficient to support risk-informed regulatory decision-making. At present it includes an assessment of the current LPSD information and the identification of risk significant concerns. Based on the results of this effort, the program could include:

   - research activities (e.g., methods development) as needed, and the investigation and analysis of methods issues,
   - the development of guidance for LPSD risk that would be sufficient to support risk-informed decision-making, and in particular Regulatory Guide (RG) 1.174, and
   - the development of an LPSD consensus PRA standard.

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3. PRESENTATIONS FROM THE NUCLEAR POWER INDUSTRY

Representatives of the nuclear power industry also gave presentations, which are summarized below. Viewgraphs are provided in Appendix D.

3.1 Westinghouse Experience and Insights from Shutdown Risk Projects

Selim Sancaktar of Westinghouse Electric Company, LLC summarized experience and insights gained from their LPSD risk projects.

Westinghouse developed the generic Outage Risk Assessment Management (ORAM™) model based on the Zion nuclear power plant and was responsible for its application to Diablo Canyon. Several NSAC and EPRI references were given (see Appendix D) for both generic and Diablo Canyon-specific documentation of applications of ORAM. It was stated that the application of ORAM has taken LPSD accident sequence evaluation to the "Boil" end state as well as core damage. ORAM has provided thermal-hydraulic analyses for LPSD POSs in terms of thermal margin and inventory margin to provide success criteria. Twelve changes in outage practices are attributable to the original application of ORAM. ORAM was also applied to an LPSD risk assessment for the AP600 PRA. CDF and large early release frequencies (LERF) were calculated for LPSD. LPSD risk was dominated by events related to low reactor coolant system (RCS) inventory conditions.

Westinghouse is developing an LPSD PRA model for the V.C. Summer plant that will be compatible with the plant's full-power PRA. Both the LPSD and full-power models will ultimately be incorporated into a model. The LPSD PRA has three end states defined; boiling, return to criticality, and core damage. Preliminary results suggest that the plant is most vulnerable during mid-loop operations.

Westinghouse has also performed several deterministic analyses to address LPSD safety issues for their customers. These analyses include:

- various loss of residual heat removal (RHR) cooling scenarios;
- plant specific calculations for ties to boiling, times to core uncovery, and guidance to address Generic Letters 87-12 and 88-13; and
- procedure guidance for a Mode 3/4 LOCA when some safeguards systems are removed from service.

A Westinghouse survey of plants owned by members of the Westinghouse Owners Group (WOG) shows that 11 of 36 units have no shutdown model of any sort, while the remaining 25 do have models of varying levels. The most common modeling tool used among these 25 plants is ORAM (12 plants).

Several insights were gained:

- Time windows for operator response to initiators are very important and must be derived from thermal hydraulic analyses.
- Time to boiling is an important indicator of the state of plant vulnerability.
- LPSD risk is dominated by a few periods of high vulnerability.
- Diablo Canyon has incorporated 12 outage risk improvements without extending outage time.
- LPSD risk assessment has been proven to be of practical value.

3.2 Shutdown Risk Monitoring – Scientech

Safety Monitor™ is a computer based risk tool developed by Scientech for the management of LPSD risk. Jeffery Julius and Thomas Morgan of Scientech summarized LPSD risk assessments from the Safety Monitor users group, which involves 18 domestic plants, 15 pressurized water reactors (PWRs) and 3 boiling water reactors (BWRs).

Scientech summarized their LPSD PRA experience as follows:

- San Onofre has used an LPSD PRA since the early 1990's.
- Ten PWR models have been built or are in progress.
- Two BWR models have been built or in progress.
- The Borssele LPSD PRA includes Levels 1, 2, and 3 analyses.
- They participated in the development of the International Atomic Energy Agency's LPSD risk assessment guidelines.

Insights gained include:

- LPSD CDF is less than, but comparable to full-power CDF.
- The instantaneous risk may be higher for LPSD than for full-power, but only for very short durations.
- Most risk is related to low inventory configurations early in the outage.

The Scientech philosophy regarding an LPSD PRA was characterized as:

- Such analyses should be optional.
- They are useful as a supplement to the Defense-In-Depth concepts of NUMARC 91-06.
- They provide insights regarding plant configurations and contingency planning.
- They may support current licensing basis changes (e.g., San Onofre diesel generator (DG) allowed outage time (AOT)).

Scientech believes that the validity of comparisons between LPSD and full-power risk estimates depends on the consistency in methods, level of detail, and modeling assumptions. With regard to a release risk metric, a surrogate Level 3 measure other than LERF might be more applicable for LPSD. It might be better to monitor the status of the containment rather than releases. Scientech also believes that current human reliability analysis (HRA) methods are adequate and that an LPSD PRA standard should be developed after benefits from the full-power standard are understood.

In conclusion, Scientech stated that the NUMARC 91-06 defense-in-depth approach provides sufficient safety margin for plants and that an LPSD PRA should be optional. Furthermore, LPSD PRAs should focus on high-risk POSs.

3.3 Shutdown PSA and EOOS™ at River Bend

Loys Bedel of Entergy Inc. summarized LPSD risk assessment experience at the River Bend plant.

An LPSD PSA has been performed on the River Bend plant using Equipment Out of Service (EOOS™). End states assessed in the analysis were boiling, core damage, fuel pool boiling, prompt criticality, exposed bundles, and containment performance for Level 2. Challenges that confronted the analysts were:

- the definition and quantification of initiating events,
- success criteria changes,
- human reliability analysis,
- recovery actions,
- defense-in-depth modeling, and
- EOOS development.

With regard to HRA and recovery actions, several observations were given. HRA issues included the applicability of the procedures for LPSD events, limited procedural guidance, the issue of what type of indications are available to the operators, and appropriate incorporation of low operator stress levels. The important recovery actions were:

- off-site power,
- decay heat removal (DHR),
- Spent fuel pool cooling, and
- OPDRV/OPDRC.

Recovery actions were assessed with data from Nuclear Safety Analysis Center (NSAC) documents.

The results of the analysis show that RCS boiling frequency is very high in the beginning of outages (0.72 yr^{-1}) and during hydrostatic testing of the reactor pressure vessel (RPV) (0.7 yr^{-1}). However, high RCS boiling frequency does not imply high CDF. CDF is driven by support system maintenance and cannot be directly tied to any Defense-In-Depth status. Fuel pool boiling frequency is very low ($\sim 10^{-9}$ yr^{-1}). However, fuel pool storage risk is not necessarily negligible for full core offload. The cumulative LPSD risk for a 21-day outage may be as great as the annual full-power risk.

It was concluded that performing a PSA for LPSD is viable method, but the situations that must be modeled are dynamic and very different than full-power plant configurations. LPSD risk is driven by the outage schedule and dominated by human error events and recovery actions. The results of LPSD risk assessment are not simple and straightforward, but they can be useful in determining the risk impact of moving maintenance and repair activities from outages to full-power, thus allowing for the overall reduction of risk.

3.4 Perspective on Shutdown Issues at South Texas Project

Steve Rosen of the South Texas Project (STP) discussed LPSD risk assessment activities at STP.

The STP has implemented a shutdown risk assessment group to perform LPSD risk assessment and manage LPSD risk during outages. The group includes an operations manager, shift technical advisor, a risk and reliability analyst, staff from STP's Nuclear Assurance, Nuclear Licensing, and Outage Management organizations. The responsibility of this group is to review the Level 2 outage schedule and prepare a report for the Outage Support Manager and the Plant Manager. The report addresses LPSD safety issues, such as mid-loop operations, RCS pressurization, loss of inventory, LOCA, loss of power, and containment integrity.

The STP has developed several compensatory actions as a result of LPSD analyses. These include procedures and rules to minimize on-site work in the switchyard and on electrical systems during outages, maintaining reactor building containment integrity during mid-loop operations, and putting RHR trains into "protected" status. Extra personnel are also assigned to critical locations during certain outage activities to facilitate the identification of undesired conditions.

LPSD risk is numerically comparable to full-power risk. Furthermore, boiling frequency is comparable to LPSD CDF. Front-end mid-loop operations contribute 15% of LPSD risk in only 1% of the outage time. These results have driven STP to identify compensatory measures (including mid-loop precautions) to protect public health and safety.

3.5 Shutdown Risk Assessment at Seabrook Station – North Atlantic Energy Service Co.

Ken Kiper of North Atlantic Energy Service Company summarized LPSD risk assessment activities as the Seabrook Station.

The Seabrook shutdown PRA was completed in 1988. The scope included analysis of Modes 4, 5, and 6, hot shutdown, cold shutdown, and refueling. Both internal and external initiators were modeled. The models accounted for plant-specific design and operation. The risk analysis included a Level 3 analysis, as well as an uncertainty analysis on the plant configuration, time after shutdown, operator action, and source term.

The results of the PRA indicate that the mean CDF is numerically comparable to full-power CDF, but that the uncertainty range of an LPSD CDF is twice that for full power operations. Estimates for health effects were negligible. The CDF estimate was dominated by loss of RHR at low inventory configurations, RCS drain-down events, and internal initiators, whereas the frequency of releases was dominated by loss-of-coolant accidents (LOCAs). Internal flood and fire events tend to be more likely to occur during LPSD, but the consequences are less likely to be serious.

Shutdown risks can be difficult to quantify because of the complexity of correlating the time available for recovery or response to operator reliability. However, Seabrook believes that LPSD risks are manageable because the risk is driven by alignments and planned equipment outages that can be controlled.

Mr. Kiper claimed that LERF is essentially zero because of the relationship between decay heat and the timing of releases, and the close-in population. Mid-loop operations do not contribute significantly to release frequency because the hatch is closed.

A consensus LPSD standard should allow for the following:

- screen out low thermal margin configurations;
- verify that generic conclusions apply to each plant; and
- apply PRA methods to those potentially high-risk plant configurations that are not screened out.

3.6 Risk Perspective from EPRI Research and Application

Jeff Mitman of the Electric Power Research Institute (EPRI) and Doug True of Engineering Research, Inc. (ERIN) gave a presentation on LPSD risk experience based on EPRI's development and application of the ORAM computer modeling tool.

EPRI identified the implementation of NUMARC 91-06 in 1991 as a benchmark for LPSD risk assessment guidance. EPRI contends that the trend in LPSD risk-significant events has been downward since NUMARC 91-06. This is attributed in part to the development and application of configuration risk management (CRM) tools (e.g., EOOS, ORAM, Safety Monitor) by the U.S. nuclear power industry.

According to EPRI, approximately 55 units have implemented the use of some sort of LPSD CRM tool for risk management at their site. Approximately 20 other units have plans for implementation of CRM tools.

The EPRI ORAM probabilistic shutdown safety assessment (PSSA) methodology was initiated in 1991. CRM has been applied to over 100 refueling outages, for which core boiling and core damage were the primary risk metrics.

EPRI has benchmarked the ORAM tool against high level PRA and PSSA analyses for the South Texas Project units. An ORAM PSSA was compared to a Shutdown PSA developed with RISKMAN™. The risk results for both analyses were comparable once differences in assumptions between the two analyses were reconciled.

EPRI has written over 20 reports on specific ORAM applications and several other technical reports on LPSD issues, such as loss of DHR event trends. EPRI has issued EOOS for use by the nuclear power industry and continues to release enhancements to the tool. ORAM-SENTINEL version 3.3 will be released in September 1999, as a tool for interfacing with LPSD PSA, and ORAM version 4.0 is being developed.

For both BWR and PWR analyses, the LPSD risk is dominated by peak risk periods characterized by relatively high instantaneous risk over short periods of time early during the outage. The risk contribution of these peaks is approximately 86% for both BWRs and PWRs. The average cumulative risk over a 48-day outage for BWRs is approximately 5.0×10^{-6} yr^{-1}, and for a 45-day outage for PWRs is approximately 2×10^{-4} yr^{-1}.

EPRI maintains that LPSD risks have been significantly reduced since the issuing of NUMARC 91-06. Since the risk is dominated by the peaks in instantaneous risk, longer outages are not necessarily safer than shorter ones. The key to reducing LPSD risk is to minimize the length of time in which the plant is in the "peak" risk configurations. The dominant contributor to risk is human error (50%). Other factors that are significant in LPSD risk estimates are the POS, the decay heat level, and the configuration of plant equipment.

EPRI also indicated that initiating event frequencies related to LPSD seem to be decreasing.

EPRI believes that the average cumulative LPSD risk estimates cannot be directly compared with full-power risk estimates. The LPSD risk models are highly outage specific, strongly influenced by the duration of key POSs, and are dominated by human performance issues.

The initiating events for LPSD are well understood and the accident sequence and system modeling is straightforward. However, appropriate success criteria issues have not been fully investigated, and the treatment of POS transitions and human reliability issues is challenging. Furthermore, there has been limited experience with flood, fire, and external event analyses, and Level 2 and 3 risks have been largely unanalyzed.

In conclusion, EPRI believes that computed changes in LPSD risk can range from negligible to huge, depending on the outage schedule. The nuclear power industry has significant expertise and experience for LPSD risk assessment, and methods are well developed but still improving. Significant uncertainties exist with regard to human performance and plant outage activities.

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4. OPEN DISCUSSION

In order to facilitate feedback on four major areas the NRC focused the general discussion:

LPSD risk analysis results,
scope and level of detail needed (or used) in LPSD risk studies,
methods and assumptions used in assessing LPSD risk, and
the appropriate structure and format of an LPSD consensus standard.

This summary includes both verbal and written comments. Time did not allow for extensive elaboration regarding the rationale or the bases for the views expressed.

4.1 Perspectives on LPSD Risk Analysis Results

The NRC opened the discussion on LPSD risk analysis results by presenting the following topics on which it was seeking feedback:

- What is the CDF and LERF range for LPSD conditions?
- What are the dominant contributors to CDF and LERF?

The NRC indicated that it was evident from the morning presentations that there were several different definitions of CDF used in the context of LPSD. These are:

Hourly risk rate ($CDF_1 = X$)
Annualized risk rate ($CDF_2 = X \times Y$, Y = number of hours in a year)
Annual LPSD risk discounted for fraction of year at LPSD configuration ($CDF_3 = X \times Y \times F$, F = fraction of time in configuration)

The NRC clarified that the third CDF definition, CDF_3, is the most appropriate for the NRC's use in risk-informed regulatory decision-making (and in particular the updating of Reg. Guide 1.174) and the development of a consensus standard.

To initiate further dialogue on appropriate LPSD risk metrics, the NRC summarized its interpretation of the main points made during the morning presentations regarding LPSD risk:

- CDF_3 can be significant—of the same order of magnitude as for full-power operations.
- What is an appropriate release metric?
- Current LPSD analyses relate high risk to low thermal margin, but what about LOCAs and drain-down events?

Comments from the discussion are summarized below:

1. Core Damage Frequency Metric

   It was generally agreed that the annual risk metric that accounts for the fraction of the year at which a plant is in LPSD (CDF_3) is the appropriate metric for comparing LPSD risk with full-power risk estimates. However, the other LPSD CDF metrics (CDF_1, CDF_2) were still considered valid for low-power risk management applications.

2. Radioactive Release Metric

Opinions regarding the need to develop a metric to measure the effects of radioactive releases from accidents during LPSD operations were mixed. Views supporting such a metric were expressed as well as views that such a release metric was not necessary. There was virtually unanimous consensus that the LERF metric, as currently defined for full-power risk analysis, was either not completely appropriate or relevant for LPSD.

In support of an LPSD release metric it was stated that the metric should account for the dynamic nature of the source term and decay heat rate as a function of time since the start of the power outage. The LERF metric might be valid for early time periods of an outage when the reactor isotopic inventory and the decay heat rate are relatively similar to the full-power accident scenarios, but as the outage time progresses, the LERF metric would become irrelevant. Possible alternatives to LERF should be investigated. The timing of the release with regard to the time frame of the power outage would be very important.

Viewpoints that questioned the validity of an LPSD release metric were expressed as well. It was argued that the mechanisms required to achieve a release with early fatalities (e.g., pressurized RCS, the appropriate source term) do not exist during LPSD outages. It was suggested that the status of the containment, with particular focus on the containment hatch, for various plant configurations would be a more useful indicator of potential release risk than a release metric.

It was generally acknowledged that there has been very little development of methods or application with regard to LPSD Level 2 and 3 risk analyses.

3. Other Risk Metrics

Two other Level 1 metrics were suggested. The frequency of boiling events, in which boiling is defined as an undesirable end state in the LPSD event tree in addition to the core damage end state, has been calculated in several LPSD risk assessments. Furthermore, several utility LPSD risk assessments have estimated the amount of time that would be required to bring the RCS inventory to boil for each plant configuration during an outage. This metric, referred to as time-to-boil, is used as a measure of the thermal margin available to the plant staff at any time.

4.2 Perspective on LPSD Scope and Level of Detail

The NRC summarized its main points of interest regarding the scope and level of detail of the LPSD program as follows:

- Should fuel handling and storage be included in the scope of LPSD risk assessment?
- Should all LPSD POSs be analyzed?
- Should the transitions between POSs be explicitly modeled?
- Should the level of detail be comparable to full-power (i.e., the same rigor)?

It summarized the main points made during the morning presentations on this subject.:

- Forced and unplanned outages, fuel handling, fuel misloading, and fuel pool storage should be within the scope of an LPSD risk assessment.
- Risk associated with the transition between POSs should be within the scope of an LPSD risk assessment. Would a dynamic type of tool be required for this?

- Internal fires and floods, and external events are potentially important for LPSD risk.
- The level of detail should be similar to the detail and rigor of a full-power risk assessment.

Public comments are summarized below.

1. Fuel Pool Cooling, Fuel Handling, and Fuel Misloading

   Comments regarding fuel handling and fuel pool cooling consistently indicated that such activities have been analyzed for risk, and that the risk associated with these activities is insignificant compared with both full-power and LPSD risk. Specifically with regard to fuel misloading, the general opinion of the attendees was that so many checks are in place that the probability of a mishandling event is negligible. With regard to fuel handling, the view was expressed that data on crane mishaps are very old and not relevant to the nuclear power industry's operational experience. It was suggested that if actual operational data were collected and developed for fuel handling risk assessment, they would show that the risk was negligible. The general consensus was that fuel handling, misloading, and fuel pool cooling should not necessarily be included in the scope of an LPSD risk assessment. However, it was pointed out that a risk management tool used by several utilities, Safety Monitor, is designed to include fuel pool cooling. Its was also stated that fuel pool cooling typically is incorporated into a plant's LPSD risk management by the users of Safety Monitor when the activity is relevant to a particular outage.

2. Unplanned Outages

   There was general acknowledgment that the risk management of unplanned outages has not been investigated as extensively as that for planned outages. One concern that was expressed was that a gap may exist in current risk assessment and risk management involving unplanned outages for which the plant configuration must be further altered from its initial shutdown state to repair the component or resolve the problem that caused the unplanned outage. The view was expressed that risk should be modeled for such unplanned outages as well as other unplanned outages in which the plant configuration is intentionally altered to perform other maintenance (i.e., the utility decides to take "advantage" of the outage to perform maintenance). It was stated that it would be useful to look at risk management of unplanned outages. However, the view was also expressed that unplanned outages cannot be fully accounted for by risk monitoring methods, and thus should be explicitly incorporated into LPSD risk models.

3. Internal Fire and Flood and External Events

   It was suggested that LPSD risks associated with fire, flood, and seismic events have not been well analyzed to date and the potential for significant risk should not be dismissed. However, it was also suggested that LPSD seismic risk assessments could benefit extensively from the work done on full-power seismic risk by using these assessments as the basis for LPSD seismic analyses. It was also suggested that, of this set of events, fire and flood risk assessments for LPSD might be the "trickiest" to analyze. However, the view that LPSD fire risks are insignificant was expressed as well. It was felt that if full-power fire risk was insignificant, then the LPSD fire risks should also be insignificant. Furthermore, it was felt that fire risk is controlled through adherence to Appendix R. However, the views that full-power fire insights cannot be extrapolated to LPSD plant activities, and that Appendix R does not eliminate the potential for significant fire risk were expressed as well.

4. Plant Operating State Transition Risk

   It was suggested that the risk associated with the transition between POSs does not need to be analyzed. It was felt that since such situations are tightly controlled and exist for only short periods of time, a risk assessment is not warranted. However, another view suggested that risk assessment of transitional conditions may be valuable for unplanned outages, especially those outages that require component repair. It was also suggested that transitional states could be modeled as additional POSs, and that no basis exists for dismissing such states as risk insignificant.

5. Need for Level of Detail Analogous to Full-Power PRA

   A range of opinions were expressed on the need for "full-power" level of detail in LPSD risk assessment. There was a strong consensus among many attendees that highly detailed methods commensurate with those of full-power risk assessment should be applied to LPSD risk assessment only when appropriate. The level of detail required should be determined by the particular application and plant configurations being assessed. A high level of detail should be needed only for cases in which the risk potential is high. It was also suggested that PRA techniques may not be needed at all, but that qualitative defense-in-depth concepts may be adequate. However, the view that "PRA type" methods should be applied to all plant configurations was expressed as well. Furthermore, it was also stated that an insufficient level of detail could yield erroneous risk assessments.

4.3 Perspectives on LPSD Methods

The NRC summarized its main points of interest as follows:

- What available methods are appropriate?
  - What scope and level of detail does the method address?
  - What are the key assumptions used in the method?
- What improvements or research are needed?
  - Methods for modeling PRA elements?
  - Tools and software needed to analyze models?

The NRC summarized its interpretation of the main points of the morning presentations as follows:

1. Key assumptions for:
   - Defining POSs
   - Identifying IEs
   - Defining success criteria
2. What codes and methods should be used for:
   - Screening criteria
   - Human error analysis
3. Data
   - Fails-to-start (FTS)/Fails-to-run (FTR)—Same as full-power PRA?
   - Unavailabilities
4. Research needs
   - Uncertainty methods
   - Release metrics
   - Code enhancements and additional analyses

The public comments are summarized under four categories: general risk methods, human error analysis, data, and research needs.

1. General Risk Methods

Statements supporting both detailed, traditional PRA-type analyses as well as qualitative methods were voiced in the discussion on risk assessment methods. The view that traditional PRA approaches are necessary for valid risk-informed regulatory decisions was expressed. However, it was also felt that qualitative methods might be sufficient to capture the majority of risk insights. It was also stated that risks associated with LPSD activities cannot be determined by generic analyses.

The need for greater work in the area of success criteria, including possible benefits from research programs designed to address special LPSD thermal-hydraulic issues, was also voiced. Such areas include boron dilution events, alternatives to DHR, and reflux cooling.

It was suggested that LPSD risk assessment methods should facilitate the evaluation of risk trade-offs between continued operation with on-line repair and shutting down to repair.

2. Human Error Analysis

There was general agreement by the participants that human error is a large contributor to LPSD risk. However, there was less agreement as to whether there is a need to improve human reliability analysis methods for LPSD. Most of the attendees believe that the methods available to quantify human error are applicable to LPSD because the human errors of commission are latent and are accounted for in the initiating event frequency and equipment failure rate. However, previous NRC work (documented in NUREG/CR-6093) suggests that there are human errors of commission that are not latent errors, and these are not generally modeled in PRAs, and can be important to LPSD risk. It was suggested in the workshop that PWR mid-loop operations might be a good application for the NRC's ATHEANA program.

3. Data

The general view expressed at the workshop was that data should be developed for plant-specific quantification of risk events. It was stated that average unavailabilities should be suitable for most component failures, but not for initiating events, maintenance frequencies and durations, and common-cause failure rates. It was further suggested that LPSD maintenance data should be collected and analyzed to support the modeling of unscheduled maintenance events during LPSD (see Unplanned Outages under Section 4.2).

4. Research Needs

Several areas of potential research were identified:

1. Boron dilution events;
2. maintenance or testing-induced drain-down events;
3. nuclear grade crane failures;
4. impact of the definition of "success terms" on the selection of computational tools;
5. fire and flood initiators;

6. impact of emergency procedures, plant technical specification., emergency action guidance/levels on LPSD modeling assumptions;
7. cold overpressure events.

4.4 Perspectives on LPSD Standards

The NRC summarized its main points of interest as follows:

- What should be the scope and structure of an LPSD risk assessment standard?
- What are the appropriate risk metrics?
- What methods should be endorsed by such a standard?

The NRC also summarized its interpretation of the main points made during the morning presentations as follows:

- An LPSD standard should be similar to the full-power standard.
- Level 2 risk metrics need additional study.
- Available methods provide a starting point for a standard, but are not sufficient to address all aspects of LPSD risk.

The public comments are summarized below.

A range of opinions on the general need for a standard were voiced during the workshop. One view was that the development of an LPSD standard should be delayed until the full-power standard has been finalized so that "lessons learned" can be incorporated into the development of an LPSD standard. However, another view was that an LPSD standard was needed sooner than later or else it would be too late to facilitate consistency in LPSD risk assessment approaches across plants. There was a strong consensus that any LPSD standard should not necessarily be similar to the full-power standard, but that it should address the unique applications needs as well as the risk needs of LPSD activities. There was no clear consensus with regard to the structure, content, and desired timing of an LPSD standard, but there was a strong consensus that such a standard would be useful. Nevertheless, a view was also expressed that an LPSD standard was unnecessary.

It was suggested that an LPSD standard should be high level in nature without detailed prescriptions of methods. However, it was also suggested that a standard should be prescriptive in certain areas with significant risks, but high level in other areas less risk significant. One opinion voiced was that a standard should include minimum requirement to ensure proper configuration control, especially during high-risk evolutions.

The NRC was encouraged to become familiar with current risk assessment tools used for LPSD risk management (e.g., Safety Monitor, ORAM) and with recent applications of these tools to LPSD risk management.

APPENDIX A. WORKSHOP AGENDA

Workshop Agenda

7:45 am to 8:15 am	Introduction, NRC presentation
8:15 am to 10:15 am	Industry presentation (Westinghouse, Scientech, Seabrook, River Bend, South Texas)
10:15 am to 10:30 am	BREAK
10:30 am to 11:40 am	Industry presentation (NEI and EPRI)
11:40 am to 1:00 pm	LUNCH
1:00 pm to 1:30 pm	General Discussion: Perspectives on LPSD results
1:30 pm to 2:00 pm	General Discussion: Perspectives on LPSD scope and level of detail
2:00 pm to 2:15 pm	BREAK
2:15 pm to 3:30 pm	General Discussion: Perspectives on LPSD methods
3:30 pm to 3:45 pm	BREAK
3:45 pm to 4:30 pm	General Discussion: Perspectives on LPSD standard
4:30 pm to 4:50 pm	General Discussion: Other issues
4:50 pm to 5:00 pm	Wrap-up

APPENDIX B. WORKSHOP REGISTRATION LIST

Table B-1 Workshop Registration

Name	Affiliation
Michael Adelizzi	PP&L Resources, Inc., Susquehanna Steam Electric Station
Loys Bedell	Entergy - River Bend Station
Biff Bradley	Nuclear Energy Institute
Robert Budnitz	Future Resources Associates Inc.
Ken Bych	PG&E Diablo Canyon
Kendall Byrd	First Energy Nuclear Operating Company, Davis-Besse Nuclear Power Plant
Allen Camp	Sandia National Laboratories
Bryan Carroll	Duke Power Co.
Mark Caruso	Nuclear Regulatory Commission (NRR/DSSA)
Pat Castleman	ECM/NJD
Richard Cathy	Southern Nuclear Plant Vogtle
Mark Cheok	Nuclear Regulatory Commission (NRR/DSSA)
Bob Christie	Performance Technology
Tsong-Lun Chu	Brookhaven National Laboratory
Fred Cietek	Millstone/NNECO
Nancy Closky	Westinghouse Electric Company
Mark Cunningham	Nuclear Regulatory Commission (DRAA/PRAB)
Nill Diaz	Nuclear Regulatory Commission (Commissioner)
Mary Drouin	Nuclear Regulatory Commission (DRAA/PRAB)
Leslie Collins	ABB CENP
Lester Ettlinger	Oxford Group and American Nuclear Society
Anees Farruk	Southern Nuclear
David Finnicum	ABB
Mark Flaherty	Rochester Gas & Electric/Ginna Station
Robin Franke	Baltimore Gas & Electric, Constilation Energy Corporation, Calvert Cliffs
Kim Green	NUS Information Services

Table B-1 Workshop Registration

Name	Affiliation
Ching Guey	FPL/Nuclear Engineering
John H. Emmett	Pennsylvania Power & Light, Susquehanna Steam Electric Station
Jim Hawley	American Electric Power
Harry Heilmeier	Framatome Tech
Tony Hsia	Nuclear Regulatory Commission
Roger Huston	Licensing Support Services
Jeffrey Julius	Scientech, Inc.
Bill Ketchum	Wolf Creek Nuclear Operating Corporation
Kenneth Kiper	North Atlantic Energy Service Corporation
Tom King	Nuclear Regulatory Commission (DRAA)
Gregory Krueger	PECO Energy
John Lehner	Brookhaven National Laboratory
Stanley Levinson	Framatome Technologies
Clem Littleton	Pilgrim Nuclear Power Station, Boston Edison
Erasmia Lois	Nuclear Regulatory Commission (DRAA/PRAB)
Stan Maingi	Pennsylvania Bureau of Radiation Protection
Asimios Malliakos	Nuclear Regulatory Commission
Michael Markley	Nuclear Regulatory Commission (ACRS Staff)
Jonathan Mawsell	General Physics
Mark Melnicoff	Commonwealth. Edison Nuclear Engineering Services - Risk Management
William Mims, Jr.	Tennessee Valley Authority
Jeff Mitman	Electric Power Research Institute
Parviz Mojini	Southern California Edison (SCE)
Thomas Morgan	Scientech, Inc.
Craig Nierode	Northern States Power Company
Gareth Parry	Nuclear Regulatory Commission (NRR/DSSA)

Table B-1 Workshop Registration

Name	Affiliation
Michael Phillips	Scientech, Inc.
Marie Pohida	Nuclear Regulatory Commission (DSSA/SPSB)
Steve Rosen	STP Nuclear Operating Company
Selim Sancaktar	Westinghouse Electric Company
Mohammed Schuabi	Nuclear Regulatory Commission (NRR/SRXB)
Leo Shanley	ERIN
Nathan Siu	Nuclear Regulatory Commission (DRAA/PRAB)
David Stellfox	McGraw Hill
Jeff Stone	Baltimore Gas & Electric, Constilation Energy Corporation, Calvert Cliffs
Theresa Sutter	Bechtel
Ashok Thadani	Nuclear Regulatory Commission (Director of the Office of Nuclear Regulatory Research)
Tatsuya Tamirami	Toloyo Electric Power Co. Inc., Washington Office
George Thomas	Nuclear Regulatory Commission (DSSA/SRXB)
Thomas Timmons	Westinghouse Electric Co.
Nick Trikouros	GPU Nuclear Corp.
Doug True	ERIN
Kenneth Tuley	Virginia Power
James Tunink	Ameren UE/Callaway
Donald Vanover	ERIN
L. Victory Jr.	Enertech Servus
Donald Wakefield	PLG/EQE
Timothy Wheeler	Sandia National Laboratories
Robert White	Consumers Energy
Donnie Whitehead	Sandia National Laboratories
Millard Wohl	Nuclear Regulatory Commission (NRR/SPSB)
Antonios Zoulis	New York Power Authority

APPENDIX C. NRC PRESENTATION MATERIAL

U.S.

NUCLEAR REGULATORY COMMISSION

NRC PUBLIC WORKSHOP ON LOW POWER SHUTDOWN RISK

Rockville, Maryland

April 27, 1999

HISTORICAL PERSPECTIVES

- Previous NRC studies and operational events indicate LPSD risk comparable to full-power risk
- ACRS recommended to Commission research activities to gain a better understanding of LPSD risk
- Commission direction

NRC Developing a LPSD Program

Objective:

- Develop an understanding of LPSD risk sufficient to support regulatory decision-making
  (Risk defined as core damage frequency and large early release frequency)

Scope:

- Assess current LPSD information and identify risk significant concerns
- Perform research activities (e.g., methodology development),if needed, to further investigate or analyze these concerns
- Develop guidance for LPSD risk sufficient to support risk-informed decision-making
- Support development of LPSD consensus PRA standard

OBJECTIVES OF WORKSHOP

Solicit and gather information to support staff LPSD program

- Share results of previous and on-going LPSD
- Identify LPSD information and methods needs sufficient for regulatory decision-making
- Identify acceptable approach and structure for LPSD PRA consensus standard

WORKSHOP AGENDA

7:45 am to 8:15 am	Introduction, NRC presentation
8:15 am to 10:15 am	Industry presentation (Westinghouse, Scientech, Seabrook, River Bend, South Texas)
10:15 am to 10:30 am	BREAK
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1:00 pm to 1:30 pm	General Discussion: Perspectives on LPSD results
1:30 pm to 2:00 pm	General Discussion: Perspectives on LPSD scope and level of detail
2:00 pm to 2:15 pm	BREAK
2:15 pm to 3:30 pm	General Discussion: Perspectives on LPSD methods
3:30 pm to 3:45 pm	BREAK
3:45 pm to 4:30 pm	General Discussion: Perspectives on LPSD standard
4:30 pm to 4:50 pm	General Discussion: Other issues
4:50 pm to 5:00 pm	Wrap-up

Workshop Structure

- Morning presentations given without interruption, questions and comments will be held in afternoon discussion sessions
- Individuals are to speak at a microphone, state their name and affiliation
- Blank forms are available in each package and at each table for written comments
- All questions and comments, whether verbal or written will be summarized in public document
- Workshop agenda times will be enforced, therefore, questions, comments and discussions may be limited
- Blank registration form in package, please complete and turn in

CURRENT UNDERSTANDING OF LPSD RISK

- Risk comparable to full-power operation
- Risk varies among plant operating states
- Contributors can be significantly different than those at full-power
- Instantaneous LPSD risk (per hour) can be higher than instantaneous full-power risk
- Based on NRC and international studies and operational experience

RESULTS FROM NUREG-1150 AND LPSD STUDIES

Distributions for Core damage frequency and aggregate risk for POS 5 and full-power operation for Grand Gulf

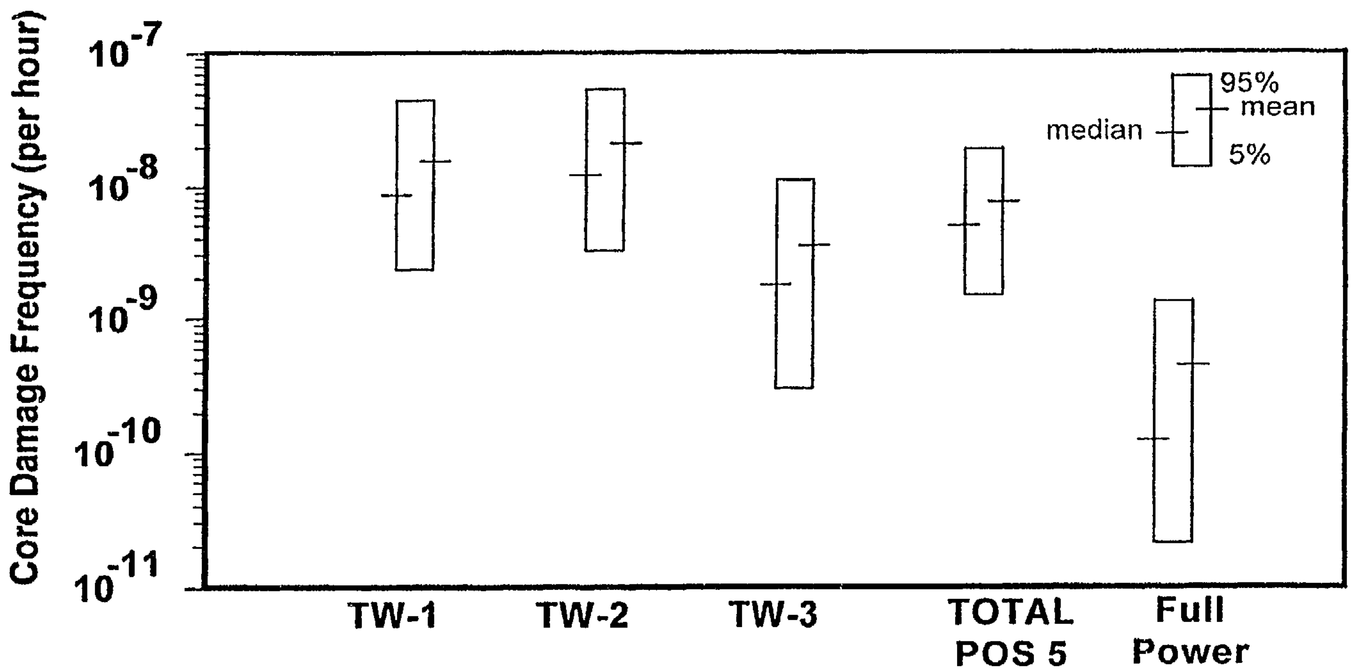
Analysis	Descriptive Statistics (All values are per calendar year) Percentiles 5th	50th	95th	Mean
	Core Damage Frequency			
POS 5	4.1×10^{-7}	1.4×10^{-6}	5.6×10^{-6}	2.1×10^{-6}
Full Power	1.8×10^{-7}	1.1×10^{-6}	1.4×10^{-5}	4.1×10^{-6}
	Early Fatality Risk			
POS 5	3.7×10^{-11}	2.8×10^{-9}	3.9×10^{-8}	1.4×10^{-8}
Full Power	2.5×10^{-12}	6.1×10^{-10}	2.6×10^{-8}	8.2×10^{-9}
	Total Latent Cancer Fatality Risk			
POS 5	4.3×10^{-4}	1.9×10^{-3}	1.2×10^{-2}	3.8×10^{-3}
Full Power	1.4×10^{-5}	2.4×10^{-4}	2.3×10^{-3}	9.5×10^{-4}

RESULTS FROM NUREG-1150 AND LPSD STUDIES

Distributions for Core damage frequency and aggregate risk for mid-loop and full-power operation for Surry

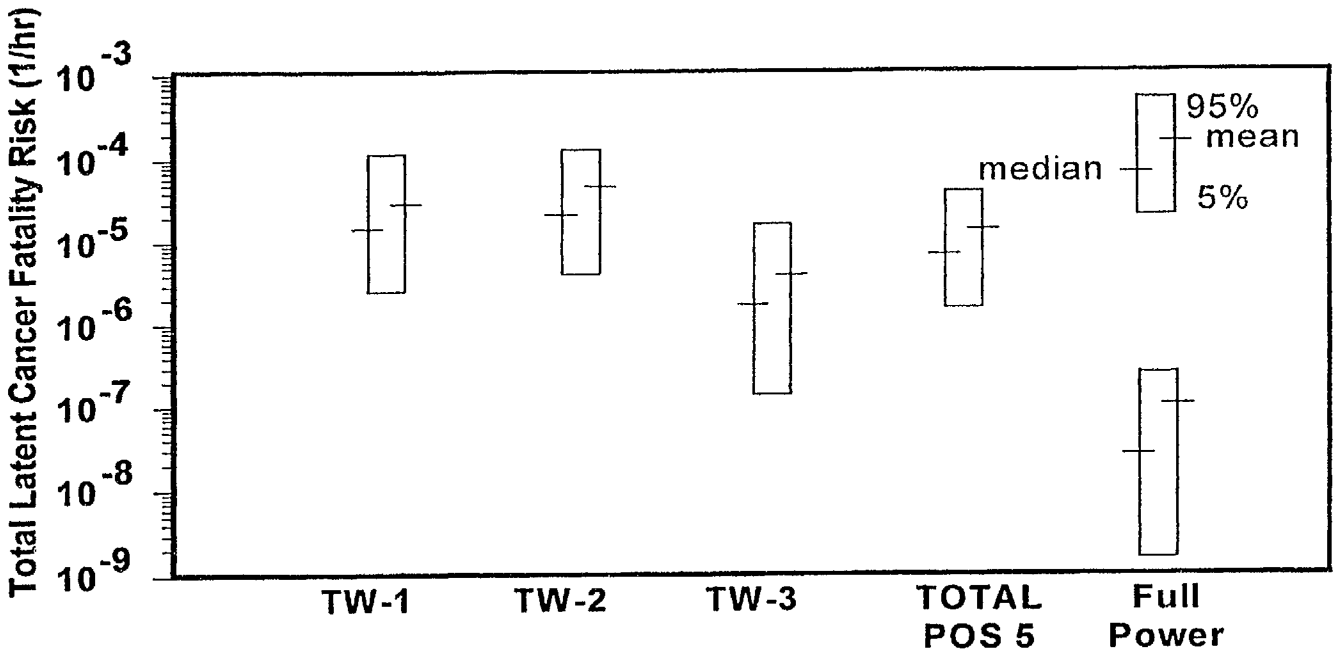
Analysis	Descriptive Statistics (All values are per calendar year)			
	Percentiles			
	5th	50th	95th	Mean
	Core Damage Frequency			
Mid Loop	3.2×10^{-7}	2.0×10^{-6}	1.9×10^{-5}	4.2×10^{-6}
Full Power	9.8×10^{-6}	2.5×10^{-5}	1.0×10^{-4}	4.1×10^{-5}
	Early Fatality Risk			
Mid Loop	1.3×10^{-10}	3.6×10^{-9}	1.6×10^{-7}	4.9×10^{-8}
Full Power	7.6×10^{-10}	7.0×10^{-8}	5.4×10^{-6}	2.0×10^{-6}
	Total Latent Cancer Fatality Risk			
Mid Loop	8.0×10^{-4}	5.3×10^{-3}	5.5×10^{-2}	1.6×10^{-2}
Full Power	3.1×10^{-4}	2.2×10^{-3}	1.9×10^{-2}	5.2×10^{-3}

Grand Gulf



TW - Time Window

Grand Gulf



TW - Time Window

OPERATIONAL EVENTS

- Wolf Creek (9/17/94) Drain Down Event (AEOD/S95-01)
  - Inadvertent blowdown of about 9200 gallons of reactor coolant through RHR system to the refueling water storage tank.
  - Involved concurrent manipulations of RHR valves while cooling down to begin a refueling outage.
  - Terminated by operators before reactor hot leg uncovered and steam introduced into supply line for the ECCS pumps potential common cause failure

- Cooper (NRC IR 50-298/98-08) Human Action Renders RHR Loop A Inoperable
  - Review of maintenance activities fails to identify potential for causing both methods of RHR room cooling (room cooler and natural air circulation)

OPERATIONAL EVENTS

- Clinton (2/13/98) Loss of Shutdown Cooling (LER 461/98-003)
  - Shutdown cooling isolated when a common line suction valve for the RHR system went shut.
  - Valve closed due to a de-energization of the Division II Nuclear Systems Protection System (NSPS) bus.
  - NSPS bus became de-energized because the inverter reverse transferred to the bypass transformer which was out of service for maintenance.
  - Inverter reverse transferred due to the failure of a 12 volt power supply that was being supplied by the Division II NSPS bus.
- Washington Nuclear Plant - Unit 2 Flooding of ECCS (LER 397/98-011)
  - A significant water hammer event in the fire protection system piping resulted in the catastrophic failure of fire protection valve.
  - Water from the ruptured fire protection valve flooded the RHR C and LPCS rooms.
    - Water entered RHR C room through the water-tight door which had not been properly dogged closed.
    - A floor drain isolation valve failed to automatically close, providing a flow path from the RHR C room sump to the floor drains in the LPCS room. Water flowed through this pathway from the sump up through the drains in the LPSC pump room

DIFFERENCES BETWEEN LPSD AND FULL-POWER

- ❒ Increased significance on human actions
- ❒ Greater reliance (dependency) on administrative procedures
- ❒ Open vessel and containment
- ❒ Varying configurations
- ❒ Mode (plant operational states) transitions

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INDUSTRY PRESENTATIONS

GENERAL DISCUSSIONS

PERSPECTIVES ON LPSD RESULTS

- ❑ What is the CDF and LERF range for LPSD conditions?
- ❑ What are the dominant contributors to CDF and LERF?

Perspectives on LPSD Results: Published Q's

1. Are LPSD core damage frequency (CDF) and large early release frequency (LERF) comparable to full power CDF and LERF?

2. Are the LPSD CDF and LERF contributors comparable to the contributors from full power?

Perspectives on LPSD Results – Insights from Industry Presentations

- Core damage frequency
  - CDF_1: "x" (hourly rate)
  - CDF_2: x *y (hours in a year)
  - CDF_3: x * y * f (fraction of time in configuration)
- CDF_3 from LPSD can be significant and of the same order of magnitude
- To support RG 1.174, need CDF_3
- What is appropriate release metric?
- LPSD dominated by configurations with short boil-off times (LOCAs and drain-down events?)

Perspectives on LPSD Scope & Level of Detail

- ❑ Should fuel handling and storage be included?
- ❑ Include all plant operating states?
- ❑ Include transitions between states?
- ❑ Should the level of detail be comparable to full-power (e.g., same level of rigor)?

Perspectives on LPSD Scope & Level of Detail: Published Q's

4. Should the scope of LPSD analyses include fuel handling and storage, e.g., full core offloading?

6. Is the CDF and LERF associated with the transition from one operational state to another important?

Perspectives on LPSD Scope & Level-of-Detail – Insights from Industry Presentations

- Include forced/unplanned outages, fuel handling and storage
- Transitional risk: a "dynamic" type of tool required?
- If internal flood, fire, seismic important at full-power, potentially important at LPSD
- Level of detail similar to full-power

PERSPECTIVES ON LPSD METHODS

- What available methods are appropriate?
  - What scope and level of detail does the method address?
  - What are the key assumptions used in the method?
- What improvements or research are needed?
  - Methods for modeling PRA elements?
  - Tools and software needed to analyze models?

PERSPECTIVES ON LPSD METHODS: PUBLISHED Q'S

1. (Are LPSD CDF and LERF comparable to full power CDF and LERF?) What methods and assumptions should be used to answer this question?
2. (Are the LPSD CDF and LERF contributors comparable to the contributors from full power?) What methods and assumptions should be used to answer this question?
3. How many plant operational states (POS) are needed to adequately represent the risk associated with LPSD operations?
4. (Should the scope of LPSD analyses include fuel handling and storage, e.g., full core offloading?) What methods and assumptions should be used to answer this question?
5. Is there a sufficient technical basis (knowledge of core melt phenomena, source terms, varying containment configurations, etc.) available to support LERF analysis for LPSD? If not, what issues require additional study? If a sufficient technical basis exists, what information sources can be cited to support the assertion?
6. (Is the CDF and LERF associated with the transition from one operational state to another important?) What methods and assumptions should be used to answer this question?

13. Can NUREG/CR-6595 be used to calculate LERF for LPSD conditions? If not, what additional guidance should be added to the report to support LERF calculations for LPSD conditions?
14. Are average equipment unavailabilities during LPSD conditions (resulting in average CDF and LERF estimates) sufficient to support risk-informed decision-making?
15. Is the following definition of an initiating event during LPSD adequate: "An event that causes loss of the function(s) necessary to maintain the plant in its existing operating state?" If not, then what changes should be made to enhance the definition?
16. Are there generic data sources for the identification and quantification of LPSD initiating events? If so, are the data sources publicly available? Are these generic data sources consistent?
17. Do certain LPSD operational states have the potential to have more human failures than full power operation? If event trees and fault trees are used to model the response of a plant to LPSD initiating events, where is the more appropriate place to model these human failures? What is the basis for this choice?
18. What improvements are required to ensure an adequate representation of human actions during LPSD conditions
19. What are the important uncertainties (parameter, model, and completeness) that should be considered in LPSD analyses? How should these uncertainties be evaluated in LPSD analyses?

Perspectives on LPSD Methods - Insights from Industry Presentations

- Using traditional PRA methods
- What are the key assumptions?
  - Defining POSs
  - Identifying IEs
  - Defining success criteria, etc.
- What codes/methods should be used?
  - Success criteria
  - HRA
- Data
  - FTS/FTR - same as full-power
  - Unavailabilities - need to be plant-specific, use running averages
- Research/improvement needs
  - Methods for simplified uncertainty
  - Release metric
  - Code enhancements and additional analyses

PERSPECTIVES ON LPSD STANDARD

- What should be the scope and structure of the standard?
- What are the appropriate risk metrics?
- What method(s) should be endorsed by the standard?

PERSPECTIVES ON LPSD STANDARD: PUBLISHED Q'S

7. Is a traditional PRA approach needed to provide an understanding of LPSD for risk-informed regulatory decision-making? If not, what other approaches are available? What are their strengths and limitations?

9. Draft NUREG-1602 provides reference material on the scope and quality of a LPSD PRA. Is the information in this draft complete and correct? Is it useful as reference material in making assessments on an application specific basis on the scope and quality of a LPSD risk assessment to support that particular application? How could it be improved?

10. Would draft NUREG-1602 be useful as a starting point to develop a standard on LPSD PRA? What would be needed? Should it specify acceptable LPSD PRA methods?

11. Given the lack of experience in performing LPSD PRAs, should a standard for LPSD PRA provide both (1) requirements for what activities should be performed and (2) detailed information/instructions on how those activities should be performed?

12. Is LERF an appropriate metric for meeting the Safety Goal Policy Statement for all POS? If not, what metrics should be used? For example, should there be a metric on long term release frequency to supplement LERF? What should it be based upon?

Perspectives on LPSD Standard – Insights from Industry Presentations

- ❑ LPSD Standard should be similar to Full Power Standard.
- ❑ Level 2 risk metrics need additional study.
- ❑ Available methods provide a starting point for a standard, but are not currently sufficient.

APPENDIX D. PRESENTATION MATERIAL OF PUBLIC PRESENTATIONS

WESTINGHOUSE EXPERIENCE AND INSIGHTS FROM SHUTDOWN RISK PROJECTS PERFORMED

PREPARED FOR PRESENTATION AT THE NRC

SHUTDOWN RISK WORKSHOP

ON APRIL 27 1999

PRESENTED BY

SELIM SANCAKTAR

RELIABILITY AND RISK ASSESSMENT SERVICES
WESTINGHOUSE ELECTRIC COMPANY LLC

OUTLINE

CONTRIBUTION TO ORAM	1992 - 1994
AP600 SHUTDOWN PRA	1990 - 1997
V.C. SUMMER SHUTDOWN PRA	1998
DETERMINISTIC ANALYSES	
PSA SURVEY RESULTS	
INSIGHTS	
QUESTIONS	

CONTRIBUTION TO ORAM

Westinghouse developed the generic ORAM model (based on Zion) and its application to the Diablo Canyon.

The generic ORAM model for PWRs is documented by the following reports prepared by Westinghouse:

1. Survey of PWR Plant Personnel on Shutdown Safety Practices..., March 1992, NSAC-174
2. Safety Assessment of PWR Risk During Shutdown Operations, August 1992, NSAC-176L
3. Risk of PWR Inadvertent Criticality During Shutdown and Refueling, December 1992, NSAC-183
4. Generic Outage Risk management Guidelines for PWRs, December 1993, EPRI TR-102970
5. Reflux Cooling: Application to decay Heat Removal During Shutdown Operations, March 1994, EPRI TR-102972

The Diablo Canyon application is documented in the following reports:

1. Safety Assessment of Diablo Canyon Risks During Shutdown Operations, June 1993, NSAC-195L
2. Contingency Strategies for Diablo Canyon During Potential Shutdown Operation Events, December 1993, EPRI TR-102969
3. Outage Risk management Guidelines for Diablo Canyon During Shutdown Operations, December 1993, EPRI TR-102981

CONTRIBUTION TO ORAM

ORAM program took the shutdown accident sequences to "BOIL" endstates, as well as CD endstates.

ORAM provided thermo hydraulic analyses for shutdown states in

thermal margin

inventory margin

to provide success criteria.

Twelve outage practice changes are attributable to the results of the original ORAM application.

AP600 SHUTDOWN PRA

For the AP600 design approval process, the NRC requested the performance of a Shutdown PRA, in addition to the power operation PRA.

The shutdown PRA was performed to be consistent with the at-power PRA and it calculated plant LERF as well as plant CDF.

The CDF frequency at shutdown and low power operations is less than one-third of the CDF from at-power events.

The LERF frequency at shutdown is about 25 % of the shutdown CDF.

63% of the early impaired containment frequency comes from events that bypass the containment (such as pre-existing containment opening during the event). The largest contributor to the early impaired containment state is an open equipment hatch, which cannot be quickly and easily closed.

In-containment refueling water storage tank has a high risk increase worth, which indicates that it is a valuable asset in keeping CDF low.

RHR pumps and EDGs rank high in risk decrease importances.

The majority (85%) of shutdown CDF risk still comes from events during RCS drained conditions.

V.C. SUMMER SHUTDOWN PRA MODEL

V.C. SUMMER has chosen to create a shutdown model compatible with and complementing the existing at-power PRA model, which is currently being updated. The final version of both models are intended to be incorporated into an EOOS model which will provide a consistent method of monitoring risk as plant components/trains are taken out of service from different plant operating states.

The shutdown model is developed for a typical refueling outage and comprise 10 distinct plant operational states. Three undesirable end states are defined:

- boiling
- return to criticality
- core damage

The shutdown model has been already generated and placed in EOOS format. The major insight from the preliminary results is that the plant is most vulnerable to events during reduced inventory (mid-loop) conditions. The goal is to develop the EOOS model with the at-power model being one plant state and the shutdown model comprising the other plant states for risk comparison and maintenance or outage optimization.

Currently, the models are being reviewed.

DETERMINISTIC ANALYSES FOR THE WESTINGHOUSE OWNERS GROUP AND INDIVIDUAL UTILITIES

Westinghouse has used WGOTHIC and other calculation tools to analyze various loss of RHR cooling scenarios during shutdown.

- WCAP-11916, 7/1988, "Loss of RHRS Cooling While the RCS is Partially Filled."
- Abnormal Response Guideline ARG-1, Rev. 1, 6/6/1996 "Loss of RHR While Operating at Mid-Loop Conditions,"

WCAP-11916, ARG-1, and plant specific calculations were performed to determine times to boiling, times to core uncovery, and for provide procedure guidance to address Generic Letters 87-12 and subsequently 88-17, "Loss of Decay Heat Removal"

- Abnormal Response Guideline ARG-2, Rev. 1, 9/30/1997 "Shutdown LOCA"

ARG-2 was issued to provide WOG utility members with procedure guidance for Mode 3 / 4 LOCA when some of the safeguards systems may be removed from service

- WCAP-15145, 2/1999, "Development and Testing of Generic Plant Models with the GOTHIC Computer Code for Analyses to Support Shutdown Operations"
- WCAP-14988, 4/1998, "Use of the GOTHIC Computer Code for Analyses to Support Shutdown Operations"
- WCAP-14089, Rev. 1, 1994, "Analyses to Develop a Basis for Surge Line Flooding Response to Support Shutdown Operations"

WCAP-14988 documents the generic models; WCAP-15145 updates the RHR, RCP, SG, and thermal conductor models and uses the latest version of GOTHIC (6.1P versus 5.0e)

WCAP-14988 and WCAP-14089, Rev. 1 models have been used to determine limiting pressures on temporary seals in the RCS, such as Steam Generator Nozzle Dams

Thus, more deterministic analysis capability is made available to support success criteria for shutdown risk models, and also to support outage optimization.

WESTINGHOUSE PWR PSA SURVEY RESULTS

FROM WOG RBT PRA SURVEY:

SHUTDOWN MODEL ?

NONE	11
ORAM	12
PRA MODEL	6
ORAM/PRA COMBINED	3
ORAM/SENTINEL	2
CAFTA/EOOS	2
Total	36

The utilities are already taking action, in different ways.

SOME ADDITIONAL INSIGHTS

During ORAM, an extensive set of deterministic analyses needed to be made. At shutdown, response to initiating events requires substantially more manual actions than during power operation. Thus, the time windows for operators to detect, diagnose, and act with an event become important and need to be determined by thermal-hydraulic analyses.

The time to boiling margin is an important parameter in determining periods of high vulnerability.

Plant shutdown risk is dominated by a few periods of high vulnerability. Risk management actions during these periods may be identified and implemented. Duration of these time periods may be minimized.

The twelve outage risk reduction improvements at Diablo Canyon did not lengthen the outage.

Postulated inadvertent losses of coolant while in Modes 5 and 6 (when the cavity is not flooded) dominate shutdown risk.

Some plants have taken to off-loading the entire core when any planned maintenance involving the RCS is scheduled to reduce perceived risk.

There is no doubt that shutdown risk assessment of some form has proven to be of practical value in understanding and reducing plant risk.

What then: see the questions next!

QUESTIONS

Can different ways of modeling shutdown risk assessment co-exist? Should they be encouraged to co-exist?

Should calculation of numerical goals (CDF, LERF) be required and compared with at-power values) ?

What is the scope ? (fire during shutdown; flooding during shutdown; seismic events during shutdown?)

Can one be going into a deeper abyss in human error modeling and calculations by getting into numerical measures in shutdown? Will this lead to prescriptive recovery procedures (EOPs and SAMG)?

How can one consolidate risk-informed applications requirements with the shutdown risk model and measure requirements?

CONCLUSIONS

Two important conclusions emerge:

1. Utilities already recognize the value of shutdown risk assessment and address in different ways that are most suitable for their needs.

2. Valuable and practical risk insights can be obtained for shutdown operations using different methods.

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Shutdown Risk Monitoring

Presented by

Jeffrey A. Julius

Thomas A. Morgan

April 1999

SCIENTECH, Inc.

Collective Response to NRC Questions

- This response of the Safety Monitor Users Group represents the input of the 18 US plants that are members
- Cross-section of the US nuclear generating facilities.
- Includes 15 PWR, 3 BWR of varying plant type and vendor.

1

Safety Monitor™ Users Group Members

- San Onofre Units 2 & 3
- Comanche Peak 1 & 2
- Wolf Creek
- Callaway
- Indian Point 2
- Perry
- Surry Units 1 & 2
- Nine Mile Point 1 & 2
- North Anna Units 1 & 2
- Beaver Valley 1 & 2
- Point Beach Units 1 & 2

Shutdown PRA Experience

- In the United States:
  - San Onofre used Shutdown PRA since early '90s.
  - Ten PWR models built or in progress.
  - Two BWR models built or in progress.
- Internationally:
  - Borssele Shutdown PRA:
    - Internal and External events, Level 1 through 3.
    - Human Errors of Commission for Power and Shutdown.
  - IAEA Shutdown Methods development participation, 1992 & 1994.
    - IAEA Guidelines for Shutdown Risk Assessment.

2

Shutdown PRA Evolution

- Full scope international study & San Onofre models developed.
  - International study completed IAEA Peer Review.
- EPRI tailored collaboration project developed focused scope shutdown template based on.
  - International study & IAEA Shutdown Guidelines.
  - San Onofre Shutdown PRA.
  - Surry and Grand Gulf NUREGs.
- Independently reviewed.

Shutdown PRA Philosophy

- *OPTIONAL* Application, Used by plants to better manage risk during outages.
- Supplements Defense-In-Depth concepts of NUMARC 91-06.
- Provide additional insights regarding:
  - Alignments and Components.
  - Contingencies or Functional Alternatives.
- May Support Current Licensing Basis changes.
  - e.g. San Onofre DG Allowed Outage Times.

3

Shutdown PRA Model Scope

- Models All Modes, All Outage Types:
  - Reactor Coolant System & Fuel Pool.
  - Continuous timeline through all shutdown states.
- Endpoints:
  - Core Damage for All plant states.
  - Boiling for Cold Shutdown Modes.
- Dependencies: Functional, Human, & Time.
- Component Level of Detail.
- Consistent with Full Power PRA.
- An Integrated Model.

Shutdown PRA Model Design

- Boundary Conditions and Assumptions are Important to Results.
- Typical PRA Quantification Process Followed.
- Integrated Model Concept Employed:
  - One set of Fault Trees for Full Power and Shutdown
  - Three sets of PWR Event Trees - Power, RHR, Fuel Pool
- Shutdown-specific Data primarily Initiator and HEPs.
- Quantification conducted in a Top Logic Model.

4

Shutdown PRA vs. PRA Model

Shutdown PRA

- "Backward"-looking.
- IPE-like:
  - Average configuration.
- Data developed by PRA:
  - Duration of states.
  - Time since shutdown.
  - Test and Maintenance.
- More expensive, wide variations in data.

Shutdown PRA Model

- "Forward"-looking tool.
- Config. Risk Management:
  - Specific outage configs.
- Data provided by Schedule:
  - Duration of states.
  - Time since shutdown.
  - Test and Maintenance.
- Minimize cost, better insights to outage managers.

Summary Response to Questions

- Shutdown CDF:
  - Less, but comparable to, Full Power PRA.
  - Comparability depends on:
    - Consistency in methods, level of detail, & dependencies.
    - Dimensions (per year vs. per hour or per POS).
  - Instantaneous Risk may by higher, but for short durations.

5

Summary Response Cont'd

- Shutdown LERF:
  - Idea of surrogate Level 3 measure applies during shutdown.
  - May require re-visiting the definition of LERF.
  - Better to monitor Containment status than model.
- Shutdown Standard:
  - Develop after Full Power Standard benefits are realized.
- NUREG-1602:
  - "Cadillac" method, heavy on data development.

Summary Response Cont'd

- Human Reliability Analysis:
  - Full Power HRA methods apply.
  - Methods have difficulty with very long time windows (beyond 24 hours) with 2 ways to treat.
    1) Apply additional recovery beyond "floor" limits.
    2) Truncate sequences rather than defend very low HEPs.
  - Errors of Commission:
    - Same treatment as in Full Power.
    - Primarily included in Initiating Events.

6

Overall Conclusions

- Defense in Depth approach of NUMARC 91-06 provides sufficient safety margin for the current plants.
- Shutdown PRA should remain as an optional tool:
  - For outage risk management.
  - If desired to support Risk-Informed Regulatory submittals.

Application of Shutdown Models - Practical Examples

- Let's look at how an existing shutdown model is being used, in conjunction with existing Defense in Depth methods.

7

Shutdown PRA Experience at One Users Group Plant

- Shutdown PRA Used Since Early 1990's
  - PRA Results *supplement Defense-in-Depth* methods used by Outage Planners.
- Shutdown Safety Monitor Models developed in the last two years.
  - Used in a similar manner as Shutdown PRA, except:
    - More detailed schedules now analyzed
    - Models include more system alignment selections.

DEFENSE IN DEPTH PLANNING SHEET

SHUTDOWN SAFETY FUNCTIONS	SAFETY FUNCTION FULFILLMENT PLAN		OTHER CONTINGENCY PLAN, [illegible] or COMMENTS
	[illegible]	[illegible]	
Reactivity Control (Core)	[illegible]	[illegible]	[illegible]
Electric Power Availability	[illegible]	[illegible]	[illegible]
Inventory Control (Core)	[illegible]	[illegible]	[illegible]
Decay Heat Removal (Core)	[illegible]	[illegible]	[illegible]
Containment Closure Control	[illegible]	[illegible]	[illegible]
Reactivity Control (SFP)	[illegible]	[illegible]	
Inventory Control (SFP)	[illegible]	[illegible]	
Decay Heat Removal (SFP)	[illegible]	[illegible]	[illegible]

8

Shutdown Safety Monitor Models

- Transition Risk Model (Modes 2-4, SDC).
- Modes 5-6, Fuel Pool:
  - Loss of SDC IE Fault Tree includes loss of support systems
  - Loss of Offsite Power (Plant/grid) & SBO
  - Loss of Inventory
  - Loss of Fuel Pool Cooling IE Fault Tree.
- Models contain similar detail as Level I.

Lessons Learned

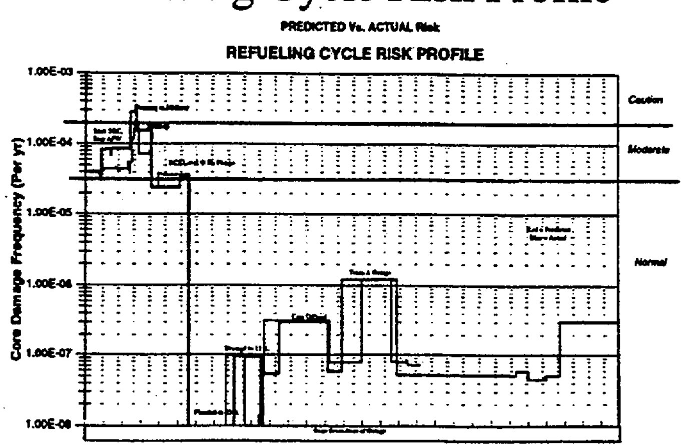
- Outage Risk is on the order of Level I Risk (1E-05/year contribution to cumulative risk)
  - High Risk Evolutions have a higher instantaneous risk than level I, which are offset by low duration.
  - Most of the outage is spent in very low risk configurations.
- Most Equipment OOS occurs during low risk configurations.

9

Lessons Learned (Continued)

- Most of the cumulative risk of shutdown comes from:
  - Low Inventory Configurations
  - Early in Outage
  - Long Duration (more than a few hours) Plant Operational States (POSs).
- Optimizing High Risk configurations is sometimes just optimizing system alignments.

Refueling Cycle Risk Profile



10

Recent Shutdown PRA Uses

- Shutdown Safety Monitor used for recent Risk-Informed IST Project.
  - Developed an "Average" Component Importance based upon a typical schedule.
- DG Allowed Outage Time Tech Spec Change
  - Compared risk of DG Outage at Full Power vs. Shutdown

Insights from Practical Applications

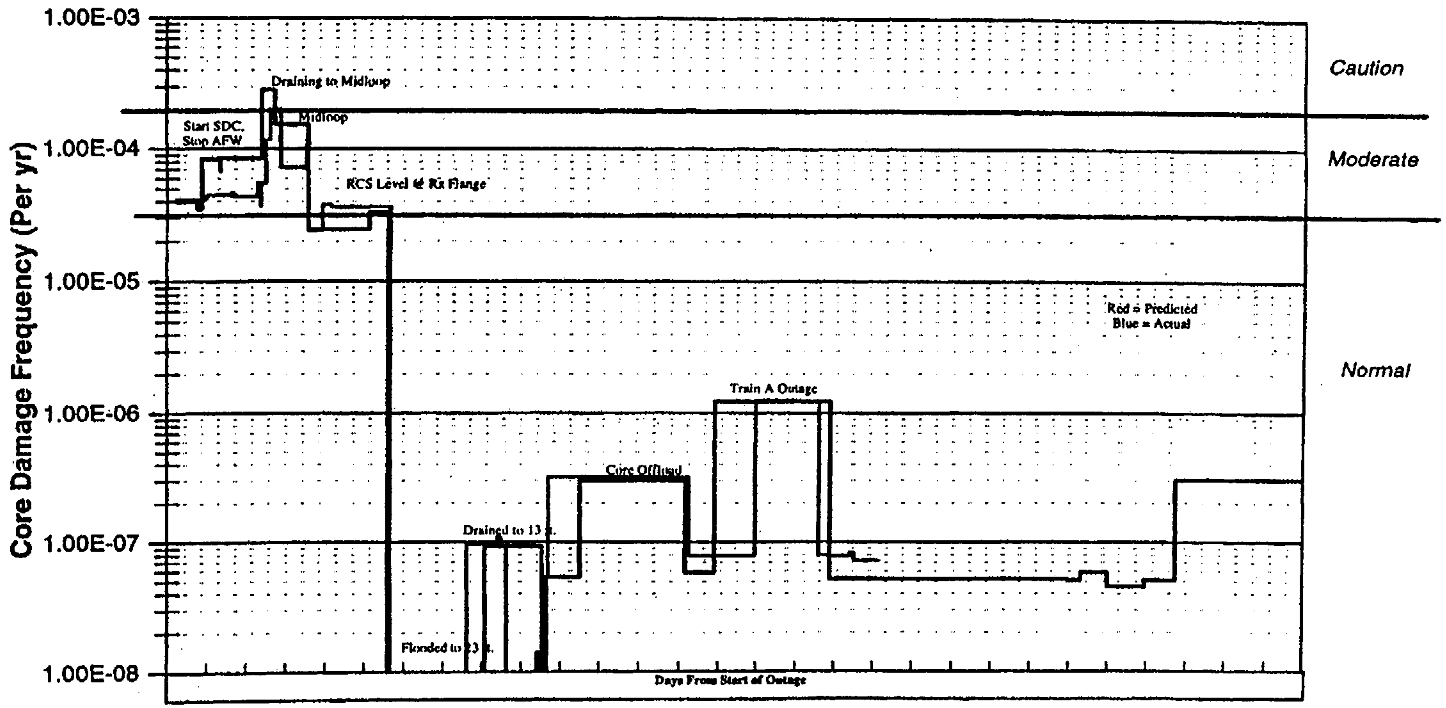
- Any shutdown PRA efforts should concentrate on High Risk POSs.
- Equipment Availability, risk optimization, etc. may not be important if the plant does not perform major equipment OOS during High Risk Evolutions.

11

Refueling Cycle Risk Profile

PREDICTED Vs. ACTUAL Risk

REFUELING CYCLE RISK PROFILE



DEFENSE IN DEPTH PLANNING SHEET

Changes from previous sheet in *Bold Italic*

Schedule Rev: 0 Outage Milestone Period: From Commence Drain to Midloop *To Pressurizer Manway Installed*

SHUTDOWN SAFETY FUNCTIONS	[1]SAFETY FUNCTION FULFILLMENT PLAN		OTHER CONTINGENCY PLAN, [2]SFPP or COMMENTS
	METHOD A (Verified Shiftly)	METHOD B (Verified Shiftly)	
Reactivity Control (Core)	Train A Boron dilution monitoring and alarm instrumentation; Verify SDM each day, and sample RCS boron every 24 hrs. Boric Acid Flowpath: RWST to a Train A HPSI Pump, to an RCS loop with SDC flow.	Train B Boron dilution monitoring and alarm instrumentation; Verify SDM each day, and sample RCS boron every 24 hrs. Boric Acid Flowpath: RWST to a Train B HPSI Pump, to an RCS loop with SDC flow.	***SFPP:*** ***Isol. of Boron Dilution Flowpaths.*** ***PRA Recommendation for RIC: Maintain all Charging Pumps available.***
Electric Power Availability	Annunciators for Train A electrical AC/DC Buses, transformers and switchyard; Daily control board monitoring and weekly surveillance. Train A 1E buses energized, with required AC available from offsite sources and the SAME Units Diesel.	Annunciators for Train B electrical AC/DC Buses, transformers and switchyard; Daily control board monitoring and weekly surveillance. Train B 1E buses energized, with required AC available from offsite sources and the SAME Units Diesel.	***SFPP:*** ***Control of AC Power Sources*** ***Res Aux Xfmrs supplying both 1E busses.***
Inventory Control (Core)	RWLI, LT1520, with a Hi/Lo alarm. Makeup Path: RWST to a Train A HPSI Pump. ***Required during RIC: miniflow valves 9306 AND 9307, Hot Leg Inj valve HV9420 AND any 2 Train A Cold Leg injection valves.***	DLMS with an alarm. Makeup Path: RWST to a Train B HPSI Pump. ***Required during RIC: miniflow valves 9347 AND 9348, Hot Leg Inj valve HV9434 AND any 2 Train B Cold Leg injection valves.***	***HJTC #8 (21"NR) alarm set.*** ***(During SLF conditions only use HJTC for level indication.)***
Decay Heat Removal (Core)	CET with readout and Hi alarm in the Control Room; SDC heat exchanger Train A inlet/outlet temperature and SDC flow indication in the Control Room with Hi/Lo alarm. SDC Train A.	An alternate CET with readout and Hi alarm in the Control Room; SDC heat exchanger Train B inlet/outlet temperature and SDC flow indication in the Control Room with Hi/Lo alarm. SDC Train B.	***SFPP:*** ***RCS Perturbation Control.*** ***PRA Recommendation for RIC: Maintain ability to align CS Pp to SDC within 40min; 2nd Train of CCW/SWC in operation.***
Containment Closure Control	Containment Evacuation Siren System. Close Containment openings or open penetrations by use of AC or DC power which is available during a loss of offsite power or by manual actions.	Communication link established with HP, Refueling Group or other work groups as appropriate for closure. Close Containment openings or open penetrations by manual actions alone.	Equipmnt Hatch power from B04 ***SFPP:: Cntmnt Closure Control: use RIC sheets during RIC.*** Two ECUs available during RIC.
Reactivity Control (SFP)	SFP Level Hi Alarm. Boric Acid Flowpath: RWST to SFP Makeup Pump P011 (A03/BF), to the SFP.	Weekly sampling of SFP boron. Boric Acid Flowpath: RWST to the suction of the SFP Cooling Pump P009 or P010, to the SFP.	
Inventory Control (SFP)	SFP Level Hi/Lo alarm in the Control Room. Makeup Path: RWST to SFP Makeup Pump P011 (A03/BF), to the SFP.	Daily local monitoring of SFP level and liner leakage. Makeup Path: RWST, direct to the suction of the SFP Cooling Pump P009 or P010, through the cooling system to the SFP.	
Decay Heat Removal (SFP)	SFP temperature Hi, Purification temperature Hi (if purification is in service) & SFP pump discharge pressure Lo or Overcurrent alarms in the Control Room. Train A SFP Cooling Pump P009, a heat exchanger and necessary CCW/SWC support.	Shiftly local monitoring of SFP temperature, pump and heat exchanger performance. Train B SFP Cooling Pump P010, a heat exchanger and necessary CCW/SWC support.	SFP Heatup Rate <3F/Hr. Intake is dewatered.

[1] A SFFP method consists of a "monitoring and detection" item (numerical) in addition to a compatible "control" item (alphabetical) from the methods menu.
[2] A SFPP is a means of protecting or minimizing the likelihood of losing a Shutdown Safety Function. Used during a HRE.

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Shutdown PSA and EOOS

Loys K. Bedell
Entergy - River Bend Station

Purpose of RBS Shutdown PSA

- Common Risk Assessment Tool for At-Power and Shutdown Operations
- Component Level Model More Flexible for Shorter Outages
- Suppression Pool Cooling & Cleanup / ADHR System Added

Shutdown PSA End States

■ End States

- RCS Boiling
- Core Damage (includes SFP Damage)
- Fuel Pool Boiling

■ Other NSAC-175L End States

- LTOP - RBS LOCA Initiator
- Promt Criticality - Maintain SDM
- Exposed Bundles - OPDRV Initiator /IFTS
- Containment Performance (Shutdown Level 2)

Shutdown PSA Challenges

- Initiating Events
- Success Criteria Changes
- Human Reliability Analysis
- Recovery Actions
- Defense-in Depth Modelling
- EOOS Development

Phased Mission Model

- **Plant Configuration Changes**
  - Systems Running
  - Systems Out of Service
  - Success Criteria Changes
- **Plant State Changes**
  - Decay Heat Level
  - RPV Water Level
  - Containment Status
- **RBS had ~65 Flags to Handle ~62 Plant Configurations**

Fault Tree Changes

- Five New Fault Trees Created
- Six Existing Fault Trees Changed
- Separated Demand and Run Failures in Certain Fault Trees to Account for Changes in System Status

Shutdown PSA Quantification

- *No Baseline CDF or Boiling Risk*
- *Quantification Done for All Combinations of Flag Settings*
- *Sequence Quantification Done for Model Testing and Enhancement*
- *Schedule Quantification Done Through EOOS*

Human Reliability Analysis

- *Procedure Applicability*
- *Limited Procedural Guidance*
- *Indications Available*
- *More Time Available (and Less Stress)*
- *Applicable HRA Methodologies*

Operator Recovery Actions

- Recovery of Off-site Power
- Recovery of Decay Heat Removal
- Recovery of Spent Fuel Pool Cooling
- Recovery from OPDRV/OPDRC
- Recovery Data from NSAC Documents

Operator Recovery Times

- **Decay Heat Level**
  - *High Decay Heat (Days 1-4)*
  - *Medium Decay Heat (Days 5-18)*
  - *Low Decay Heat (After Day 18)*
- **RPV Level**
  - *Normal RPV Level*
  - *RPV Level > 23 feet above Flange*

Defense-In Depth Logic

- SSFAT Logic not Well Documented
- Force Color Code with Fault Tree
- Develop Consistent Color Codes Based on Technical Specifications
  - Green - Exceed LCO Requirements
  - Yellow - Meet LCO Requirements
  - Orange - In LCO Action Statement
  - Red - Tech. Spec. Violation

RCS Boiling Results

- *RCS Boiling Frequency is High at the Beginning of the Outage (0.72/yr)*
- *RCS Boiling Frequency is High during RPV Hydrostatic Testing (0.7/yr)*
- *High RCS Boiling Frequency Does Not Imply High Core Damage Frequency*

Core Damage Results

- *Core Damage Frequency Driven by Support System Maintenance*
- *Highest for Maintenance not Allowed At-Power (DC Power, Off-site AC)*
- *Core Damage Frequency Cannot be Directly Tied to Any Defense-in-Depth Status Measure*

Spent Fuel Pool Boiling

- *Fuel Pool Boiling is Very Low Frequency Event (10^{-9}/Yr)*
- *Fuel Pool Boiling Risk Negligible Before Fuel Movement*
- *Not a Dominant Contributor to Fuel Damage (Except for Full Core Offload)*

Shutdown PSA Limitations

- Difficult to Perform Sensitivity and Uncertainty
- No Overall Importance Ranking
- Must Check Alignment before Performing SHEOOS Run
- No Simple Results

Results and Conclusions

- Shutdown PSA is Viable, but much more Dynamic than At-Power PSA
- No Baseline Risk Number
- Shutdown Risk Driven By Schedule
- Human Reliability Analysis and Operator Recovery Important
- Defense-In Depth Does Not Imply Low Shutdown Risk

Results and Conclusions

- *Shutdown Risk Comparable to At-Power Risk*
- *Cumulative Risk for 21 Day Outage Could Be as High as Yearly At-Power Risk*
- *Limitations to Short Outages without Impacting Outage Risk*
- *Can Be Physical Limitations for Short Outages*

Results and Conclusions

- Shorter Outage = Higher Average Risk, but possibly Lower Overall Outage Risk.
- Can Determine the Impact of Moving Activities from Outage to At-Power
- Can Reduce Overall Risk By Doing More On-Line Maintenance

PERSPECTIVE ON SHUTDOWN ISSUES AT STP

Presented to the Use of Low Power and Shutdown Risk in Regulatory Activities Public Workshop

April 27, 1998

Steve Rosen, Department Manager Risk Management and Industry Relations

TOOLS IN USE AT STP

- **ORAM/Sentinel Shutdown Model including Shutdown Safety Functions and Shutdown Probabilistic Safety Assessment**
- **Shutdown PRA Using RISKMAN**
- **Shutdown Risk Assessment Group and Shutdown Risk Assessment Procedure**

SHUTDOWN RISK ASSESSMENT GROUP

- **Members Include:**
  - **Operations Manager**
  - **Shift Technical Advisor**
  - **Risk and Reliability Analysis Member**
  - **Nuclear Assurance**
  - **Nuclear Licensing**
  - **Outage Representative**

SHUTDOWN RISK ASSESSMENT GROUP
(continued)

- **Duties Include**
  - **Review Level 2 Outage Schedule**
  - **Prepare Report for Outage Support Manager and Plant Manager**
  - **Report Addresses Shutdown Safety Issues - Mid-Loop, RCS Pressurization, Loss of Inventory, Loss of Cooling, Loss of Power, Containment Integrity, etc.**

SHUTDOWN RISK ASSESSMENT

- **Example of Compensatory Actions (Mid-Loop)**
  - **On-Site, Switchyard, etc. Electrical Work Minimized**
  - **RCB Containment Integrity Maintained During Mid-Loop**
  - **RHR Trains "Protected"**
  - **Extra Personnel Assigned**

SHUTDOWN RISK PERSPECTIVES

- **Risk At-Power and Risk During Refueling are Comparable (Same Order of Magnitude)**

- **Front-End Mid-Loop Contributes Approximately 15% of the Risk During Shutdown in 1% of the Total Refueling Hours**

CONTAINMENT ISSUES

- **Issue Raised Prior to Outage Concerning Early Mid-Loop, Reduced Inventory Operations, and Containment Status**
- **Containment Is Closed Prior to Entry into Reduced Inventory**
- **Training on Closure of Containment Equipment Hatch Performed**

REVIEW OF PAST OUTAGES

- **The Risk per Hour for the Current Outage is Lower than the Risk from the Previous Two Outages. [Longer Length, No-Mode Longer]**

- **The Cumulative Core Damage Risk and Boiling Risk Are Comparable**

COMPARISON

Outage Number	Duration (hours)	CD Risk (per hour)	CD Risk (Cumulative)	Boiling Risk (per hour)	Boiling Risk (Cumulative)
*1RE08	*667	*6.3E-08	* 4.2E-05	* 4.0E-06	*2.7E-03
2RE06	464	8.7E-08	4.0E-05	5.9E-06	2.7E-03
1RE07	482	8.2E-08	4.0E-05	7.0E-06	3.4E-03

- Note Results Based on ORAM Calculation

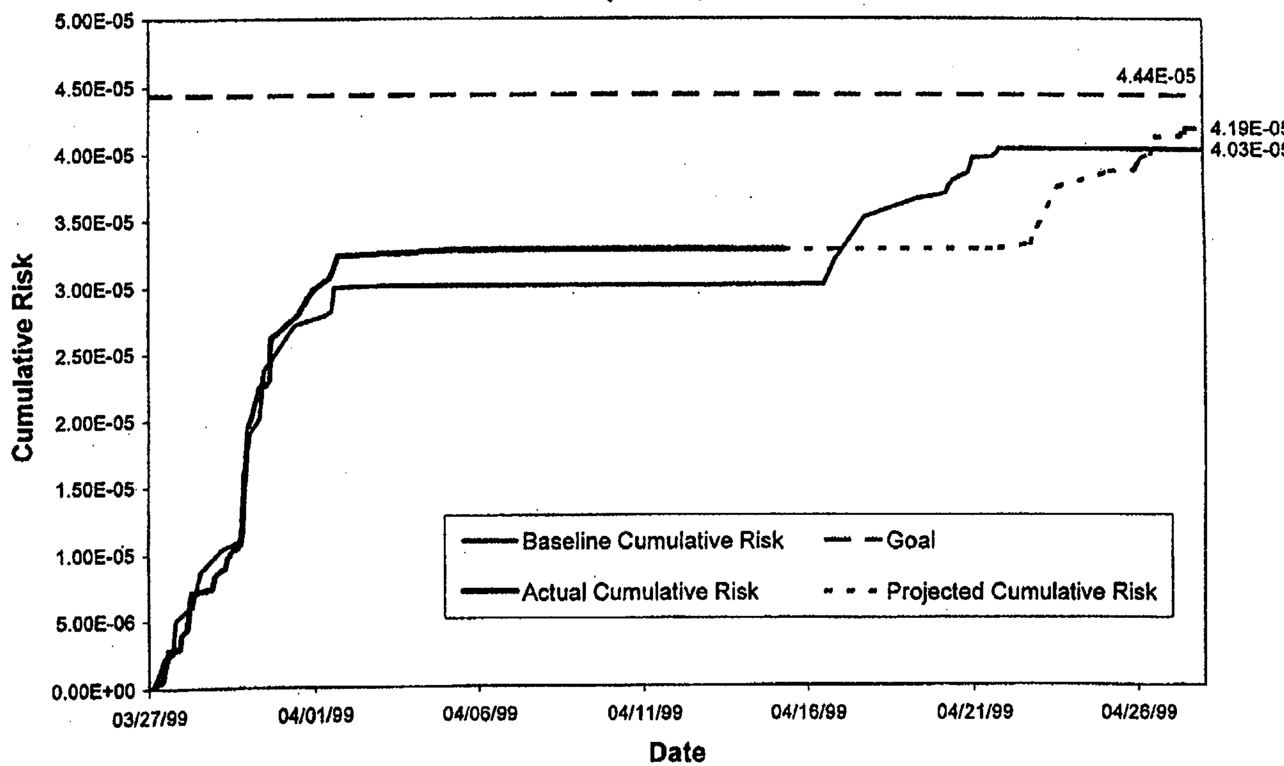
  * Estimated value prior to start of the current outage

CONCLUSIONS FROM COMPARISON

- **The Risk for the Current Refueling Outage is Comparable to the Risk Seen in Previous Outages**

- **Compensatory Measures (Including Mid-Loop Precautions] are Adequate to Protect the Health and Safety of the Public**

1RE08 Cumulative Core Damage Probability as of April 15, 1999



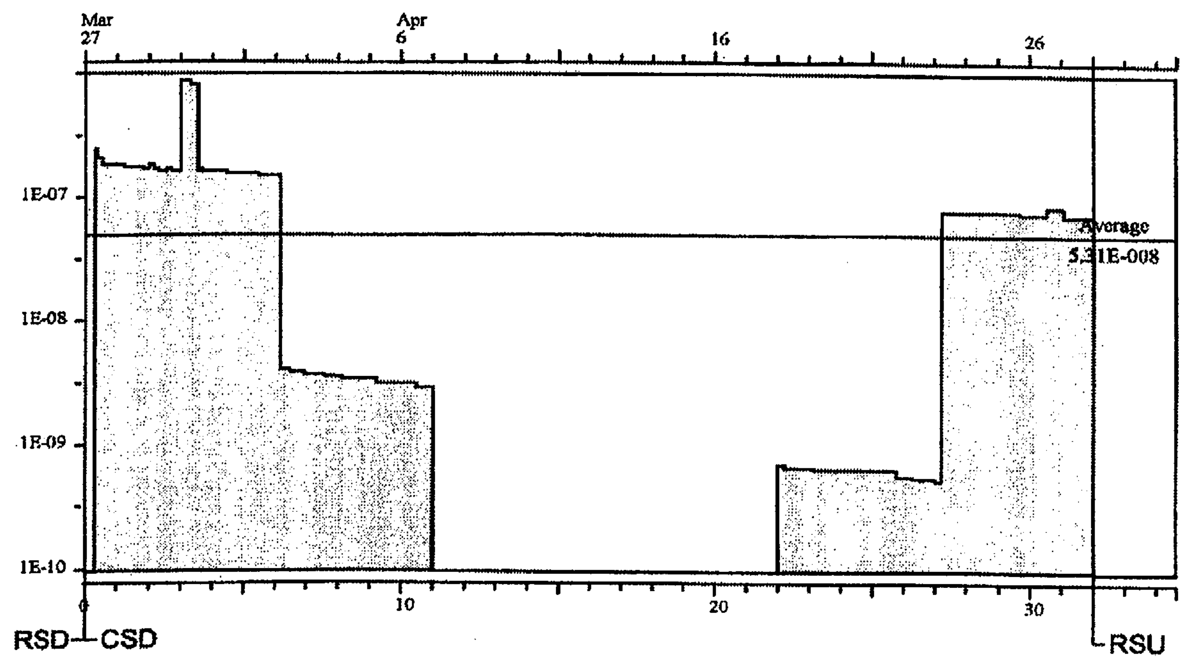
Date: 04/21/99 14:28

SOUTH TEXAS PROJECT
OUTAGE RISK ASSESSMENT AND MANAGEMENT
Core Damage Risk Profile

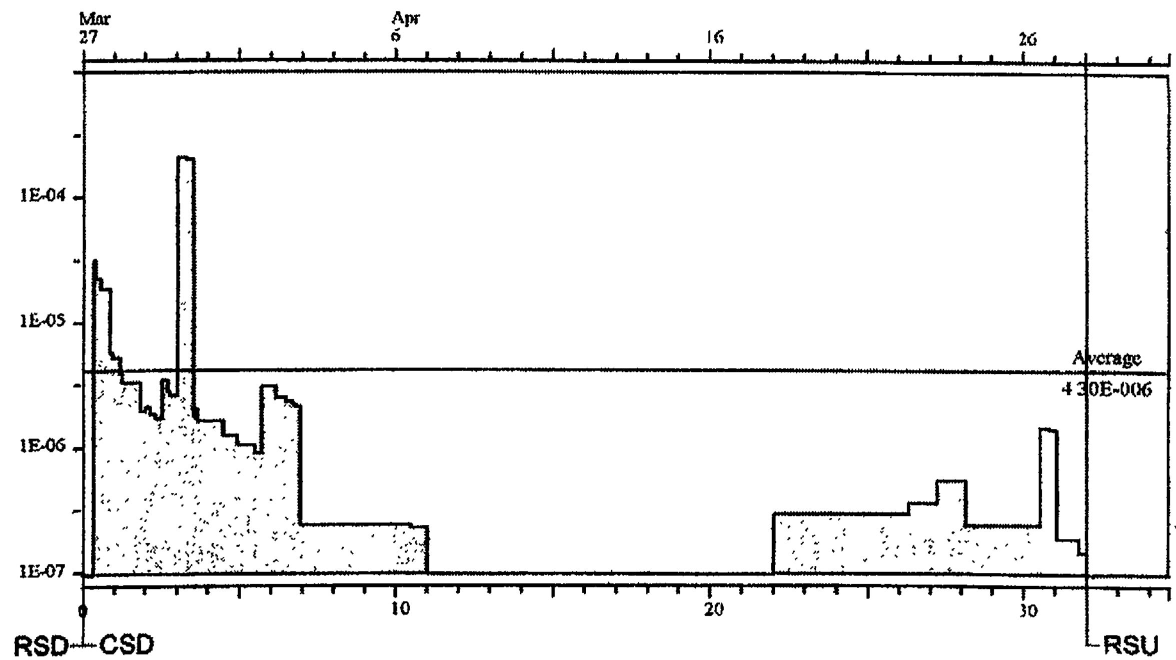
Report Page: 001

Outage: 1RE08 : MODEL FOR SOUTH TEXAS PROJECT

Model: STP1RE8G : 1RE08 4/20/99 (SFP Boil Xfr Ga



ORAM Version 2.10


Mar
27
Apr
6
16
26
1E-04
1E-05
1E-06
1E-07
Average
4.30E-006
0
10
20
30
RSD
CSD
RSU

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Shutdown Risk Assessment at Seabrook Station

April 1999

Kenneth L. Kiper
Seabrook Station
North Atlantic Energy Service Co.
Seabrook, NH

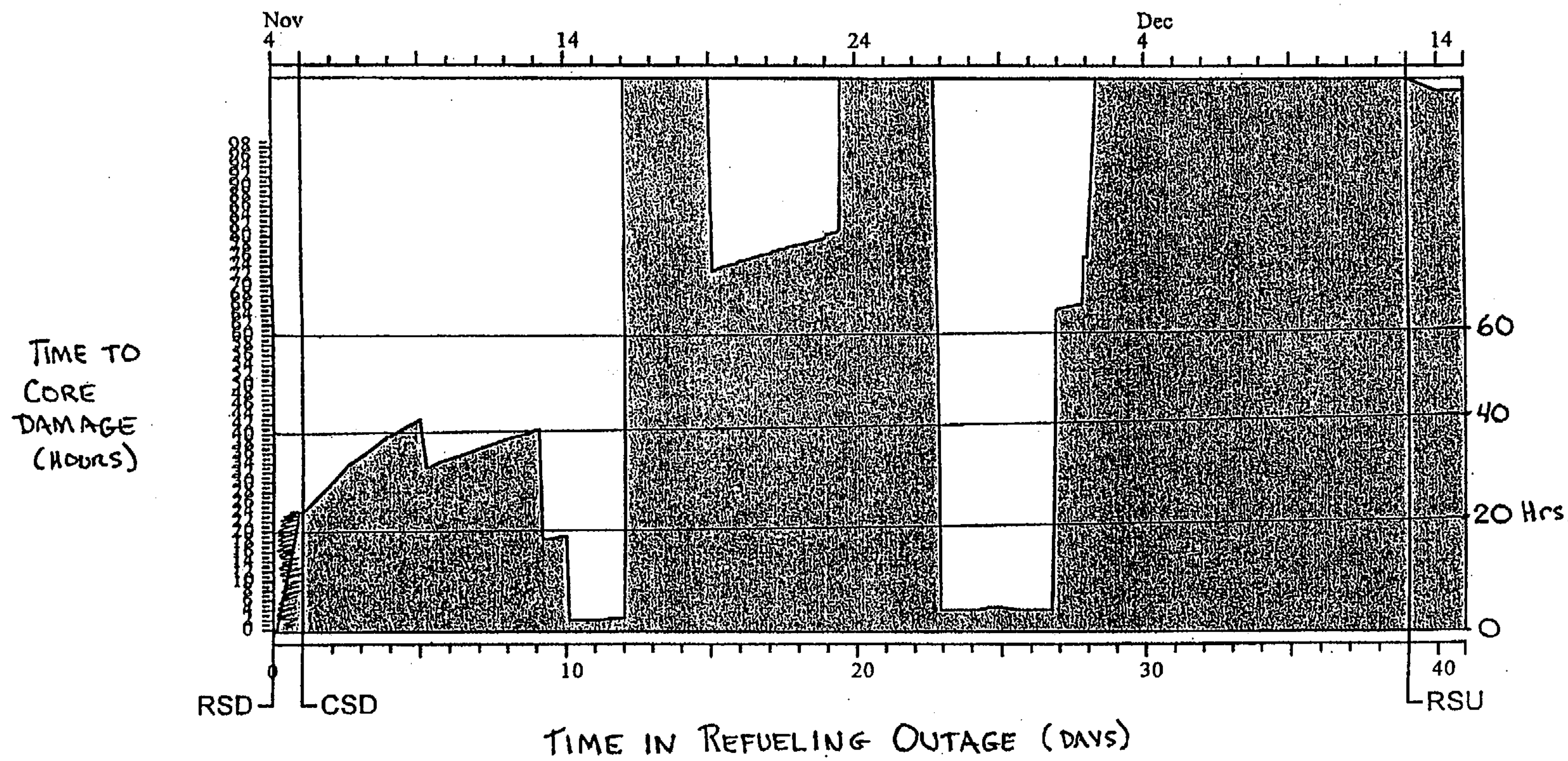
Date: 10/12/95 10:58

SEABROOK STATION
OUTAGE RISK MANAGEMENT AND ASSESSMENT
Time to Core Damage

Report Page: 001

Outage: DRAFT : SB OR04 PRODUCTION MODEL

Model: OR04-3C : Draft 10/11/95 - Imp matrix ch



ORAM Version 2.00

Outline

- ✓ Seabrook Shutdown PRA
- ✓ Shutdown PRA Insights
- ✓ Outage Risk Management Insights
- ✓ Shutdown Risk Standard

KLK 4/19/99 Slide #2

Seabrook Shutdown PRA (1)

- ✓ WHEN: Completed 1988
- ✓ WHAT: Scope
  - Modes 4, 5, & 6 - hot shutdown, cold shutdown, and refueling.
  - Internal and external events.
  - Plant specific design and operation.
  - Level 3 - plant, containment, and site analysis.
  - Uncertainty analysis - plant configuration, time after shutdown, operator action, source term, etc.

KLK 4/19/99 Slide #3

Seabrook Shutdown PRA (2)

✓ HOW: Methodology
- Initiating events - internal and external events and "procedure" event trees
- Event trees - Support system tree, Transient tree, LOCA tree
- Containment - isolation
- Source terms - shutdown specific
- Data - industry shutdown experience

✓ WHY:
- EPZ reduction effort; BNL review

KLK 4/19/99 Slide #4

Seabrook Shutdown PRA (3)

✓ Results
- Average core damage risk
  - Mean CDF in the range of At-Power CDF
  - Uncertainty range (5th to 95th) twice as large as At-Power CDF.
- Average public health risk
  - NO early health effects (fatalities) assuming realistic source term and at least 2-mile evacuation.

KLK 4/19/99 Slide #5

Seabrook Shutdown PRA (4)

✓ Results -
Dominant Contributors to Risk

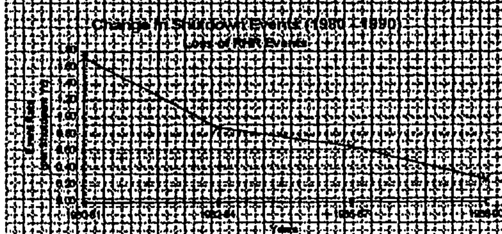
ACCIDENT TYPE	CONTRIBUTION CDF	CONTRIBUTION LRF*
Loss of RHR-RCS "drained"	71%	26%
Loss of RHR - RCS filled	11%	20%
LOCA	18%	54%
Internal Events	88%	82%
External Events (seismic, fire, flood)	12%	18%

* LRF: Large-opening Release Frequency

KLK 4/19/99 Slide #6

Seabrook Shutdown PRA (5)

✓ Conservative estimate of current shutdown risk:
- Shorter outages
- Better planning, procedures, instrumentation
- Fewer precursors



KLK 4/19/99 Slide #7

3

Shutdown PRA Insights (1)

- ✓ Core damage risk - dominated by loss of RHR at low inventory configurations.
- ✓ LOCAs - important contributor to "large opening" release frequency
- ✓ LOCAs - primarily loss of level control / over-draining (and other procedure-driven events), not pipe break
- ✓ External events - floods and fires tend to be more likely to occur, less significant to risk

KLK 4/19/99 Slide #8

Shutdown PRA Insights (2)

- ✓ Shutdown Risk - **difficult to quantify** because of the importance of operator action and the difficulty in correlating time available vs operator reliability.
- ✓ Shutdown Risk - **manageable** because it is driven more by alignments and planned equipment outages - events that can be controlled, planned, and sequenced.

KLK 4/19/99 Slide #9

4

Shutdown PRA Insights (3)

- ✓ LERF – essentially zero at shutdown. This a general conclusion dependent (only) on decay heat and close-in population.
- ✓ Accidents importance to release – those with the equipment hatch off (due to large opening and longer time to restore). NOTE: midloop accidents are not important to release since the equipment hatch is on by requirement.

KLK 4/19/99 Slide # 10

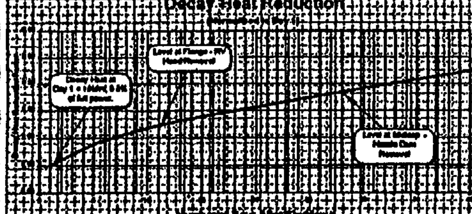
Risk Management Insights (1)

- ✓ Shutdown risk – varies greatly over an outage, from extremely low risk to some of the highest conditional risk periods at power or shutdown.
- ✓ For PWRs, two times of concern:
  - Level at the Flange – low thermal margin
  - Level at Midloop – low thermal margin, low margin to RHR pump cavitation

KLK 4/19/99 Slide # 11

Risk Management Insights (2)

- ✓ Time to Boiling/Core Damage - a functional index of risk.
- ✓ Decay heat reduction with time.



XLK 4/19/99 Slide #12

Standards

- ✓ Details of Outage Risk Management are plant specific but many of the risk assessment conclusions are generic (at least to PWRs).
- ✓ Standards approach
  1. Generically screen out all but "low thermal margin" configurations.
  2. Assure that generic conclusions apply to specific unit.
  3. Apply PRA techniques to remaining (potentially) risk significant configurations.

XLK 4/19/99 Slide #13

6

RISK PERSPECTIVES FROM EPRI RESEARCH & APPLICATIONS

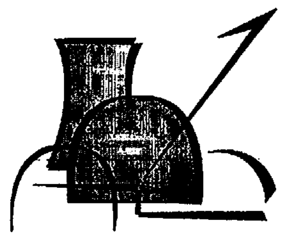
Presented At:

NRC Workshop on LPSD PSA
April 27, 1999

Presented By:

Jeff Mitman - EPRI

Doug True - ERIN Engineering & Research, Inc.



OVERVIEW

- Background
- ORAM Background
- Benchmarking of ORAM PSSA Models
- EPRI Resources
- BWR Shutdown Risk Profiles
- PWR Shutdown Risk Profiles
- General PSSA Insights
- Technology Assessment
- Use of Shutdown PSA Results

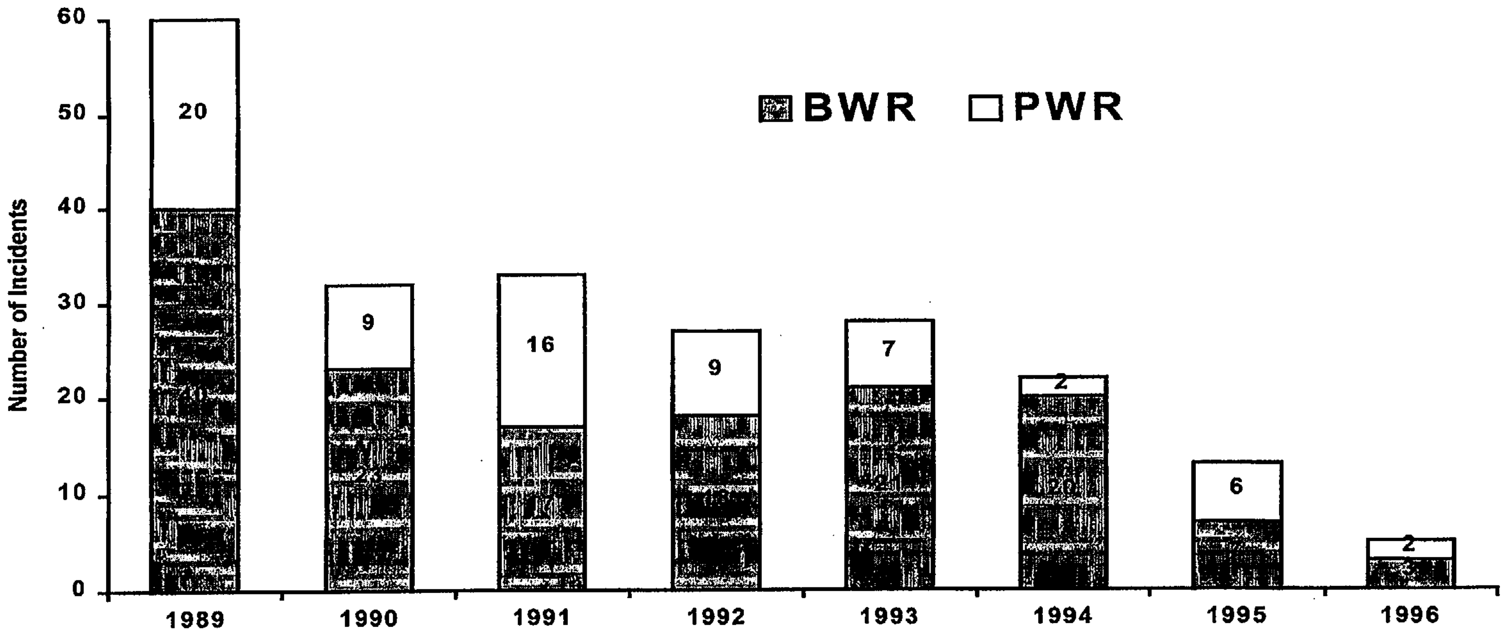
Slide 2

EPRI

BACKGROUND

- NUMARC 91-06 Implemented in 1991
- Trend is Downwards Since then in Significant Shutdown Events
- U.S. Industry has Deployed Multiple CRM Tools to Help Ensure Shutdown Safety
  - EPRI's EOOS™
  - EPRI's ORAM™ family
  - Scientech's Safety Monitor™

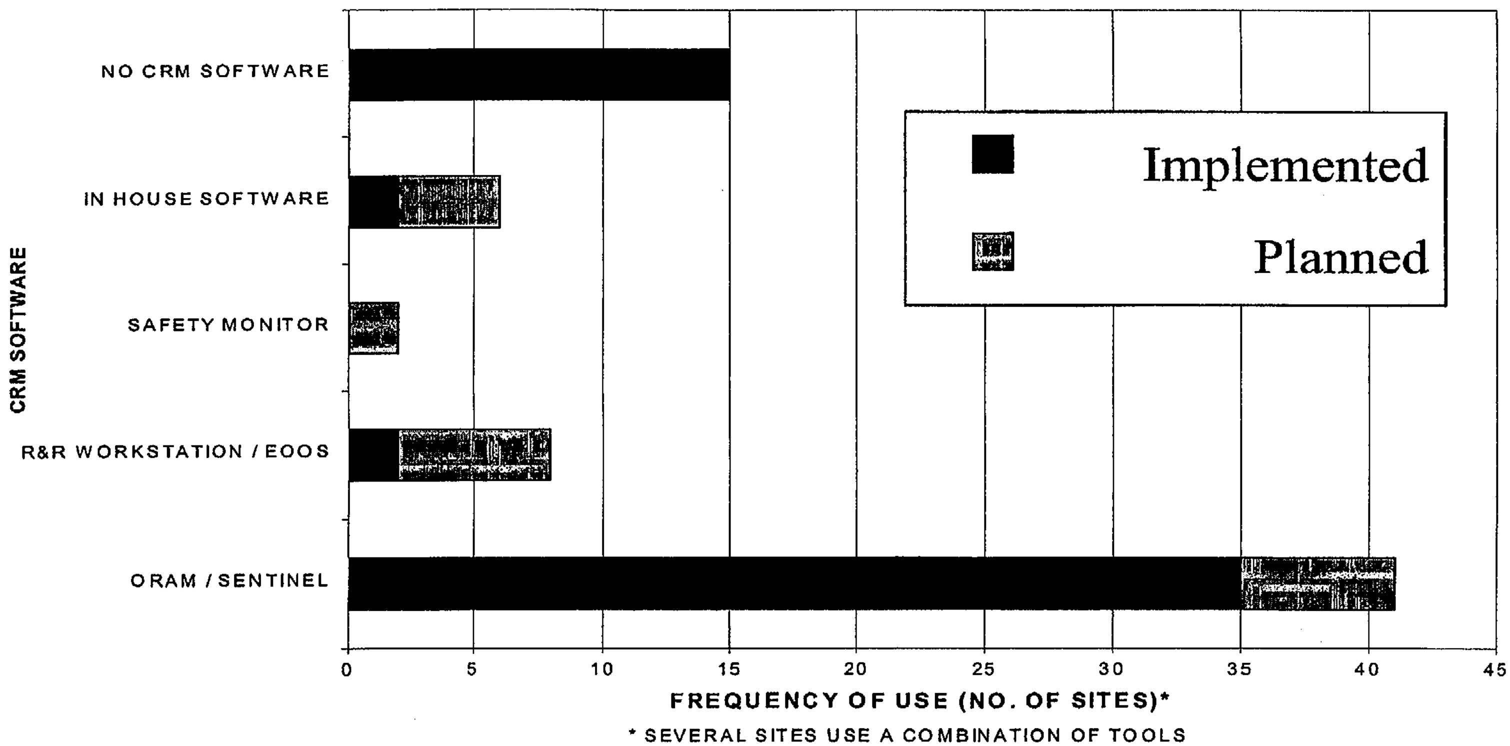
Shutdown Event Trends 1989 -- 1996



Based on EPRI's Report TR-109014

Slide 4

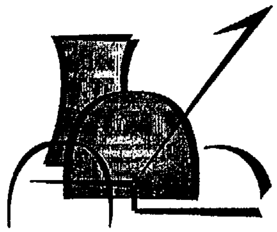
Approximate Number of Sites Using Shutdown CRM Tools in 1996



Based on EPRI Survey in 1996 Include 47 U.S. Sites (TR-102975)

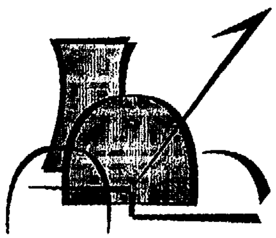
ORAM BACKGROUND

- EPRI's ORAM PSSA Methodology Initiated in 1991
- Over 40 Unit PSSA Models Have Been Developed
- Span More Than 100 Refueling Outages
- Generally Include Both Core Boiling Risk and Core Damage Risk



BENCHMARKING OF ORAM PSSA MODELS

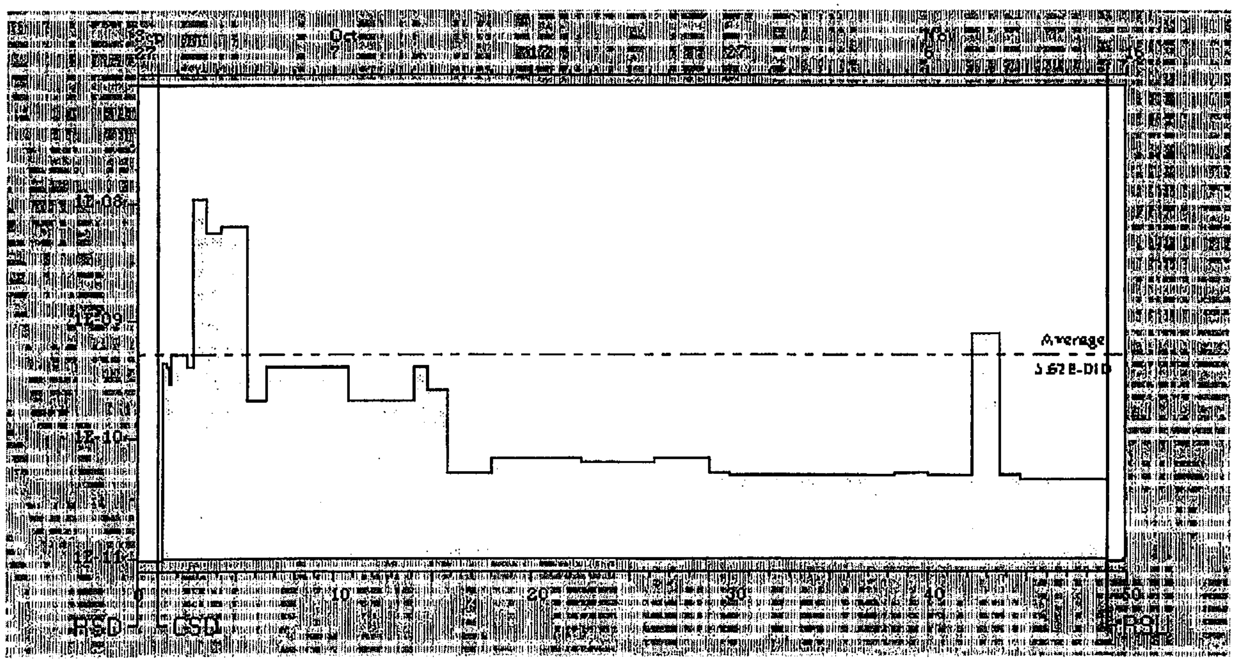
- **Core Boiling Models Correlate Well With Industry Experience of Boiling Events**
- **Developed Both ORAM PSSA and a RISKMAN Shutdown PSA for STPEGS**
  - **Detailed Review of 11 POSs Identified Differences Due to Specific Modeling Assumptions**
  - **Once Assumptions Were Reconciled, PSSA & PSA Provided Comparable Results**



EPRI's OUTAGE SAFETY RESOURCES

- Base PSSA Technology Reports (BWR & PWR)
- Over 20 Reports on Specific ORAM Applications
- Analysis of Loss of Decay Heat Removal Event Trends (TR-109014)
- EPRI's EOOS™ Issued for Use (Enhancements Continue)
- ORAM-SENTINEL v3.3 to be Released Sept. 1999 will Interface with Shutdown PSA
  - ORAM V4.0 Under Development
- Scientech's Safety Monitor™ Issued for Use (Current EPRI TC Applications In Process)

EXAMPLE BWR RISK PROFILE



"Average" CDF = 5.6E-10/hr.*8760 hr./yr. ~4.9E-6/yr.

Slide 9

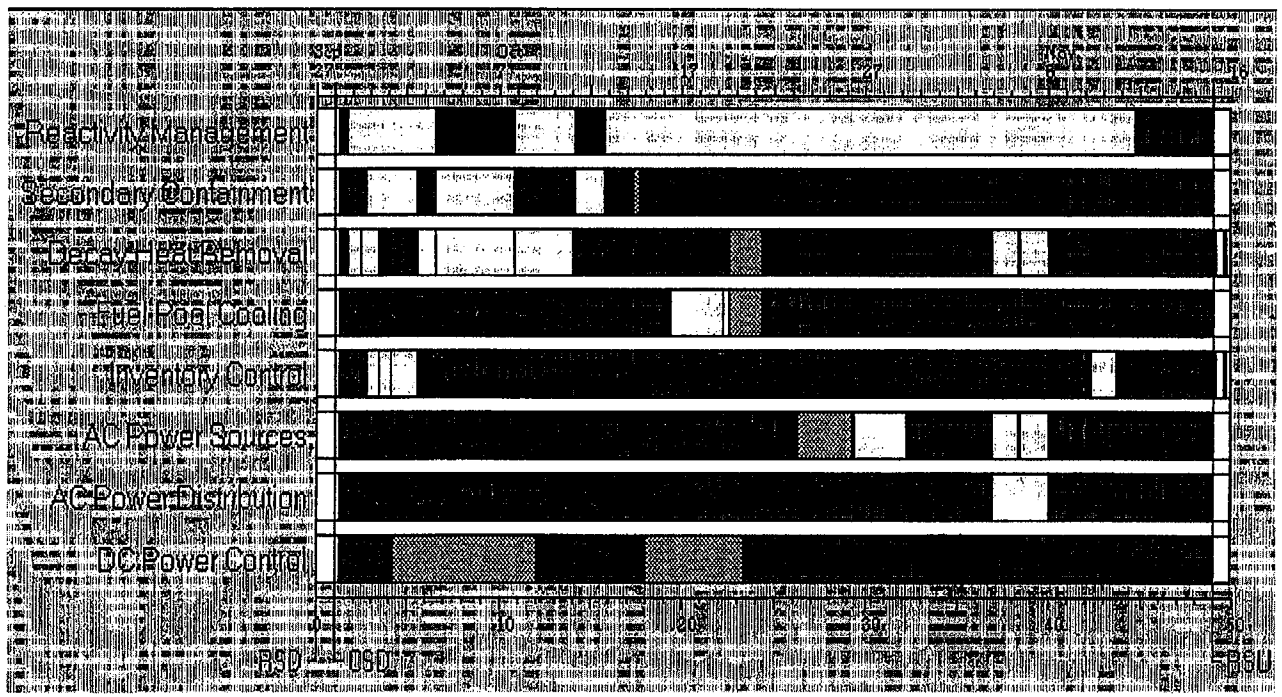
ANALYSIS of EXAMPLE BWR TYPICAL OUTAGE

- Average Configuration CDF (48 day outage) ~4.9E-6/yr.
- Peak CDF ~6.1E-5/yr.
- Min CDF ~4.4E-7/yr.
- CDF Max/Min Ratio ~140
- Contribution to Annual Average CDF (4.9E-6*48/365) ~6.5E-7/yr.
- Contribution of Peaks to AA CDF (5 of 48 days) ~5.5E-7/yr.
- Contribution From Peaks ~86%

Slide 10

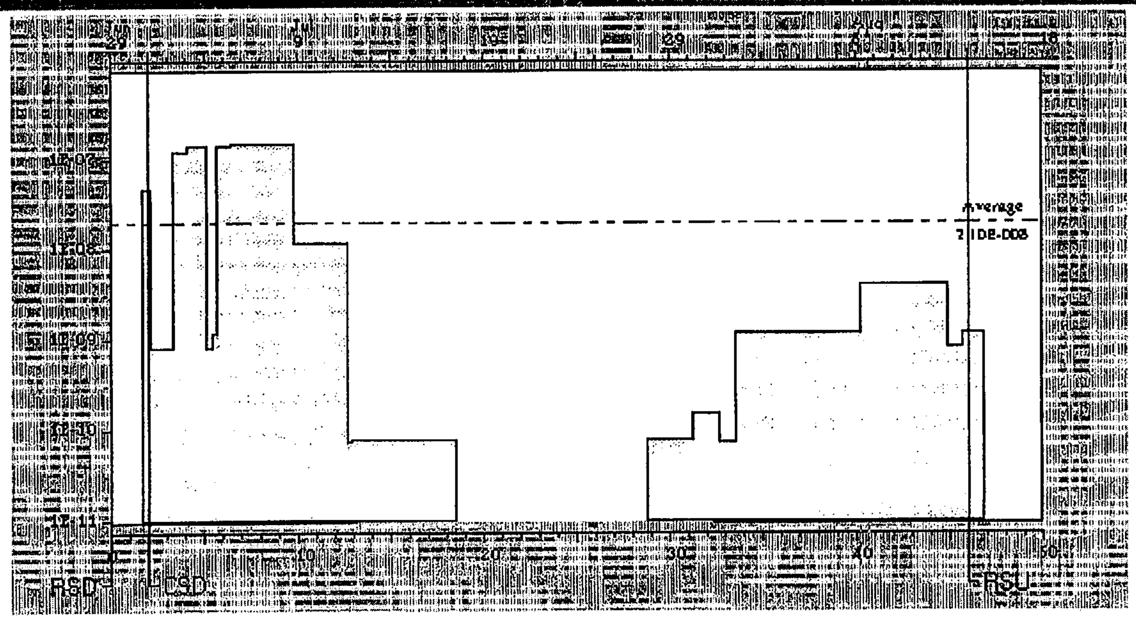
EPRI

EXAMPLE BWR SAFETY FUNCTION PROFILE



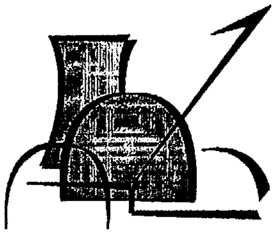
Slide 11

EXAMPLE PWR RISK PROFILE



"Average" CDF = 2.1E-8/hr*8760 hr/yr. ~1.8E-4/yr.

Slide 12



ANALYSIS of EXAMPLE PWR TYPICAL OUTAGE

- Average Configuration CDF (45 day outage) — ~1.8E-4/yr.
- Peak CDF — ~1E-3/yr.
- Min CDF — ~7E-7/yr.
- Range of CDF — ~1000
- Overall Contribution to Annual Average CDF (1.8E-4*45/365) — ~2.2E-5/yr.
- Contribution of Peaks to AACDF (6 days @ ~1E-3/yr.) — ~1.9E-5/yr.
- Contribution From Peaks — ~86%

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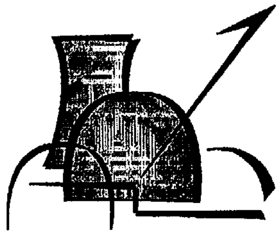
GENERAL SHUTDOWN PSA INSIGHTS

- Shutdown Risks Have Been Significantly Reduced Since NUMARC 91-06 Was Issued
- Due to Impact of "Peaks" Longer Outages are not Necessarily Safer - SD Risk Controlled By Minimizing Time of Peaks
- Strong Relationship Between How Outage is Planned and "Average" Risk
- More than 50% of "Average" Shutdown CDF is Due to Human Errors (During Peaks)

GENERAL SHUTDOWN PSA INSIGHTS (Cont.)

- **Major Factors in Shutdown Risk Level:**
  - **Plant Operating State**
  - **Human Performance**
  - **Decay Heat Level**
  - **Equipment Configuration**
- **Initiating Event Frequencies Seem to be Going Down**



SHUTDOWN PSA TECHNOLOGY ASSESSMENT

- Initiating Events Relatively Well Understood - Transitions are Challenging
  - Drain Down to Mid-loop
  - Switch Over of Running Pumps
  - Treatment of Instantaneous Risk Spikes?
- Accident Sequence & System Modeling Straightforward
- Success Criteria Not Fully Investigated, Probably Conservative for High Risk Intervals

Slide 16

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SHUTDOWN PSA TECHNOLOGY CONSIDERATIONS

- Component Performance Data Not Well Known
- Unavailability Data for Specific Outages Readily Available, Average Data is Not
- Human Reliability Most Difficult Aspect
- Limited Experience With Flooding & External Events - Fire & Flood Trickiest
- Quantification Tools are Not the Limiting Factor
- Level 2 and 3 Largely Unanalyzed

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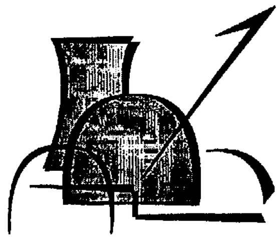

USE OF SHUTDOWN PSA RESULTS

- "Average" Shutdown Risks Are Not Comparable to At-Power Values:
  - Highly Outage Specific
  - Strongly Influenced By Durations of Key Plant Operating States
  - Dominated by Human Performance
- Computed Changes in Shutdown Risk Can Range from Negligible to Huge, Depending Upon Outage Schedule Assumed
- Decisions Should be Based on Level of Safety of Plant Configuration Regardless of Plant Mode

Slide 18

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CONCLUSIONS

- Trend in Shutdown Events is Significantly Downward
- A Significant Amount of Shutdown PSA Technology & Experience Exists Within the Industry
- Technology Well Developed but Still Improving
- Significant Uncertainties Exist:
  - Human Reliability
  - Plant Activities - Plant Response Linkage

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