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Development and Use of Risk-Based Inspection Guides

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Prepared by
J. H. Taylor, A. Fresco, J. Higgins, J. Usher, Brookhaven National Laboratory
S. M. Long, U.S. Nuclear Regulatory Commission

Brookhaven National Laboratory
Upton, NY 11973

Prepared for
Division of Radiation Protection and Emergency Preparedness
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555
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ABSTRACT

Risk-based system inspection guides, for nuclear power plants which have been subjected to a probabilistic risk assessment (PRA), have been developed to provide guidance to NRC inspectors in prioritizing their inspection activities. Systems are prioritized, and then dominant component failure modes and human errors within those systems are identified for the above-stated purposes. Examples of applications to specific types of NRC inspection activities are also presented.

Thus, the report provides guidance for both the development and use of risk-based system inspection guides. Work is proceeding to develop a method methodology for risk-based guidance for nuclear power plants not subject to a PRA.

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1. INTRODUCTION

1.1 Background

The Probabilistic Risk Assessment (PRA) is an analytical technique for integrating diverse aspects of design and operation in order to assess the risk of a particular nuclear power plant and to develop an information base for analyzing plant-specific and generic issues.

An assessment of the plant-specific risk provides both a measure of potential accident risks to the public and insights into the adequacy of plant design and operation.

The assessment of the adequacy of plant design and operation is achieved by identifying those sequences of potential events that dominate risk and by establishing which features of the plant contribute most to the likelihood of such sequences. These plant features may be subject to hardware failures due to human errors involving test, maintenance, or operational activities. Thus a probabilistic analysis provides a logical mechanism for revealing those features of a plant that may merit close attention and provides a focus for improving safety.

Information developed in the assessment could help in making decisions about the allocation of resources for safety maintenance or improvements, by directing attention to the features and their failure modes that dominate plant risk. The analysis may uncover new issues potentially generic to the industry. The Nuclear Regulatory Commission (NRC) can use this information to focus its resources on investigating problems most important to safety and to eliminate or reduce requirements and the expenditure of resources on issues of lesser importance.

To effectively utilize the insights gained from PRA, BNL developed a methodology for the integration of PRA insights into routine inspection activities. The methodology calls for analyzing the PRA to identify important plant systems and the failure modes of their risk significant components. System and programmatic based preventative inspections are then performed utilizing these insights and existing NRC inspection procedures.

1.2 Risk-Based Inspection Guides

In studying the content of the various NRC inspection procedures, it was determined that the best method by which to incorporate risk based inspection guidance into the inspection program was by providing guidance on the direction the inspector's efforts should take once a procedure was selected, rather than by modifying the content of the existing inspection procedures per se. The PRA integration technique relies on the existing NRC inspection program because the specification of levels and frequency provide a logical and effective general inspection methodology. It provides for inspections of all aspects of the nuclear facility within a framework that allows customization to the many plant designs. Therefore, under this PRA applications program, plant-specific Risk-Based Inspection Guides (RIGs) have been developed to be used in conjunction with the NRC inspection manual to provide pertinent PRA insights for each plant. (The guides have been previously entitled, "PRA-Based System Inspection Plans".) The guides developed to date and other relevant documents are listed in Section 5.

For example, when a system inspection is required, the RIGs can provide guidance to aid the inspector in selecting a system, and once selected, what items within that system to inspect. Furthermore, when a system walkdown is required, the RIG's provide abbreviated walkdown lists which focus only on risk sensitive components.

The RIGs contain material from the systems or event tree level and also from the component or fault tree level. The front part of the guide contains a systems priority list and the remainder of the plan identifies risk significant items by system with accompanying inspection recommendations. Additional detail on the content of a RIG is contained in Sections 2 and 3.

1.3 Report Objectives

The objectives of this report are threefold: 1) to standardize the format for the RIG, 2) to define the methodology for development of a RIG, and 3) to provide examples of applications of these guides in the NRC inspection program.

2. RIG FORMAT

This section defines the standard format for a plant specific, Risk Inspection Guide. Subsequent sections will describe the RIG development methodology and applications of the inspection plans.

The typical format for a RIG is shown in Figure 1, and discussed in succeeding paragraphs.

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Figure 1 RIG Format

A sample RIG format for Peach Bottom Unit 2 is shown in Attachment 1. The attachment should be referred to in the course of reading Chapters 2 and 3, which follow.

2.1 Introductory Sections

1. Introduction - A brief introduction which identifies any unique PRA features and states the level of PRA treatment. Guidance on uses of this document is provided. The use of PRA jargon is kept to a minimum. Plant specific design vulnerabilities and peculiarities of the PRA will be noted.

2. Dominant Accident Initiators and Sequences - Each accident sequence which either has a core melt frequency greater than 10 per year, or which contributes to the upper 95% of core melt frequency is discussed. The treatment is brief,

but should contain sufficient detail, so that initiating events and top-level systems, equipment and human actions are understood. The percent contribution of each sequence to the overall core melt frequency is presented in graphical form. The overall core melt frequency is also noted.

3. System Priority List - The plant systems are prioritized using risk based importance measures (e.g., Fussell-Vesely or Inspection Importance) which are representative of each system's contribution to core melt frequency. The Inspection importance measure results in ordering similar to the Fussell-Vesely measure. These measures describe the risk significance of systems when a plant is in normal operation. The Birnbaum measure is another commonly used measure which represents the importance of a system assuming that it has failed or is unavailable, and is then an indication of the importance of restoring it to service. This latter measure is particularly suitable for technical specification outage considerations.

Although the importance numbers are not included with the system ranking, the systems are rank-ordered based on the results of the importance measures. The measures are calculated based on the accident sequence cutsets representing approximately 95% of core damage frequency.

4. Common Cause Failures - A brief listing of the most important common cause failures identified in the PRA is provided. Typically, these consist of such items as common cause failures of the diesel generators, failure of the station batteries, or common aging of similar components.

5. Important Human Errors - A brief description of the most important human errors, categorized as either pre-accident or post-accident errors, are identified and discussed in sufficient detail to impart an understanding of the reasons why an error is particularly important. Typically, pre-accident errors consist of miscalibration of instrumentation or failure of the operators to restore a standby system to its proper alignment after testing or maintenance. Post-accident errors usually involve operator failure to initiate a standby system upon failure of automatic initiation.

6. System Inspection Tables (Discussion). - This section describes the content of the actual system inspection tables included in the Appendices A and B. The following general caution should be provided in this section:

> "The information in these tables allows an inspector to quickly identify the components most important to public risk - a combination of failure probability and the consequences of the failure, or more commonly the core damage frequency. In particular, the system walkdown tables can be used to rapidly review the line up of important system components on a routine basis. These tables can also be used when selecting systems for the performance of more detailed inspection activities.
>
> In using these tables, however, it is essential to remember that other systems can also be important. If, through inattention, the likelihood of other systems failing was allowed to increase significantly, their risk significance might exceed that of systems in the tables. Consequently, a balanced inspection program is essential to ensuring that the licensee is

minimizing plant risk. The following tables allow an inspector to concentrate on systems and components that are most significant to risk. In so doing, however, cognizance of the status of systems performing other essential safety functions must be maintained."

7. References - This section lists all reference material used in preparing the RIG, with the actual plant-specific PRA typically included as a minimum.

2.2 Appendices

The actual inspection tables are included in the RIG as appendices to the introductory sections described above. The individual appendices are as follows:

1) Appendix A System Inspection Tables

Table A-1, Importance Basis and Failure Mode Identification - The failure mode listing is preceded by a brief description of system configurations and the success criteria assumed by the PRA. The failure modes are rank ordered by probability, and include all failure modes which contribute to 95% of the system unavailability. Failure combinations are not presented in this table; each component or human error is treated separately. It is a goal to include those systems which contribute to the upper 95% of the core damage frequency in the RIG.

Failure mode information is provided in a brief statement, with additional clarification as warranted when a failure mode is determined to be particularly important due to specific plant design features or basic assumptions in the PRA.

Inspection activities are identified for each failure mode and are categorized as follows: operations and training, periodic surveillance and calibration, maintenance, technical specification, or inservice inspection-related.

Table A-2, Modified System Walkdown - This table provides an abbreviated version of the licensee's system checklist, where available, but includes only those items which are related to the dominant failure modes. It is generally less than one-third of the normal checklist. Caution should be observed when using the checklists, since they are based on certain versions of the licensee's system operating instructions. The revision date of the licensee's checklist is indicated at the end of the modified checklist. Attachment 1 contains Tables A3-1 and A3-2 for the Peach Bottom Unit 2 Low Pressure Core Spray System.

Simplified System Drawing - For each system, a simplified process and instrumentation diagram (P&ID), or electrical one-line diagram, upon which the PRA is based, is included for two reasons. The first reason is to make the inspectors aware of the system configuration upon which the PRA is based so that any significant changes which may affect relative component importances may be highlighted. The second reason is to provide a means for inspectors unfamiliar with the plant in question to quickly visualize and understand the operation and configuration of the important systems.

The following precautionary note should be placed in this section:

> "Note: This drawing is merely a simplified schematic of the actual P&IDs in effect at the time that the PRA was prepared. It is neither a complete representation of the P&IDs nor is it a controlled document. It was utilized in the preparation of the PRA and any significant differences between this drawing and actual plant conditions may affect the information provided in Tables 'AX-1' and 'AX-2,' and should be reported to the appropriate NRC personnel."

2) <u>Appendix B Plant Operations, Surveillance and Calibration, and Maintenance Inspection Guidance</u>

These tables are based on sorting information from the Appendix A Failure Mode Tables prepared for all the systems. For example, the table for the "Plant Operations Inspection Guidance" consists of the failure modes which relate to operator errors from all the Table 1s for failure modes which relate to operator errors. Similarly, the tables on surveillance and calibration inspection guidance, and maintenance guidance are prepared in the same fashion.

3) <u>Appendix C Containment (or Drywell) Walkdown</u>

The table for "Containment (or Drywell) Walkdown" is formed by selecting all of the components in the various modified system walkdowns (Table 2's) which are located inside the containment (PWRs) or drywell (BWRs). This allows the inspector to quickly look at the most safety-significant components when access is possible.

4) <u>Appendix D - Dependency Matrix</u>

Whenever it is readily available from a PRA, a matrix format system dependency chart should be included. Such a chart clearly delineates the relationship between front-line systems and their supporting systems. The interrelationship between support systems and other support systems should be shown on a separate chart.

A good example taken from a non-PRA source, the IDCOR Individual Plant Evaluation Method Applied to the Shoreham Nuclear Power Station (IDCOR-DKT-50322), is shown in Attachment 1 as Appendix D.

3. DEVELOPMENT OF A PLANT SPECIFIC RISK INSPECTION GUIDE

As an aid in development, this section discusses the portions of a RIG which require additional explanation beyond that provided in Section 2; namely the system ranking table and the individual system failure mode tables. The authors have assumed that the readers of this section have a working knowledge of Probabilistic Risk Assessments and Importance Measures. If not, Reference 15 provides a good treatment of the subject.

3.1 Methodology Discussion

Risk-Based Inspection Guides (RIGs) present inspectors with PRA insights at three different degrees of detail,: the accident sequence, the system, and the basic event level.

Prior to describing the methodology used for the ranking of systems and components, it is necessary to review the influence of the various levels of PRA's on this process. With a level 3 PRA (offsite consequence effects), there are several possible risk measures that could be used to determine the relative importance of a sequence or system or basic event. These include its contribution to the core melt frequency, the probability of early fatalities, and the total population dose. Each of these measures is likely to produce a significantly different ranking. For instance, LOCAs outside of containment usually do not contribute to a large fraction of the core melt frequency while they do often contribute to a large fraction of the early fatality probability. The differences in rankings for the different risk measures are due to the effects of the accident mitigation systems (e.g., containment, containment spray) and to the assumptions about severe accident phenomena (e.g., direct containment heating). Because the amount of uncertainty increases substantially between the core melt results and the radiological release results, and because many of the available PRAs do not go beyond core melt (level 1), the core melt frequency has been chosen as the risk measure for the RIGs calculations. However, the use of core melt frequency does not provide a means for ranking the accident mitigation systems, which are certainly important to public health and safety. Therefore, an inspector must separately direct his attention to these systems. This focus can be easily done if the plant has a level 3 PRA because these mitigation systems are specifically addressed. If the plant's PRA goes only to level 1 or 2, the list can be based on insights derived from analysis of several level 3 PRAs for similar plants.

In the discussions that follow, it will be assumed that core melt frequency is the risk measure being used to establish the importance of sequences, systems and components. Ranking of accident sequences according to their contributions to core melt frequency is a straight forward process; the frequencies of the cutsets assigned to each sequence are simply totaled and the results sorted into decreasing order. Most published PRAs contain such a ranking. However, sequence definitions used by PRA analysts are sometimes more narrowly drawn than the definitions that inspectors find useful for understanding the results. For instance, PRA reports may provide rank ordered lists that intersperse several small LOCA sequences, several ATWS sequences, and several LOSP sequences. The inspector is better served by combining the probabilities of the similar sequences to establish the ranking, and then for each general sequence type, describing the important variations in the path that the sequence can take. (The core melt frequencies for each of the sequence variations may still be of interest to inspectors, so they are included in the RIG with the sequence descriptions.)

At the system and basic event levels, there are two somewhat different insights that can be useful to inspectors. One is the contribution (or"impact") to the core melt frequency of the system or component when it is in the normal or operating condition. This can be determined by calculating one of the importance measures that include the actual reliability of the system or component. The Fussell-Vesely, Risk Reduction, and Inspection Importance Measures all provide essentially similar rankings of this type. The other insight of interest is the increase in the core melt frequency that results when the system or component fails or is out of service. The Risk Achievement Importance Measure[1] , as well as the Birnbaum Importance Measure provides this insight. Actual calculation of these importance measures is often complicated by the structure of the PRA, and may require considerable approximation if they must be calculated from published PRA results without benefit of the computerized plant models.

If the PRA has produced core melt cutsets that go to the basic event level, then it is straight forward to calculate the importance of the basic events (i.e., component failures and operator errors) to core melt. Calculation of the system's importance, in this case, requires assigning the various basic events to the appropriate systems and then, for each system, summing the frequencies of the cutsets containing one or more events assigned to that system. (Note that the importance of a system is not the algebraic sum of the importances of that system's basic events, since this would lead to multiple counting of the same cutsets for systems that have redundant components which appear jointly as basic events in numerous cutsets.) Usually, PRAs that produce core melt cutsets at the basic event level truncate the cutsets at a fixed number or fixed core melt frequency contribution. If the number of cut sets available after truncation is too small, it may not result in very accurate calculation of the Risk Achievement or Birnbaum Importances, because the associated assumption of system or component failure (i.e., the basic event failure probability is assumed equal to 1.0 as opposed to its normal value) can change the magnitude of the cutset frequencies by several orders of magnitude. A small number of cutsets may also result in identification of very few of the important components in some systems.

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[1]The Birnbaum Importance Measure for a system or component is the difference between the core melt probabilities assuming that it will always fail and assuming that it will never fail. The Risk Achievement Importance Measure is the difference between the core melt probabilities assuming the system has its normal level of reliability and assuming that it always fails. It differs from the Birnbaum measure by the amount that the system normally contributes to core melt frequency (also known as the Risk Reduction Importance Measure), i.e., the reduction in core melt frequency assuming it never fails. For systems that have high reliability (i.e., about 10-2 failure/demand), the Birnbaum measure approximates the Risk Achievement measure.

In contrast to the above case, many PRAs produce core melt cutsets that are composed of event tree top events, with most of these top events supported by fault trees that derive system unavailability from the basic events for a particular system. Calculation of the total core melt frequency in this type of PRA requires linking of the fault trees to the event trees with plant support states, to account for interactions/common dependencies of systems. As in the previous case, system importance is determined by assigning the events in the top level core melt sequences to the appropriate systems, and then, for each system, summing the frequencies of the event tree cutsets involving one or more events for that system. For this type of PRA, where only event tree cutsets are available, calculation of a basic event's importance must be related to a system's unavailability rather than core melt, which can be accomplished using the fault tree for that system. It should be noted that ranking components within a system according to their importance with respect to that system's unavailability is not necessarily the same as ranking them according to their importance with respect to core melt. For instance, in PWRs, motor driven pumps are as important as the turbine driven pump in the auxiliary feedwater system fault tree, but the turbine driven pump is often more important to core melt because it appears in the station blackout sequences and loss of emergency-bus sequences that do not require failure of the motor driven pumps. In this case, the RIG developer will usually have to make due with the importance of the basic events to system unavailability and some subjective rearrangement of the resulting ranking to account for the support states associated with dominant accident sequences.

The methods used by many PRAs create special problems for determining the importances of support systems and their components. Support systems often do not appear in the event trees, and their fault trees may not have been solved. However, some support systems have been shown to be very important for most reactors, (e.g., AC power and Service Water are important for all reactors) so they should not be ignored. When the support systems are not treated in a manner equivalent to the front-line systems, the method for determining their importances will require ad hoc development to take advantage of whatever information the PRA does contain. If the computerized plant models are available, linking the support system fault trees into the sequence event trees and rerunning the code may be practical. More commonly, the support system importances must be estimated by determining which of the event tree top events can be caused by failure of each support system (or train thereof) and assigning in a weighted fashion, the importance of the cutsets containing those events to the support systems, as well as to the systems that are explicitly involved.

3.2 System Ranking

With the aforementioned calculations completed, the system ordering is done in a numerically decreasing fashion with systems of approximately equal importance clustered together in groups separated by dashed lines from other groups which differ significantly in numerical importance ranking. As an example, in Attachment 1, Table 1 shows a hypothetical system priority ranking according to the importance of the systems in preventing core damage for the Peach Bottom Station. In the left-hand column, the systems are ranked according to their Fussell-Vesely importance while in the right hand column, the systems are ranked according to their Birnbaum importance. The discussion in Section 3 (System

Priority List) of Attachment 1 is provided for the benefit of the individual inspectors in interpreting and applying the lists to the inspection process.

3.3 System Failure Modes

Using the basic event importances that were previously calculated, the failure mode tables are constructed by listing, in order of decreasing importance, the components and operator actions that contribute to approximately 95% of the system unavailability. As previously discussed, this basic event importance may be based on its contribution to core melt or its contribution to system unavailability, depending on PRA format.

It should be noted that "like" components or human actions are grouped together in the system failure mode table (e.g., pump A and pump B are listed as one line item).

Attachment 3 provides examples of system failure mode tables (Tables A-1 Importance Basis and Failure Mode Identification) for the Peach Bottom Plant.

4. APPLICATIONS OF RISK-BASED INSPECTION GUIDES

Risk-Based Inspection Guides are intended to provide Resident Inspectors with risk insights that are applicable to a wide variety of inspection activities required by the NRC Inspection Manual. The Manual contains the Inspection Procedures used by NRC for all routine and occasional inspection activities. Section 2515.10 of the manual discusses use of PRA insights and References Appendix C, which describes the RIGs and lists those that are presently available. In addition, risk insights from the RIGs can be useful during planning for a variety of team inspections, including Safety System Functional Inspections, Maintenance Team Inspections, and Operational Safety Assessment Risk Based Inspections. The examples provided below illustrate several of the methods for using risk insights from the RIGs during the planning or conduct of inspections.

<u>NRC Inspection Manual 2515</u> - This chapter delineates the routine inspection activities for power reactors after they have completed their initial power ascension testing. Under Inspection Procedure (IP) 71707, Resident Inspectors are required to perform specified activities on daily, weekly, monthly and longer periods.

During daily tours of the control room and reviews of operations logs, familiarity with the accident sequences and system failure modes described in the RIG will aid in recognizing situations that are potentially risk significant. Important system line-up errors are often detectable by control room observations.

Required weekly activities include walking down a plant system, with considerable latitude allowed to the inspector for determining the thoroughness of this inspection. Although each system in the plant should be covered eventually, the relative risk significance indicated by the RIG can be used in determining the order for selecting the systems, and more importantly, the thoroughness of the inspection for a particular system. The modified system walkdown tables provided in the RIG should be used to ensure that the most risk significant items are included in even the most abbreviated walkdowns.

Inspections of maintenance activities are required on approximately monthly intervals, using IP 62703. Similarly, monthly inspections of surveillance activities are required, using IP 61726. The inspector is urged to begin his inspections of this type by directly observing the licensee's performance of a maintenance or surveillance activity that is important to risk. The inspector can use the system and component importance information in the RIG to help identify the most risk significant maintenance and surveillance activities scheduled during the appropriate periods.

<u>Safety System Functional Team Inspections (SSFI)</u> - These are conducted in accordance with Appendix D to Manual Chapter 2515 at the discretion of the NRC's Regional Office. They are intensive inspections that go into great depth on a single system. They usually begin with verification that the system design is consistent with its design requirements, progress through

the adequacy of installation, history of operation, adequacy of surveillance and maintenance procedures, and include a detailed walkdown of the system.

The information in the RIG can be useful in selecting a system for this inspection and for ensuring that important failure modes are included in the inspection planning process.

Maintenance Team Inspections - These are intensive inspections of a plant's maintenance program that are being conducted in accordance with Temporary Instruction 2515/97 at each plant during the 1988-89 period. part of the inspection procedure involves selection of specific equipment for detailed review of maintenance procedures, records, and failure information. The information contained in the RIG can be used to ensure that components with high risk significance are chosen for the inspection sample and that their important failure modes are adequately addressed by the maintenance program.

Risk-Based Operational Safety and Performance Assessment (ROSPA) - These team inspections are conducted at the discretion of the NRC Regional Office in accordance with IP 93804. They focus on a plant's ability to respond to the accident sequences that dominate its core melt frequency. Preparation usually involves direct extraction of importance information from the plant's PRA. However, information in the RIG can also be utilized if time or expertise is not available for analyzing the PRA. Approximately 40 basic events that contribute the most to core melt frequency are chosen for inspection. These usually include both component failures and operator actions. Thorough inspections of the components are planned, including direct inspection and reviews of maintenance, surveillance and calibration records and procedures. Operator actions are reviewed through accident simulation exercises, reviews of Emergency Operating Procedures and plant walk-throughs. In all of these activities, the role of the equipment and operator actions in particular accident sequences is used to check for adequacy under the conditions that would be imposed by the accidents.

During any inspection of a system or an individual component, it is useful to consider the type of failures that create significant risk and the circumstances under which these failures are important. For instance, if the RIG indicates that failure of a normally open valve to remain open is significant during a transient, then inspection should concentrate on verifying that the valve is in the open position, ensuring that the disk has not separated from the stem, and examining the efficacy of the measures used by the licensee to ensure that the valve is not inadvertently closed. In contrast, if the RIG indicates that failure of a valve to open under Station Blackout conditions is an important step toward core melt, the inspector would concentrate his attention on those things that could prevent the valve form opening, with special emphasis on the conditions created by loss of AC power. An experienced inspector's knowledge of failure mechanisms, used in conjunction with the failure modes information provided by the RIG, can effectively focus inspection efforts for the greatest safety benefit.

5. REFERENCES

Risk-Based Inspection Guides (formerly called PRA-Based System Inspection Plans) Published by BNL

1. Millstone Unit No. 1 - BNL Technical Report A-3453-3-87, March 1987.

2. Indian Point Unit 3 - BNL Technical Report A-3453-87-1, May 1987.

3. Shoreham Nuclear Power Station - BNL Technical Report A-3453-87-3, May 1987.

4. Limerick Generating Station Unit 1 - BNL Technical Report A-3453-87-2, May 1987.

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7. Brunswick Steam Electric Plant Unit 2-BNL Technical Report A-3872-T4 (Proprietary), (Draft to be issued).

8. Washington Nuclear Power Plant No. 2 Generic Risk Based Inspection Plan, BNL Technical Report A-3453-87-6, June 1987.

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ATTACHMENT 1

SAMPLE RISK-BASED

INSPECTION GUIDE

BASED ON THE

PEACH BOTTOM ATOMIC

POWER STATION

UNIT 2

(Contains Sections 1 - 7, and selected Appendices from the Peach Bottom Risk Inspection Guide)

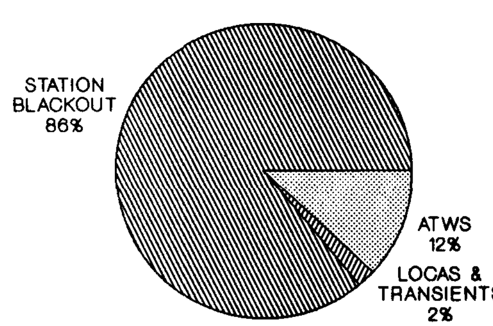
1. INTRODUCTION

This inspection guide has been prepared to provide inspection guidance based on review of the NUREG/CR-4550/Peach Bottom Probabilistic Risk Assessment (PRA)[1]. The guidance should be used to aid in the selection of areas to inspect and is not intended either to replace current NRC inspection guidance or to constitute an additional set of inspection requirements. The information contained herein is based almost entirely on the Peach Bottom PRA issued in 1986. Hence, recent system experience, failures, and modifications should be considered when reviewing these tables. **Since plant modifications are normally an ongoing process it is recommended that relevant changes be catalogued so that this inspection guidance can be periodically revised as required.**

2. DOMINANT ACCIDENT SEQUENCES

The Peach Bottom PRA has a number of different accident sequences that contribute significantly to overall core damage frequency (CDF), which is 8.2E-6/year. The sequences that dominate core damage frequency at Peach Bottom are grouped below by their initiating events.

- Station Blackout
  (86% of core damage frequency)
- Anticipated Transients
  Without Scram (ATWS)(12%)
- Intermediate LOCA (1%)
- Transient with Loss of
  Core Cooling (<1%)
- Large LOCA (<1%)



Each of these dominant accident sequence groups is composed of several similar but distinct sequences of systems failures. There are five dominant station blackout sequences (three short-term and two long-term), four dominant ATWS sequences (two dependent and two independent of containment failure), and four loss-of-core-cooling sequences (two LOCAs and two transients). Because of similarities, the sequences have been grouped and summarized below.

2.1 Station Blackout Sequences (86% CDF)

2.1.1 Short Term Blackout

There are two sequences resulting from short-term station blackout which comprise a total of 56% of core damage frequency:

a) The first is characterized by transients leading to station blackout (loss of all AC power) as a result of coincident DC power failures. The loss of DC power causes failure of the diesel, High Pressure Coolant Injection (HPCI), and Reactor Core Isolation Cooling (RCIC) systems which results in the loss of all core and containment cooling. Without the restoration of AC/DC power in 30-to-40 minutes, primary system inventory boils off and core damage results.

   In addition, AC power recovery is affected by the DC power loss severely hampering the recovery process for reclosing breakers, etc. Instrumentation in the plant is also significantly degraded under these circumstances. For these reasons, the probability of power recovery is considered negligible in the required 30 minute time frame to prevent core damage (54%).

b) The second is very similar to the above except; there are no DC common mode failures. The diesels, HPCI, and RCIC fail by other mechanisms.

These sequences may include RCS depressurization by the ADS (if DC power is available) or by a stuck open SRV. However, low pressure core cooling is not functional without AC power, so this only affects whether core damage occurs with the RCS at high pressure or low pressure.

2.1.1 Long Term Blackout

There is one major damage state resulting from long-term station blackout which contributes to 30% of core damage frequency. It is characterized by transients leading to a long-term station blackout (loss of all AC power). Core cooling is successful with either HPCI or RCIC providing coolant injection until about six hours into the sequence. At that time, the batteries deplete, affecting the ability to continue operation of these systems. Without AC power recovery within three hours of battery depletion, core damage results. While the primary system may be initially at relatively low pressures, depletion of the batteries causes loss of ADS/SRV control. Core damage occurs either at high pressure conditions or at low pressure conditions caused by a stuck open relief valve.

2.2 Anticipated Transients Without Scram (ATWS) (12% CDF)

There are two major damage states attributable to ATWS scenarios comprising 7% and 5% CDF respectively,

2.2.1 ATWS with Core Damage Independent of Containment Failure

This plant damage state is characterized by an ATWS with their Main Steam Isolation Valve (MSIV) closure or an event with MSIVs initially open but subsequently closed. This isolates the primary system under high power conditions, thereby rapidly increasing the pressure and temperature conditions within containment since RHR cooling under these circumstances is inadequate.

At this point, there are two pathways leading to core damage:

1) The Standby Liquid Control (SLC) system is started within ~4 minutes into the accident, but initial HPCI failure under high power conditions and operator failure to rapidly depressurize the vessel (so that low pressure systems can be used immediately) lead to a core damage event. Subsequent containment failure may or may not occur depending on the need for, and success or failure of, containment venting (5% CDF).

OR,

2) Timely start of the SLC system is not performed or it fails from being left in an inappropriate configuration after the last test of the system. Core cooling is maintained for a short time (~1/4 hour) before HPCI fails because of high pool temperature. The operator then fails to rapidly depressurize (so that low pressure systems can be used) which leads to core damage. Subsequent containment failure may or may not occur depending on the need for, and success or failure of, containment venting (2% CDF).

2.2.2 ATWS with Core Damage Dependent Upon Containment Failure

This plant damage state is characterized by an ATWS with either MSIV closure or an event with MSIVs initially open but subsequently closed. This isolates the primary system under high power conditions, thereby rapidly increasing the pressure and temperature conditions within containment since RHR cooling under these circumstances is inadequate. Timely SLC system start is not performed or it fails because it was left in an inappropriate configuration after the last test of the system. ADS is not inhibited, resulting in vessel blowdown. Low pressure system operation and control are successful. Venting of the containment is not successful.

The status of the containment determines how core damage occurs. Three general containment conditions assumed are:

a) containment leak failures,

b) no containment failure at least up until vessel breach, and

c) catastrophic containment failure.

Cases a) and b) preclude continued operation of the low pressure cooling systems. This is because maximum air pressure to the SRVs is ~100-to-125 psig, which is under the estimated 150+ psig pressure for containment failure. Therefore, the vessel remains pressurized and all core cooling is lost. Case c) depressurizes the containment, but the saturated conditions in the pool cause failure of Low Pressure Core Spray (LPCS) and RHR pumps. Condensate and High Pressure Service Water (HPSW) are either not available or the operator fails to start their injection into the core; core damage results (5% CDF).

2.3 <u>LOCAs or Transients With Loss of Core Cooling (2% CDF)</u>

2.3.1 Intermediate LOCA

Subsequent to an intermediate size LOCA, HPCI successfully operates for about two hours until pressure in the primary system can no longer support operation of the HPCI steam turbine. Low pressure injection systems are required to provide sufficient flow, but they fail. Core damage results soon after (1% CDF).

2.3.2 Large LOCA

Subsequent to a large LOCA, there is failure of the low pressure systems resulting in a core damage event (<1% CDF).

2.3.3 Transients with Loss of the Power Conversion System

1) This plant damage state is characterized by a transient causing loss of the Power Conversion System (PCS). Early loss of all core cooling occurs because of failures associated with the high pressure systems and the inability of the available low pressure systems to inject because of miscalibration of the low reactor pressure permissive circuitry. This latter event disables LPCS and LPCI, as well as HPSW injection which uses the LPCI injection paths. Without recovery of the PCS and accompanying condensate or feedwater in about 30 minutes, core damage results. The vessel can, and will likely be, depressurized with ADS leading to core damage under low pressure conditions in the reactor vessel (<1% CDF).

2) Similar to (1) above, this state involves a transient causing loss of the PCS and early failure of all injection. Injection loss is because of failures associated with the high pressure systems, ADS, and operator failure to manually depressurize so that low pressure systems can be used. Core damage results in about 30 minutes without recovery (<1% CDF).

3. SYSTEM PRIORITY LIST

The Peach Bottom systems have been ranked in Table 1 according to their importance in preventing core damage. Two different rankings are provided for use under two types of circumstances. Under normal conditions, the left-hand column should be used. For degraded or inoperable systems, the right-hand column should be used, as discussed below. Other plant systems not appearing on these lists are generally of lesser importance than those that are included here.

The two system prioritization lists have been included in Table 1 because they provide different types of risk insights that are useful in the inspection process. The left-hand column indicates the system's contribution to the core damage frequency as provided by the Fussell-Vesely Importance Measure, given that the system is operating with the reliability assumed by the PRA. Generally, when planning an inspection without knowledge of specific system problems, those systems that contribute most to core damage frequency should be given priority attention in order to most efficiently minimize risk.

However, when one or more systems exhibit unusually high failure rates or unusual types of failures, then the probabilities assumed in the PRA are not really appropriate for the failures of those systems. While their problems persist, the affected systems contribute more to the risk of core damage than is indicated by the left-hand column. The increase in the core damage

Table 1 - (Hypothetical) System Priority Ranking

By Contribution to Core Damage Frequency[1]	By Risk Significance of the System Being Unavailable[2]
Emergency Power	Reactor Protection
Containment Venting	
- -	- - - - - - - - - - - - - - - - -
Emergency Service Water	Emergency Service Water
Reactor Protection	Emergency Power
Automatic Depressurization	Emergency Ventilation
- -	- - - - - - - - - - - - - - - - -
Standby Liquid Control	Standby Liquid Control
High Pressure Coolant Injection	Containment Venting
Low Pressure Coolant Injection	Automatic Depressurization
- -	High Pressure Coolant Injection
Reactor Core Isolation Cooling	Reactor Core Isolation Cooling
Emergency Ventilation	Low Pressure Core Spray
Control Rod Drive	Control Rod Drive
- -	- - - - - - - - - - - - - - - - -
Low Pressure Core Spray	
Condensate/Feedwater	Condensate/Feedwater
Containment Sprays	Containment Sprays
High Pressure Service Water	High Pressure Service Water
Instrument Air	Instrument Air
Shutdown Cooling	Shutdown Cooling
Suppression Pool Cooling	Suppression Pool Cooling

Notes:

1. The ranking in column 1 is appropriate to use for systems that are functioning normally. It is based on the Fussell-Vesely Importance Measure, which is the system's contribution to the core damage frequency, assuming that the system is operating with normal reliability.

2. The ranking in column 2 is appropriate to use for determining the significance of known system degradation or inoperability. It is based on the Birnbaum Importance Measure, which indicates the increase in the core damage frequency that results when the system is assumed to be inoperable.

3. The containment systems shown on these lists are ranked with respect to their contributions to core damage frequency, only. Their importance for accident consequence mitigation was not considered.

4. The dashed lines represent significant differences between importances of systems that are adjacent in the lists. Systems not separated by dashed lines should be assumed to have importances approximately equivalent to each other, within the precision of the PRA quantification.

5. The containment spray system, shutdown cooling system and suppression pool cooling system have been combined under the residual heat removal system in the inspection tables that follow.

frequency when the system is inoperable is indicated by the right-hand column, based on the Birnbaum Importance Measure. The right-hand column can be used to estimate how much more important these systems have become when they are having problems. (Affected systems with high rankings in the right-hand column should be considered to have become much more important than indicated by their rank in the left-hand column, while systems with lower rankings in the right-hand column would have smaller increases above the rank indicated in the left-hand column.) Similarly the right-hand column is the appropriate choice for estimating the risk significance of inspection findings that indicate a system is inoperable or degraded.

Adjacent systems on the list should be considered to have approximately equal contributions to risk because of the uncertainties in the PRA. Where the difference between importance measures of adjacent systems is significant, they have been separated by the dashed lines.

4. COMMON CAUSE FAILURES

The failure of multiple items from some common cause can be very significant to risk. The Peach Bottom PRA has identified several common cause failures that are particularly important:

- Loss of offsite power,
- Common mode failure of the DC batteries,
- Common mode failure of diesel generators,
- ADS valves fail because of a common cause.

Other common cause failures, not considered to be as important as those above, are identified in the failure mode tables which follow.

5. IMPORTANT HUMAN ERRORS (Including Recovery Actions)

Human errors can be very significant to overall plant risk. The Peach Bottom PRA has identified several human errors as particularly important contributors to risk:

5.1 Pre-Accident Errors

1) Miscalibration of the reactor pressure sensors (PISL-2-3-52A-D) shared by the Low Pressure Core Spray (LPCS) and Low Pressure Coolant Injection (LPCI) systems.

   While a low probability event, this error could cause failure of the LPCS, LPCI, and High Pressure Service Water (HPSW) system (which injects through the LPCI line) since low reactor pressure permissives to open the injection valves in these lines would become unavailable.

2) Failure to restore the correct standby alignment of the Standby Liquid Control (SLC) system after test. Failure to restore certain valve after tests of the SLC system could cause recirculation of the borate solution rather than injection into the vessel upon a real demand. Although the valves are painted to direct closure after the tests, there are no control room position indicators.

5.2 Post-Accident Errors

1) Operator fails to initiate Standby Liquid Control within four minutes of ATWS.

2) Operator controls level with HPCI too low. Following an ATWS, at 100°F torus temperature since power is above 3% and an SRV is open, the operator must lower reactor pressure vessel level by terminating and preventing all injection into the vessel, except boron injection and control rod drive, until power is below 3% or all SRVs are shut or the top of the active fuel (TAF) is reached. As the TAF is reached, HPCI must be throttled to maintain the level. One outcome is operator failure by maintaining the level too low.

Upon failure of automatic ADS initiation:

3) The operator fails to rapidly depressurize the primary system (using the ADS valves), or

4) The operator fails to operate the non-ADS SRVs manually.

Other human errors are also identified in Table D1, Plant Operations Inspection Guidance.

6. SYSTEM INSPECTION TABLES

Taken together, the systems ranked by their risk importance in the first column of Table 1 contribute 95% of the core damage frequency for Peach Bottom. For each of those systems, inspection guidance is provided in the form of a failure mode table, an abbreviated walkdown checklist, and a simplified system diagram. Each of these is explained in detail below.

In using these tables, however, it is essential to remember that other systems and components are also important. If, through inattention, the failure probabilities of other systems were allowed to increase significantly, their contributions to risk might equal or exceed that of the systems in the following tables. Consequently, a balanced inspection program is essential to ensuring that the licensee is minimizing plant risk. The following tables allow an inspector to concentrate on systems and components that are most significant to risk. In so doing, however, cognizance of the status of systems performing other essential safety functions must be maintained.

APPENDIX A

Table AX-1 - System Failure Modes

The introduction to this table provides a brief description of the system and the success criteria used for the system in the PRA. (Note that the PRA success criteria may be different from the success criteria contained in the FSAR).

The entries in this table are the dominant events (component failures, operator errors, etc.) contributing to system failure, provided in rank order according to their risk significance. Since most systems are designed with redundant trains, it will generally take more than one of these events to fail the entire system. No effort has been made to list all of the combinations of the events that are sufficient to produce system failure because that is usually apparent from the system description in the introduction. Where single events are sufficient to fail the entire system, that is noted in the brief discussion of the event. For certain events that are important primarily because of the circumstances of a particular accident sequence, that information is also noted.

Inspection focussed on the items in the table will address approximately 95% of the risk for that system. Because PRAs do not contain the detail necessary to attribute the listed failures to the most probable specific root causes, it is necessary for the inspector to draw from his experience, plant operating history, ASME Codes, NRC Bulletins and Information Notices, INPO

SOERs, vendor notices and similar sources to determine how to actually conduct his inspections of the listed items. Were appropriate, codes have been included following each event description to indicate which licensee programs/activities provide inspectable aspects of the risk. These codes are as follows:

PC - Periodic calibration activities, procedures and training.

PC - Periodic testing activities, procedures and training.

MT - Preventive or unscheduled maintenance activities, procedures and training.

OP - Normal and emergency operating procedures, check-off lists, training, etc.

TS - Technical specifications.

ISI - In-service inspection.

Table AX-2 - Modified System Walkdown

This table provides an abbreviated version of the licensee's system checklist, where available, but includes only those items which are related to the dominant failure modes. It is generally much less than the normal checklist. It can be used to rapidly review the line up of important system components on a routine basis. Caution should be observed when using the checklists, since they are based on certain versions of the licensee's system operating instructions. Valve numbers used are those identified in the licensee system checklists, or P&ID's.

Figure AX - Simplified System Diagram

A simplified line diagram is provided for each system treated. These are intended to aid in visualizing the system configuration and the location of the components discussed in the two tables. Since they are neither complete nor controlled, they should not be used in place of up-to-date P&IDs during inspection activities.

APPENDIX B

Table B1 - Plant Operations Inspection Guidance

This table is a collection of all of the risk significant operator actions listed in the preceding system tables. It is provided as a cross reference for use in observing operator actions and training.

Table B2 - Surveillance and Calibration Inspection Guidance

This tale is a collection of all of the risk significant components listed in the preceding system tables that are considered to be significantly influenced by surveillance and calibration activities. It is provided as a cross reference to assist in selecting risk important activities for observation during inspections of the licensee's surveillance and calibration programs.

Table B3 - Maintenance Inspection Guidance

This table is a collection of the risk significant components listed in the preceding system tables that are considered to be significantly influenced by maintenance activities. It is provided as a cross reference to assist the inspector in selecting risk important activities for observation during inspections of the licensee's maintenance program. Important factors include the frequency and duration of maintenance as well as errors that degrade the component or render it inoperable when it is returned to service.

APPENDIX C

Table C1 - Containment and Drywell Walkdown Table

Because they are normally inaccessible during operation, a separate walk-down checklist is provided for those components listed in the preceding system tables that are located inside the containment or drywell. This is intended for efficient inspection of those items when the opportunity arises.

APPENDIX D

System Dependency Matrix

In performing a Probabilistic Risk Assessment for a power reactor, it is necessary to determine the dependencies (and interdependencies) of front-line ESF systems and support systems. This information is often provided with the PRA in the form of a matrix. The system dependency matrix from the Peach Bottom PRA is included here to aid the inspector in determining what other systems (or trains of systems) are affected when a particular system or train fails. This can be helpful in appreciating the importance of systems and in reviewing the adequacy of operator actions when systems become inoperable.

7. REFERENCES

1. A.M. Kolaczkowski, et al., "Analysis of Core Damage Frequency From Internal Events; Peach Bottom, Unit 2," NUREG/CR-4550, SAND86-2084, Volume 4, Sandia National Laboratories, Albuquerque, NM, October 1986.

2. NUREG/CR-5022, "System Analysis and Risk Assessment," System (SARA) User's Manual Revision 3 and Software, Idaho National Engineering Laboratory, September 1987.

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Low Pressure Core Spray (LPCS) System

Table A3-1 Importance Basis and Failure Mode Identification

CONDITIONS THAT CAN LEAD TO FAILURE

Mission Success Criteria

The purpose of the Low Pressure Core Spray (LPCS) System is to provide a makeup coolant source to the reactor vessel during accidents in which vessel pressure is low. ADS can be used in conjunction with LPCS to attain a low enough system pressure for injection to occur. LPCS is a two loop system consisting of motor-operated and manual valves and two 50% capacity motor-driven pumps per loop. Injection of flow from any two LPCS pumps to the reactor constitutes system success. The LPCS pumps take water from the suppression pool and can be manually religned to the CST. LPCS is automatically initiated and controlled. The operator may be required to manually start the system if an automatic actuation failure occurs. The operator can stop or control flow during ATWS if required.

1. Common Cause Miscalibration of Reactor Pressure Sensors

   LPCS will not actuate unless system pressure is sufficiently low. (PC)

2. ESW PS-5 Hardware Failures: CV 513 Fails to Open or Manual Valve XV 502 Plugs

   ESW PS-5 is the common injection line to all of the LPCS pump and room coolers. Failure of pump room cooling is assumed to fail the LPCS pumps in four hours. (MT, PT)

3. Operator Fails to Backup LPCS Actuation

   LPCS is automatically actuated but the opertor may be required to manually actuate the system given auto failure. (OP)

4. Common Cause Miscalibration of Reactor Water Level Sensors

   Given sufficiently low system pressure, low reactor water level sensors actuate LPCS. (PC)

5. MOV 12A/B Fail to Open

   Failure of injection lines PS-13 and PS-27 disable LPCS. (MT, PT)

6. Bus 4160A/B/C/D Power Permissive Sensors Fail

   Failure of Bus 4160 A or C causes failure of PS-13, while failure of Bus 4160 B or D disables PS-27. (MT, PT)

7. LPCS Pump A/B/C/D Fail

Failure of three of the four LPCS pumps or one pump in conjunction with the alternate loop's injection line disables LPCS. (MT, PT)

8. ESW PS-8 Fails and Operator Fails to Switch to EHS Mode

Maintenance on MOV 0498 disables ESW PS-8, the primary heat sink discharge line; unavailability of the primary heat sink requires that the emrgency heat sink mode be actuated. EHS Mode is also disabled by ECW pump failure or MOV 0841 failing to open (PS-19). (MT, OP, PT)

9. LPCS, LPCI Low Reactor Pressure Sensors C&D/A&B Fail

LPCS and LPCI share actuation logic. The low pressure systems will not actuate unless reactor pressure is sufficiently low. (MT, PT)

10. MOV 11A/B Out for Maintenance

See Item 5. above. (MT, PT)

11. MOV 5A/B/C/D Out for Maintenance

Pump discharge lines are disabled when they are blocked for MOV maintenance. (MT, PT)

12. MOV 26A/B Fail to Remain Closed

PS-12 and PS-26 are discharge lines to the suppression pool. These lines will divert flow from LPCS reactor injection. (MT, OP, PT)

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Low Pressure Core Spray (LPCS) System

TABLE A3-2 MODIFIED SYSTEM WALKDOWN

Description	ID No.	Location	Desired Position	Actual Position	Pow.Sup. Breaker#	Location	Required Position	Actual Position
LPCS & INJ Line MOV's	MOV 12A	CR Panel CO3	Auto Closed		52-3621	MCC 20B36	Closed	
LPCS A INJ Line MOV's	MOV 12B	CR Panel CO3	Auto Closed		52-3952	MCC 20B39	Closed	
Emer. Aux. Swgr. Bus 4 kV	E-12	CR Panel C26A	AM. I. Indic.		152-1501	Emer. Aux Swgr. 20A15	Closed	
Emer. Aux. Swgr. Bus 4 kV	E-22	CR Panel C26B	AM. I. Indic.		152-1601	Emer. Aux Swgr. 20A16	Closed	
Emer. Aux. Swgr. Bus 4 kV	E-32	CR Panel C26C	AM. I. Indic.		152-1701	Emer. Aux Swgr. 20A17	Closed	
Emer. Aux. Swgr. Bus 4 kV	E-42	CR Panel C26D	AM. I. Indic.		152-1801	Emer. Aux Swgr. 20A18	Closed	
LPCS Pump	2A P37	CR Panel CO3	Auto		152-1504	Emer. Aux Swgr. 20A15	Closed	
LPCS Pump	2B P37	CR Panel CO3	Auto		152-1604	Emer. Aux Swgr. 20A16	Closed	
LPCS Pump	2C P37	CR Panel CO3	Auto		152-1703	Emer. Aux Swgr. 20A17	Closed	
LPCS Pump	2D P37	CR Panel CO3	Auto		152-1803	Emer. Aux Swgr. 20A18	Closed	
ECW Pump	MDPA-C	See Note 1						
PS 12 Discharge Line MOV	MOV 26 A	Cont. SW CR Panel CO3	Closed		52-3823	MCC 20B38	BKR Closed	

Low Pressure Core Spray (LPCS) System

TABLE A3-2 MODIFIED SYSTEM WALKDOWN

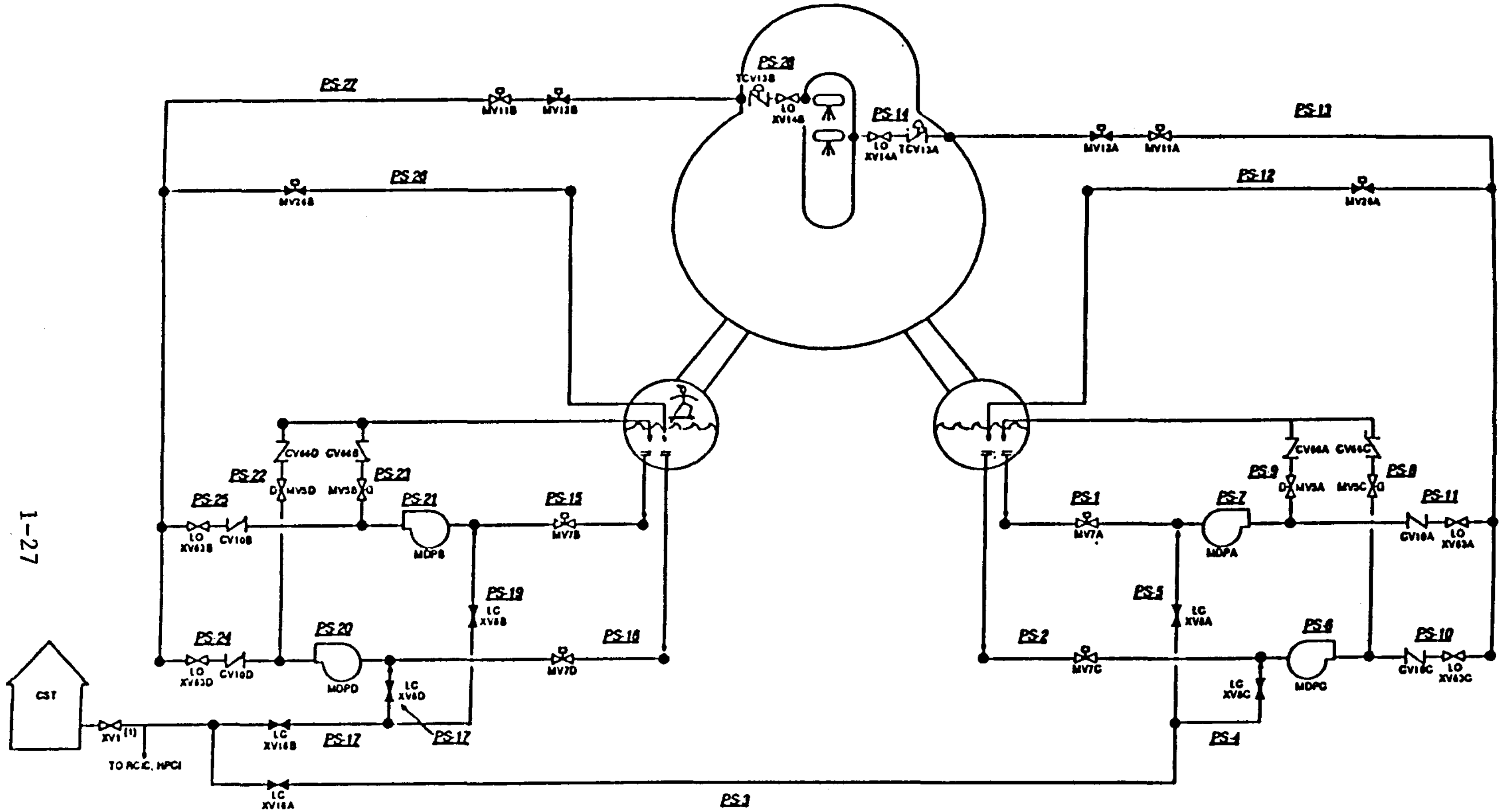
Description	ID No.	Location	Desired Position	Actual Position	Pow.Sup. Breaker#	Location	Required Position	Actual Position
PS 26 Discharge Line MOV	MOV 26 B	Cont. SW CR Panel CO3	Closed		52-3932	MCC 20B39	BKR Closed	

Note 1: For failure of EHS Mode by ECW Pump failure or MOV 0841 Failing to open – See A2-3 ESW.

TABLE A3-2

REFERENCE DOCUMENTS

TITLE	I.D. NO.	REV	DATE
Systems Procedures:			
Core Spray – Normal – Value Line-up	S.3.4.E.2	6	6/24/87
System Descriptions:			
Core Spray	3.4	1	1/09/80
Training Plans:			
Doc Type 284 – Core Spray	0350	3	7/07/87
Electrical One Line No.:			
Standby Diesel Gen. & 4160 Volt Emerg.	E-8	10	5/06/86
Power System, Unit No. 2	Sheets 1&2		
P&ID's No.:			
Core Spray Cooling System	M-362	31	9/14/87
	Sheets 1&2		



VALVE POSITIONS ARE SHOWN IN THEIR STANDBY POSITION
(1) VALVE ALSO LOCATED ON HPCI SCHEMATIC, SEE HPCI SCHEMATIC FOR DEFINITION OF PIPE SEGMENT

Figure A3-1. Low Pressure Core Spray System Schematic.

Note: This drawing is merely a simplified schematic of the actual P&ID's in effect at the time that the PRA was prepared. It is neither a complete representation of the P&IDs nor is it a controlled document. It was utilized in the preparation of the PRA and any significant differences between this drawing and actual plant conditions may effect the information provided in Tables A3-1 and A3-2 and should be reported to the appropriate NRC personnel.

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TABLE B1 - PLANT OPERATIONS INSPECTION GUIDANCE

Recognizing that the normal system lineup is important for any given standby safety system, the following human errors are identified in the PRA as important to risk.

SYSTEM	FAILURE	DISCUSSION
Emergency Service Water (ESW)	Operator Fails to Initiate EHS Mode	Table A2-1, Item 8
Low Pressure Core Spray (LPCS)	Operator Fails to Backup LPCS Actuation	Table A3-1, Item 3
	Operator Fails to Switch to EHS Mode	Table A3-1, Item 8
	MOV 26A/B Fail to Remain Closed	Table A3-1, Item 12
Residual Heat Removal (RHR)	Operator Fails to Backup LPCI Actuation	Table A5-1, Item LPCI-3
	Operator Fails to Switch to EHS Mode	Table A5-1, Item LPCI-6
	Operator Fails to Align CS Mode	Table A5-1, Item CS-1
	Operator Fails to Switch to EHS Mode	Table A5-1, Item CS-7
	Operator Fails to Initiate or Align SDC Mode	Table A5-1, Item SDC-1
	Operator Fails to Switch to EHS Mode	Table A5-1, Item SDC-7
	Operator Fails to Align SPC Mode	Table A5-1, Item SPC-1
	Operator Fails to Switch to EHS Mode	Table A5-1, Item SPC-6
Automatic Depressurization (ADS)	Operator Fails to Manually Depressurize the Reactor Given Auto Failure	Table A6-1, Item 2
Control Rod Drive (CRD)	Operator Fails to Realign CRD for Injection	Table A7-1, Item 1
Standby Liquid Control (SLC)	Operator Fails to Start SLC	Table A10-1, Item 1
	Failure Due to Improper Realignment Following Test	Table A10-1, Item 2
	Failure to Reclose Manual Test Valve F041 after Suction Test	Table A10-1, Item 3
High Pressure Service Water (HPSW)	Operator Fails to Align HPSW for Injection	Table A11-1, Item 1

TABLE B2 – SURVEILLANCE AND CALIBRATION INSPECTION GUIDANCE

The listed components are the risk significant components for which surveillance and/or calibration should minimize failure.

SYSTEM	FAILURE	DISCUSSION
Emergency Electric Power (EPS)	Unit 2 Battery A/B/C/D Fails	Table A1-1, Item 1
	Common Mode Failure(s) of Additional Batteries	Table A1-1, Item 2
	DG E1/E2/E3/E4 Fail to Start or Run or Out for Maintenance	Table A1-1, Item 3
	Common Mode Failure(s) of Additional Diesel Generators	Table A1-1, Item 4
	DG Actuation Fails	Table A1-1, Item 5
	Failure of Diesel Generator Room Coolers	Table A1-1, Item 6
Emergency Service Water (ESW)	ESW Valve MV0498 Fails to Open or Is Out for Maintenance	Table A2-1, Item 1
	ESW Pump A Fails to Start or Run or Is Out for Maintenance	Table A2-1, Item 2
	ESW Pump B Fails to Start or Run or Is Out for Maintenance	Table A2-1, Item 3
	ESW Valve AV21 Fails to Open or Is Out for Maintenance	Table A2-1, Item 4
	ESW Valve AV22 Fails to Open or Is Out for Maintenance	Table A2-1, Item 5
	ESW Valve AV23 Fails to Open or Is Out for Maintenance	Table A2-1, Item 6
	ESW Valve AV24 Fails to Open or Is Out for Maintenance	Table A2-1, Item 7
	ECW Pump Fails to Start or Run or Is Out for Maintenance	Table A2-1, Item 9
	ESW Valve MV0841 Fails to Open or Is Out for Maintenance	Table A2-1, Item 10
	ESW Valves CV515A/B Fail to Open	Table A2-1, Item 11
	ESW Valve CV513 Fails to Open	Table A2-1, Item 12

Low Pressure Core Spray (LPCS)	Common Cause Miscalibration of Reactor Pressure Sensors	Table A3-1, Item 1
	ESW PS-5 Hardware Failures: CV 513 Fails to Open or Manual Valve XV 502 Plugs	Table A3-1, Item 2
	Common Cause Miscalibration of Reactor Water Level Sensors	Table A3-1, Item 4
	MOV 12A/B Fail to Open	Table A3-1, Item 5
	Bus 4160A/B/C/D Power Permissive Sensors Fail	Table A3-1, Item 6
	LPCS Pump A/B/C/D Fail	Table A3-1, Item 7
	ESW PS-8 Fails	Table A3-1, Item 8
	LPCS, LPCI Low Reactor Pressure Sensors C&D/A&B Fail	Table A3-1, Item 9
	MOV 26A/B Fail to Remain Closed	Table A3-1, Item 12
Residual Heat Removal (RHR)	Common Cause Miscalibration of Reactor Pressure Sensors	Table A5-1, Item LPCI-1
	ESW PS-5 Hardware Failures	Table A5-1, Item LPCI-2
	Common Cause Miscalibration of Reactor Water Level Sensors	Table A5-1, Item LPCI-4
	MOV 25A/B Fail to Open	Table A5-1, Item LPCI-5
	ESW PS-8 Fails	Table A5-1, Item LPCI-6
	LPCI, LPCS Low Reactor Pressure Sensors A&B/C&D Fail	Table A5-1, Item LPCI-7
	MOV 154A/B Out for Maintenance or Plugged	Table A5-1, Item LPCI-9
	CV 46A/B Fail to Open	Table A5-1, Item LPCI-11
	Common Cause Miscalibration of High Drywell Pressure Sensors	Table A5-1, Item CS-2
	ESW PS-5 Hardware Failures	Table A5-1, Item CS-3
	MOV 26A/B Fail to Open or Out for Maintenance	Table A5-1, Item CS-4
	MOV 31A/B Fail to Open or Out for Maintenance	Table A5-1, Item CS-5
	RHR Control Logic A/B Circuitry Fails	Table A5-1, Item CS-6
	ESW PS-8 Fails	Table A5-1, Item CS-7
	LPCI, LPCS High Drywell Pressure Sensors A&B/C&D Fail	Table A5-1, Item CS-8
	HPSW MOV 2804A/B and MOV 2486 Fail to Open or Out for Maintenance	Table A5-1, Item CS-9

	MOV 17 Fails to Open or Is Out for Maintenance	Table A5-1, Item SDC-2
	MOV 18 Fails to Open or Is Out for Maintenance	Table A5-1, Item SDC-3
	ESW PS-5 Hardware Failures	Table A5-1, Item SDC-4
	MOV 25A/B Fail to Open or Out for Maintenance	Table A5-1, Item SDC-5
	RHR Control Logic A/B Circuitry Fails	Table A5-1, Item SDC-6
	ESW PS-8 Fails	Table A5-1, Item SDC-7
	CV 46A/B Fail to Open	Table A5-1, Item SDC-11
	ESW PS-5 Hardware Failures	Table A5-1, Item SPC-2
	MOV 34A/B Fail to Open or Out for Maintenance	Table A5-1, Item SPC-3
	MOV 39A/B Fail to Open or Out for Maintenance	Table A5-1, Item SPC-4
	RHR Control Logic A/B Circuitry Fails	Table A5-1, Item SPC-5
	ESW PS-8 Fails	Table A5-1, Item SPC-6
	HPSW MOV 2804A/B and MOV 2486 Fail to Open or Out for Maintenance	Table A5-1, Item SPC-7
Automatic Depressurization (ADS)	Common Mode ADS Valve Failure	Table A6-1, Item 1
	Common Mode Non-ADS Valve Failure	Table A6-1, Item 3
Control Rod Drive (CRD)	MDPB Fails to Start or Run or Is Out for Maintenance	Table A7-1, Item 2
	MDPA Fails to Continue to Run or Is Out for Maintenance	Table A7-1, Item 3
High Pressure Coolant Injection (HPCI)	HPCI Turbine-driven Pump Fails or Is Out for Maintenance	Table A8-1, Item 1
	MOV 19 Fails to Open or Is Out for Maintenance	Table A8-1, Item 2
	MOV 14 Fails to Open or Is Out for Maintenance	Table A8-1, Item 3
	MOV 20 Plugs or Is Out for Maintenance	Table A8-1, Item 6
	PCV 50 Fails or Is Out for Maintenance	Table A8-1, Item 7
	ESW PS-5 Hardware Failures: Check Valve CV 513 Fails to Open or Manual Valve 502 Plugs	Table A8-1, Item 8
	Check Valve CV 18 Fails to Open	Table A8-1, Item 9
	Check Valve CV 32 Fails to Open	Table A8-1, Item 10

	HPCI Flow Controller FIC-23-108 Fails	Table A8-1, Item 11
	Check Valve CV 65 Fails to Open	Table A8-1, Item 12
	MOV 57 or MOV 58 Fail to Open	Table A8-1, Item 13
	Check Valve CV 61 Fails to Open	Table A8-1, Item 14
Reactor Core Isolation Cooling (RCIC)	RCIC Pump Fails to Start or Run or Is Out for Maintenance	Table A9-1, Item 1
	MOV 132 Fails to Open or Is Out for Maintenance	Table A9-1, Item 2
	PCV 23 Fails or Is Out for Maintenance	Table A9-1, Item 3
	MOV 131 Fails to Open or Is Out for Maintenance	Table A9-1, Item 4
	MOV 21 Fails to Open or Is Out for Maintenance	Table A9-1, Item 5
	MOV 20 Plugs or Is Out for Maintenance	Table A9-1, Item 8
	ESW PS-5 Hardware Failures: Check Valve CV 513 Fails to Open or Manual Valve 502 Plugs	Table A9-1, Item 9
	RCIC Flow Controller FIC-91 Fails	Table A9-1, Item 10
	CV 50 Fails to Open	Table A9-1, Item 11
	CV 22 Fails to Open	Table A9-1, Item 12
	MOV 41 or MOV 39 Fails to Open	Table A9-1, Item 13
	CV 40 Fails to Open	Table A9-1, Item 14
	PS-1 Hardware Failure: MOV 18 Plugs or CV 19 Fails to Open	Table A9-1, Item 15
Standby Liquid Control (SLC)	Failure Due to Improper Realignment Following Test	Table A10-1, Item 2
	Failure to Reclose Manual Test Valve F041 after Suction Test	Table A10-1, Item 3
	One of Two Check Valves in Injection Line Fails	Table A10-1, Item 4
	Pump Suction Inlet Valve XV11 Plugs	Table A10-1, Item 5
	Check Valves in Both Pump Discharge Lines Fail to Open	Table A10-1, Item 6
	SLC Pumps Fail to Start or Run or Are Out for Maintenance	Table A10-1, Item 7
High Pressure Service Water (HPSW)	MOV 174/176 Fail to Open	Table A11-1, Item 2
	RHR MOV 25B Fails to Open	Table A11-1, Item 3
	CV 5 Fails to Open	Table A11-1, Item 5

Manual Valve XV 516A Fails to Remain Closed	Table AII-1, Item 6
RHR CV 46B Fails to Open	Table AII-1, Item 7
HPSW MOV 2804A/B and MOV 2486 Fail to Open or Out for Maintenance	Table AII-1, Item 8

TABLE B3 – MAINTENANCE INSPECTION GUIDANCE

The components listed here are significant to risk because of unavailability for maintenance. The dominant contributors are usually frequency and duration of maintenance, with some contribution due to improperly performed maintenance.

SYSTEM	FAILURE	DISCUSSION
Emergency Electric Power (EPS)	Unit 2 Battery A/B/C/D Fails	Table A1-1, Item 1
	Common Mode Failure(s) of Additional Batteries	Table A1-1, Item 2
	DG E1/E2/E3/E4 Fail to Start or Run or Out for Maintenance	Table A1-1, Item 3
	Common Mode Failure(s) of Additional Diesel Generators	Table A1-1, Item 4
	DG Actuation Fails	Table A1-1, Item 5
	Failure of Diesel Generator Room Coolers	Table A1-1, Item 6
Emergency Service Water (ESW)	ESW Valve MV0498 Fails to Open or Is Out for Maintenance	Table A2-1, Item 1
	ESW Pump A Fails to Start or Run or Is Out for Maintenance	Table A2-1, Item 2
	ESW Pump B Fails to Start or Run or Is Out for Maintenance	Table A2-1, Item 3
	ESW Valve AV21 Fails to Open or Is Out for Maintenance	Table A2-1, Item 4
	ESW Valve AV22 Fails to Open or Is Out for Maintenance	Table A2-1, Item 5
	ESW Valve AV23 Fails to Open or Is Out for Maintenance	Table A2-1, Item 6
	ESW Valve AV24 Fails to Open or Is Out for Maintenance	Table A2-1, Item 7
	ECW Pump Fails to Start or Run or Is Out for Maintenance	Table A2-1, Item 9
	ESW Valve MV0841 Fails to Open or Is Out for Maintenance	Table A2-1, Item 10
	ESW Valves CV515A/B Fail to Open	Table A2-1, Item 11
	ESW Valve CV513 Fails to Open	Table A2-1, Item 12

Low Pressure Core Spray (LPCS)	ESW PS-5 Hardware Failures: CV 513 Fails to Open or Manual Valve XV 502 Plugs	Table A3-1, Item 2
	MOV 12A/B Fail to Open	Table A3-1, Item 5
	Bus 4160A/B/C/D Power Permissive Sensors Fail	Table A3-1, Item 6
	LPCS Pump A/B/C/D Fail	Table A3-1, Item 7
	ESW PS-8 Fails	Table A3-1, Item 8
	LPCS, LPCI Low Reactor Pressure Sensors C&D/A&B Fail	Table A3-1, Item 9
	MOV 11A/B Out for Maintenance	Table A3-1, Item 10
	MOV 5A/B/C/D Out for Maintenance	Table A3-1, Item 11
	MOV 26A/B Fail to Remain Closed	Table A3-1, Item 12
Residual Heat Removal (RHR)	ESW PS-5 Hardware Failures	Table A5-1, Item LPCI-2
	MOV 25A/B Fail to Open	Table A5-1, Item LPCI-5
	ESW PS-8 Fails	Table A5-1, Item LPCI-6
	LPCI, LPCS Low Reactor Pressure Sensors A&B/C&D Fail	Table A5-1, Item LPCI-7
	MOV 26A/B Out for Maintenance	Table A5-1, Item LPCI-8
	MOV 154A/B Out for Maintenance or Plugged	Table A5-1, Item LPCI-9
	MOV 39A/B Out for Maitenance	Table A5-1, Item LPCI-10
	CV 46A/B Fail to Open	Table A5-1, Item LPCI-11
	ESW PS-5 Hardware Failures	Table A5-1, Item CS-3
	MOV 26A/B Fail to Open or Out for Maintenance	Table A5-1, Item CS-4
	MOV 31A/B Fail to Open or Out for Maintenance	Table A5-1, Item CS-5
	RHR Control Logic A/B Circuitry Fails	Table A5-1, Item CS-6
	ESW PS-8 Fails	Table A5-1, Item CS-7
	LPCI, LPCS High Drywell Pressure Sensors A&B/C&D Fail	Table A5-1, Item CS-8
	HPSW MOV 2804A/B and MOV 2486 Fail to Open or Out for Maintenance	Table A5-1, Item CS-9
	MOV 39A/B Out for Maintenance	Table A5-1, Item CS-10
	MOV 17 Fails to Open or Is Out for Maintenance	Table A5-1, Item SDC-2
	MOV 18 Fails to Open or Is Out for Maintenance	Table A5-1, Item SDC-3

	ESW PS-5 Hardware Failures	Table A5-1, Item SDC-4
	MOV 25A/B Fail to Open or Out for Maintenance	Table A5-1, Item SDC-5
	RHR Control Logic A/B Circuitry Fails	Table A5-1, Item SDC-6
	ESW PS-8 Fails and Operator Fails to Switch to EHS Mode	Table A5-1, Item SDC-7
	MOV 39A/B Out for Maintenance	Table A5-1, Item SDC-8
	MOV 26A/B Out for Maintenance	Table A5-1, Item SDC-9
	MOV 154A/B Out for Maintenance	Table A5-1, Item SDC-10
	CV 46A/B Fail to Open	Table A5-1, Item SDC-11
	ESW PS-5 Hardware Failures	Table A5-1, Item SPC-2
	MOV 34A/B Fail to Open or Out for Maintenance	Table A5-1, Item SPC-3
	MOV 39A/B Fail to Open or Out for Maintenance	Table A5-1, Item SPC-4
	RHR Control Logic A/B Circuitry Fails	Table A5-1, Item SPC-5
	ESW PS-8 Fails	Table A5-1, Item SPC-6
	HPSW MOV 2804A/B and MOV 2486 Fail to Open or Out for Maintenance	Table A5-1, Item SPC-7
Automatic Depressurization (ADS)	Common Mode ADS Valve Failure	Table A6-1, Item 1
	Common Mode Non-ADS Valve Failure	Table A6-1. Item 3
Control Rod Drive (CRD)	MDPB Fails to Start or Run or Is Out for Maintenance	Table A7-1, Item 2
	MDPA Fails to Continue to Run or Is Out for Maintenance	Table A7-1, Item 3
High Pressure Coolant Injection (HPCI)	HPCI Turbine-driven Pump Fails or Is Out for Maintenance	Table A8-1, Item 1
	MOV 19 Fails to Open or Is Out for Maintenance	Table A8-1. Item 2
	MOV 14 Fails to Open or Is Out for Maintenance	Table A8-1, Item 3
	MOV 57 Is Out for Maintenance	Table A8-1, Item 4
	MOV 17 Is Out for Maintenance	Table A8-1, Item 5
	MOV 20 Plugs or Is Out for Maintenance	Table A8-1, Item 6
	PCV 50 Fails or Is Out for Maintenance	Table A8-1, Item 7
	ESW PS-5 Hardware Failures: Check Valve CV 513 Fails to Open or Manual Valve 502 Plugs	Table A8-1, Item 8

	Check Valve CV 18 Fails to Open	Table A8-1, Item 9
	Check Valve CV 32 Fails to Open	Table A8-1, Item 10
	HPCI Flow Controller FIC-23-108 Fails	Table A8-1, Item 11
	Check Valve CV 65 Fails to Open	Table A8-1, Item 12
	Check Valve CV 61 Fail to Open	Table A8-1, Item 14
Reactor Core Isolation Cooling (RCIC)	RCIC Pump Fails to Start or Run or Is Out for Maintenance	Table A9-1, Item 1
	MOV 132 Fails to Open or Is Out for Maintenance	Table A9-1, Item 2
	PCV 23 Fails or Is Out for Maintenance	Table A9-1, Item 3
	MOV 131 Fails to Open or Is Out for Maintenance	Table A9-1, Item 4
	MOV 21 Fails to Open or Is Out for Maintenance	Table A9-1, Item 5
	MOV 18 Is Out for Maintenance	Table A9-1, Item 6
	MOV 39 Is Out for Maintenance	Table A9-1, Item 7
	MOV 20 Plugs or Is Out for Maintenance	Table A9-1, Item 8
	ESW PS-5 Hardware Failures: Check Valve CV 513 Fails to Open or Manual Valve 502 Plugs	Table A9-1, Item 9
	RCIC Flow Controller FIC-91 Fails	Table A9-1, Item 10
	CV 50 Fails to Open	Table A9-1, Item 11
	CV 22 Fails to Open	Table A9-1, Item 12
	CV 40 Fails to Open	Table A9-1, Item 14
	PS-1 Hardware Failure: MOV 18 Plugs or CV 19 Fails to Open	Table A9-1, Item 15
Standby Liquid Control (SLC)	Failure Due to Improper Realignment Following Test	Table A10-1, Item 2
	Failure to Reclose Manual Test Valve F041 after Suction Test	Table A10-1, Item 3
	One of Two Check Valves in Injection Line Fails	Table A10-1, Item 4
	Pump Suction Inlet Valve XV11 Plugs	Table A10-1, Item 5
	Check Valves in Both Pump Discharge Lines Fail to Open	Table A10-1, Item 6
	SLC Pumps Fail to Start or Run or Are Out for Maintenance	Table A10-1, Item 7
High Pressure Service Water (HPSW)	MOV 174/176 Fail to Open	Table A11-1, Item 2

RHR MOV 25B Fails to Open	Table A11-1, Item 3
RHR MOV 154B Is Out for Maintenance	Table A11-1, Item 4
CV 5 Fails to Open	Table A11-1, Item 5
Manual Valve XV 516A Fails to Remain Closed	Table A11-1, Item 6
RHR CV 46B Fails to Open	Table A11-1, Item 7
HPSW MOV 2804A/B and MOV 2486 Fail to Open or Out for Maintenance	Table A11-1, Item 8

APPENDIX C

PEACH BOTTOM ATOMIC POWER STATION, UNIT 2
RISK-BASED INSPECTION GUIDE

TABLE C-1 - CONTAINMENT AND DRYWELL WALKDOWN

Discussion

Since the drywell is generally inaccessible during normal plant operation, those components listed in the preceding tables which are located either within the drywell or otherwise in the containment are listed below:

Description	ID No.	Location	Desired Position	Actual Position
1. SLC Inboard Stopcheck - Handwheel	CV 17	Containment	Locked Open	
2. RHR System Check Valves	CV 46A CV 46B	Containment	Open Open	
Other Valves in Containment - Not in Top 95% Risk				
(A) RCIC	MV 15 CV 1 HV 1	Containment Containment Containment	Open Open Locked Open	
(B) HPCI	MV 15 CV 1 HV 2	Containment Containment Containment	Open Open Open	
(C) SLC	HV 18	Containment	Locked Open	
(D) RHR	HV 81 A&B MV 18	Containment Containment Containment	Open Open Closed	

APPENDIX D

TABLE D1 - HYPOTHETICAL "PEACH BOTTOM" MATRIX FOR FRONT LINE DEPENDENCIES ON SUPPORT SYSTEMS

FRONT LINE SYSTEMS & TRAIN

		RODS	SLC	HPCI	RCIC	CS		LPCI		MFW	CRD		ADS		SRV		DW Coolers		DW Spray		RHR		RWCU/
						A	B	A	B		A	B	A	B	A	B	A	B	A	B	A	B	PCS
Offsite Power			C	C(9)	C(9)	C	C	C	C	B	C	C					C	C	C	C	C	C	B
Emer AC(1)	I		C(3)		C(9)	B		B			B						B		B		B		
	II		C(3)	C(9)			B		B			B						B		B		B	
	III							C	C										C	C	C	C	
DC	I				B	B		B			B		B		C				B		B		
	II			B			B		B			B		B		C				B		B	
III						C	C										C	C	C	C			
RBSW(5)											C	C					B	B	C	C	C	C	B
Instrumentation (7)			C	C	C	C	C	C	C	C	C	C	C	C	C	C	C	C	C	C	C	C	C
Air/Gas										C(8)	B(6)	B	B	B	C	C							
Closed Loop C.W (RBCLCW)											C	C					B	B					B
TBSW (2) (4)										C	C	C					C	C	C	C	C	C	C
Room Cooling (8)																							

A = Inter-dependent
B = Complete dependence
C = Partial or delayed dependence
No Entry = No dependence

(1) = Assumes Loss of Offsite Power (LOOP)

(2) = Assumes loss of Reactor Bulding Service Water (RBSW)

(3) = Loss of Div. I or Div. II will reduce flow by 50% to 43 gpm.

(4) = Turbine Building Service Water (TBSW) can be cross-tied with RBSW.

(5) = The Fire Protection System (FPS) can be aligned to the RBSW.

(6) = Scram on loss of air.

(7) = System cap run w/o instr; however, a minimal level of instr. in desirable.

(8) = See discussion in Sect. 5.

(9) = Loop level pump is AC powered.

APPENDIX D (Cont'd)

TABLE D2 - HYPOTHETICAL "PEACH BOTTOM" MATRIX FOR THE SUPPORT SYSTEM DEPENDENCIES ON OTHER SUPPORT SYSTEMS

			DG			DC			RBSW				RBCLCW			ROOM
		OSP	I	II	III	I	II	III	A	B	INST	AIR	A	B	FPS	COOLING
Offsite Power (OSP)(1)		A				C	C	C	C	C	C	B(6)	C	C		C
Emergency AC (DG)(2)(4)	I		A			(3)			A		C(5)		B			B
	II			A			(3)			A	C(5)			B		B
	III				A			(3)	C	C	C(5)		C	C		C
DC Power	I		B			A			C		C					
	II			B			A			C	C					
	III				B			A	C	C	C					
RBSW (4)	A		C	C	C				A				B	C		B
	B		C	C	C					A			C	B		B
Instrumentation		C	C	C	C	C	C	C	C	C	A	C	C	C	C	C
Air/Gas												A				
Closed Loop C.W. (RBCLCW)	A												A			
	B													A		
Fire Prot. Sys. (FPS)									(7)	(7)					A	
Room Cooling			C	C	C											A

A = Inter-dependent
B = Complete dependence
C = Partial or delayed dependence
No Entry = No dependence

(1) = "C"s Assume ability to power from Emerg. Bus.

(2) = Assumes LOOP has occurred.

(3) = Batt. required for DG start. Each DG has its own battery for starting.

(4) = Each diesel draws SW from both loops.

(5) = W/O emerg., AC rod pos. indication, Rx water level, & Reactor pressure indication are available.

(6) = Instrument N_2 System is partially dependent.

(7) = Fire Protection System (FPS) can be used to supply RBSW.

ATTACHMENT 2

INDIAN POINT 3

SYSTEMS IMPORTANCE RANKING

BASED ON NSPKTR II

James C. Higgins

Plant Systems and Equipment Analysis Group
Engineering Technology Division
Department of Nuclear Energy

Rev. 0 - January 1986
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INDIAN POINT 3
SYSTEMS IMPORTANCE RANKING
BASED ON NSPKTR II

1. General Discussion

The computer program, NSPKTR II, calculates the importance of each of the 51 "systems" that constitute the top events of the 13 Indian Point event trees. These events are not full systems in the traditional sense, and hence must be combined in some fashion to obtain a ranking of the standard systems in the plant. This report describes how the events are combined to form the standard plant systems and provides the ordered list of systems. NSPKTR provides importances for the top events using several different importance measures, both with and without offsite health effects included. For developing the actual ranking we have selected the Inspection Importance Measure with health effects. For closely related systems comparison we also compute the full systems' importance using the Birnbaum Importance Measure, again including offsite health effects.

Table 1 provides a list from NSPKTR II of the 51 top events of the Indian Point-3 event trees.

2. Combination of Events

The below discussion assumes a knowledge of importance calculations based on the minimal cutset representation of risk, as described in Section 3.3 of NUREG/CR-4565.

The combination of the top events from the event trees into standard systems is straightforward in most cases. As an example, consider the events HH1 (#12) and HH2 (#15). HH1 appears in Event Tree 2, Medium LOCA, and consists of two out of three high head injection pumps automatically starting and delivering water to the reactor coolant system. HH2 appears in Event Tree 3, small LOCA, and Event Tree 4, Steam Generator Tube Rupture. HH2 consists of one out of three high head injection pumps starting and delivering water tothe reactor coolant system. Thus it is seen that HH1 and HH2 both represent the High Pressure Injection (HPI) System, are in separate event trees, and would not appear in the same minimal cutsets. Therefore, their importances (either Birnbaum or Inspection) can be added together to get a system importance for HPSI. Other top events are similarly combined into systems, e.g., R1 through R4 are combined into the Recirculation System.

While most of the events were easily assigned to systems, as described, some events were not so clearly assignable. For example, the IPPSS used OP or operator functions as events, which consisted of the control room operator taking certain actions involving several systems. It was determined that the importances of these events should be assigned to the system operated but had to be apportioned differently depending on the importance measure used. For Inspection Importance their importance should be split among the involved systems in proportion to the individual system's unavailabilities. For the Birnbaum Importance the full value of each event importance should be assigned to each involved system. Figure 1 provides a derivation which illustrates this apportionment. Figure 2 provides the results of the Inspection Importance apportionment for all OP functions.

A couple of system assignments were not completely straightforward and merit further discussion, namely, the Reactor Coolant System (RCS) and the Secondary System. A number of events and OP functions used portions of the RCS, primarily the power-operated relief valves (PORV). Hence, these were assigned to the RCS/PORV. A number of other events addressed the main steam isolation valves (MSIV), the main turbine trip (TT) system, or other parts of the secondary side systems. These were all grouped together. This particular grouping was found useful for ranking purposes. In any case, the details at the component level of what is important will be listed in the individual system Appendix, regardless of which system you choose to assign an event.

The actual combination of pertinent events into a system importance is presented in Table 2 for Inspection Importance and Table 3 for Birnbaum Importance. Table 4 then summarizes the system rankings based on these two tables.

The importance values of support systems, service water and component cooling water were not explicitly calculated, since this would require reconfiguring the fault trees and event trees of the IPPSS, which would require considerable time. In general, however, PRAs have found support systems to be quite important since their failure leads to failure of a number of front line systems. Appendix B of NUREG-1050, "Probabilistic Risk Assessment (PRA) Reference Document," concludes from studies of about 15 PWR PRAs that service water is among the 10 most important PWR systems.

3. Limitations

When using the ordered list of systems described above, one must recognize that it is not necessarily a precise ordering. A number of factors contribute to the uncertainty. Listed below are some of these factors:

1. The uncertainties inherent in the IPPSS itself are naturally reflected in the systems list.

2. The event trees for Steam Generator Tube Rupture and Anticipated Transient Without Scram were modified after the PRA was completed, and not all areas affected were modified. Also, all pertinent numbers are not documented in the PRA.

3. The choice of importance measure used for the ranking somewhat affects the order.

4. The modeling of electric power states and success states creates a few difficulties in the rankings.

5. Not all details of the IPPSS calculations were available.

6. Support systems were not rigorously ranked.

7. The method of grouping the events into systems could create some misimpressions.

4. Summary

Despite the above limitations, a useful ordered list of the Indian Point-Unit 3 systems has been developed and is presented in Table 4 and in the main text of this report. There is basically good agreement between the two importance measures used. There is also good agreement between the list of ordered systems and engineering judgment. The list can be very useful for general priority setting. It should be realized that systems close together on the list do not differ much in their importance ranking.

Table 1. Menu 3: Event Tree Systems/Functions/Actions

1. E.PWR W.O.39	20. RCTR TRIP * SIS	38. OP51 DPRS*MKP*HH WO L3
2. RFUEL W.STR.TK	21. S.I.ACT.SIG.SA2	39. E.PWR TT LOP
3. S.I. ACT SIG SA1	22. RCTR TRIP K2 SGTR	40. RCP SEAL LOCA
4. L.P. INJ/ACC	23. ATWS ACT * SEC COOL	41. OP52 DPRS*MKP*HH*L3*SL
5. CNMT SPRY	24. R3 RECIR COOL	42. OP53 DPS*MP*L3 WO HH*SL
6. NaOH	25. FAN COOLERS	43.
7. R.C. FAN COOL	26. SL2 NO SEC LK WO OP4	44.
8. L.H. RECIR	27. MSIV TRIP MS1	45. PWR > 80%
9. RECIR SPRY	28.	46. TT2/MSIV CLS
10. OP41 OP CN FLO W L3	29. EP1 EL PWR > 1 MIN	47. AUX FDWTR*SEC COOL L2
11. L.P.2 INJ	30. TURBINE TRIP TT1	48. RODS IN < 1 MIN
12. HH1 INJ	31. MSIV TRIP MS2	49. ATWS PRES RLF
13. OP42 OP CN FLO WO L3	32. BLEED*FEED OP2	50. SAF INJ OP
14. RCTR TRIP K3	33. EP2 EL PWR > 1 HR	51. MAN DENRG RCCA FALL
15. HH2 INJ	34. OP3 STAB TRNST	52. SECURE PR
16. AUX FDWTR/S.COOL L1	35. EP3 EL PWR > 3 HR	53. NO SEC LKG TO ATMOS
17. BLEED*FEED OP1	36. P50 DPS*MP*HH*L3*OP4	54.
18. R2 RECIR COOL	37. PWR RUNBK K5	55. R4 LONG TERM COOL
19. SLI NO SEC LK W OP4		56. ALL SYSTEMS (1-55)

Table 2. Inspection Importance of Systems

System Name	IPPSS Event Code	NSPKTR-II Item No. (per Table 1)	NSPKTR-II Inspection Importance
1. Electric Power	Done by power states	1	5.86 - 7*
		39	8.9 - 9
		33	< 10^{-10}
		29	< 10^{-10}
		35	< 10^{-10}
			5.95 - 7
2. RPS	K1	Not used	-
	K2	22	1.8 - 10
	K3	14	3.9 - 10
	K4	Not used	-
	K5	37	< 10^{-10}
	OP5	51	3.4 - 10
	OP6	48	3.17 - 7
	.1 (S)	20	< 10^{-10}
			3.17 - 7
3. HPI	HH1	12	9.3 - 10
	HH2	15	2.69 - 7
	.1 (OP2)	32	.1(1.5-9) = 1.5-10
	.1 (OP42)	13	< 10^{-10}
	SO	50	< 10^{-10}
			2.69 - 7
4. Secondary System (MSIV)	MS1	27	1.76 - 7
	MS2	31	< 10^{-10}
	TT1	30	< 10^{-10}
	TT2	46	1.92 - 10
			1.76 - 7
5. Recirculation System	R1	8	5.71 - 8
	R2	18	4.81 - 8
	R3	24	2.23 - 9
	R4	55	< 10^{-10}
			1.07 - 7
6. RCS/(PORV)	PR1	49	< 10^{-10}
	PR2	52	< 10^{-10}
	OP1	17	4.81 - 8
	.9 (OP2)	32	.9(1.5-9) = 1.35-9
	OP3	34	< 10^{-10}
	OP41	10	< 10^{-10}
	.9 (OP42)	13	< 10^{-10}
	SL	53	< 10^{-10}
			4.95 - 8

*5.86 - 7 is 5.86×10^{-7}

Table 2 (Cont'd.)

System Name	IPPSS Event Code	NSPKTR-II Item No. (per Table 1)	NSPKTR-II Inspection Importance
7. AFW	L1	16	1.21 - 8
	L2	47	8.27 - 9
	L3	23	5 - 10
			2.08 - 8
8. Safeguards Actuation (SAS)	SA1	3	1.20 - 8
	SA2	21	$< 10^{-10}$
	.9(S)	20	$< 10^{-10}$
			1.20 - 8
9. LPIS	.5(LP1)	4	7.75 - 9
	LP2	11	4.18 - 9
			1.19 - 8
10. Accumulator	.5(LP1)	4	7.75 - 9
11. Containment Spray	CS	5	$< 10^{-10}$
	RS	9	$< 10^{-10}$
12. RWST	TK	2	$< 10^{-10}$
13. Cont. Fan Coolers	CF1	7	$< 10^{-10}$
	CF2	25	$< 10^{-10}$
14. NaOH System	NA	6	$< 10^{-10}$

Note: OP 50-53, all $< 10^{-10}$

Table 3. Birnbaum Importance of Systems

Main System Name	IPPSS Event Code	NSPKTR-II Item No. (per Table 1)	NSPKTR-II Birnbaum Importance
1. SAS	SA1	3	1.93 - 3
	SA2	21	< 10^{-10}
	S	20	3.27 - 2
			3.46 - 2
2. RPS	K2	22	4.58 - 6
	K3	14	1 - 5
	K5	37	< 10^{-10}
	OP5	51	6 - 10
	OP6	48	2 - 6
	S	20	3.27 - 2
			3.27 - 2
3. HPSI	HH1	12	5 - 6
	HH2	15	1.92 - 3
	OP2	32	4 - 6
	OP42	13	< 10^{-10}
			1.92 - 3
4. AFW	L1	16	8.09 - 4
	L2	47	7.5 - 7
	L3	23	4.58 - 6
			8.14 - 4
5. Recirculation	R1	8	1.08 - 5
	R2	18	1.17 - 5
	R3	24	5.4 - 7
	R4	55	< 10^{-10}
			2.3 - 5
6. Secondary Sys.(MSIV)	MS1	27	1.17 - 5
	MS2	31	4 - 10
	TT1	30	2.3 - 6
	TT2	46	1.1 - 6
			1.51 - 5
7. RCS (PORV)	PR1	49	< 10^{-10}
	PR2	52	< 10^{-10}
	OP1	17	1.09 - 5
	OP2	32	3.4 - 7
	OP3	34	4 - 9
	OP41	10	< 10^{-10}
	OP42	13	< 10^{-10}
			1.12 - 5

Table 3 (Cont'd.)

Main System Name	IPPSS Event Code	NSPKTR-II Item No. (per Table 1)	NSPKTR-II Birnbaum Importance
8. RWST	TK	2	1.11 - 5
9. LPIS	LP1	4	5.16 - 6
	LP2	11	5.15 - 6
			7.03 - 5
10. Accumulator	LP1	4	5.16 - 6
11. Electric Power		1	5.86 - 7
		39	1 - 8
		33	$< 10^{-10}$
		29	$< 10^{-10}$
		35	$< 10^{-10}$
			6.0 - 7
12. Containment Spray	CS	5	3.94 - 7
	RS	9	$< 10^{-10}$
			3.94 - 7
13. Cont. Fan Coolers	CF1	7	4.19 - 8
	CF2	25	6.57 - 8
			1.07 - 7
14. NaOH	NA	6	$< 10^{-10}$

Table 4. System Ranking

	Inspection Importance (HE)	Birnbaum Importance (HE)
1.	Electric Power	SAS
2.	RPS	RPS
3.	HPSI	HPSI
4.	Secondary System (MSIV)	AFW
5.	Recirculation	Recirculation
6.	RCS (PORV)	Secondary System (MSIV)
7.	AFW	RCS (PORV)
8.	SAS	RWST
9.	LPIS	LPIS
10.	Accumulator	Accumulator
11.	Containment Spray	Electric Power
12.	RWST	Containment Spray
13.	Cont. Fan Coolers	Cont. Fan Coolers
14.	NaOH	NaOH

Note: Support systems (Service Water and Component Cooling Water) are not included. However, based on generic PRA studies (NUREG-1050), Service Water (at least) would be among the top systems.

Figure 1. Apportionment of OP Functions

Consider the minimal cutset representation of Risk, R, from the text of the Indian Point 3 report (NUREG/CR-4565)

$$R = P_1 A+B$$

Say an OP function consists of 3 systems, S_1, S_2, and S_3 functioning. Then the probability of failure of the OP function P(OP) can be given as follows:

$$P(OP) = P(S_1 + S_2 + S_3) = (S_1 + S_2 + S_3)$$

$$= P(S_1) + P(S_2) + P(S_3) \text{ (using the small probabilities assumptions)}$$

or for short

$$OP = S_1 + S_2 + S_3,$$

since the failure of any involved system fails the OP function. Here we consider human failure to operate a system as a failure of the system.

For simplicity of derivation, let us assume that only one minimal cutset to core melt contains OP, and that cutset is $OP \cdot P_2 \cdot P_3$. Results can be easily extended to other cutsets.

Risk is then $R = C_i \cdot OP \cdot P_2 \cdot P_3 + B$, where B are those terms not containing OP. For the Birnbaum Importance Measure (I^B):

$$I^B_{OP} = \frac{\partial R}{\partial_{OP}} = C_i P_2 P_3$$

$$I^B_{S_1} = \frac{\partial R}{\partial S_1} = \frac{\partial}{\partial S_1} [C_i (S_1 + S_2 + S_3) P_2 \cdot P_3 + B]$$

$$= C_i P_2 P_3$$

Similarly $I^B_{S_2} = I^B_{S_3} = C_i P_2 P_3$

Thus, for the Birnbaum importance measure, the full value of I^B_{OP} should be assigned to each involved system, S.

For the Inspection Importance Measure (I^I):

$$I^I_{OP} = P(OP) \cdot I^B_{OP} = OP \cdot C_i \cdot P_2 \cdot P_3 = (S_1 + S_2 + S_3)\ C_i \cdot P_2 \cdot P_3$$

$$I^I_{S_1} = P(S_1)\ I^B_{S_1} = S_1\ (C_i \cdot P_2 \cdot P_3)$$

Similarly:

$$I^I_{S_2} = S_2\ C_i \cdot P_2 \cdot P_3$$

$$I^I_{S_3} = S_3\ C_i \cdot P_2 \cdot P_3$$

From this we can see that:

$$I^I_{OP} = I^I_{S_1} + I^I_{S_2} + I^I_{S_3}$$

Also: $I^I_{S_1} = \dfrac{S_1}{(S_1 + S_2 + S_3)} \cdot I^I_{OP}$; where $\dfrac{S_1}{(S_1 + S_2 + S_3)}$ is the fractional part of the unavailability of OP that S_1 constitutes.

Thus for the inspection importance measure, I^I_{OP} should be split up among systems constituting OP in the ratio of their unavailabilities. Table 4 illustrates how this was done for the various OP functions.

Figure 2. Split of OP Functions for Inspection Importance

Code	Title	Fraction to System
OP-1	Primary Cooling Bleed	1.0 to RCS
OP-2	Primary Bleed and Feed	0.9 to RCS; 0.1 to HPI
OP-3	Operator Stabilizes Transient	1.0 to RCS
OP-4	Operator Controls Break Flow	
OP-41	Given success of AFW	1.0 to RCS
OP-42	Given failure of AFW	.9 to RCS; .1 to HPI
OP-5	Rods in by 1 minute	1.0 to RPS
OP-50	Depressurization and Makeup	.5 to HPI; .5 to AFW
OP-51	" " " "	.9 to RCS; .1 to HPI
OP-52*	" " " "	.4 to RCS; .3 to AFW; .3 to Sec.Sys.
OP-53*	" " " "	.4 to LPIS; .3 to AFW; .3 to Sec.Sys.

*Split assumed for these two functions due to lack of data in IPPSS.

A good example taken from a non-PRA source, the IDCOR Individual Plant Evaluation Method Applied to the Shoreham Nuclear Power Station (IDCOR-DKT-50322), is shown in Attachment 1 as Appendix D.

ATTACHMENT 3

SAMPLE IMPORTANCE CALCULATIONS

PEACH BOTTOM UNIT 2

LOW PRESSURE CORE SPRAY SYSTEM

PEACH BOTTOM UNIT 2

SYSTEM: LOW PRESSURE CORE SPRAY SYSTEM

C.S. #	Cutset Events *	Event Probabilities	Sum of Event Probabilities	Fraction of Cutsets	Cutset Probability	† % of System A
1	ESF-XHE-NC-RXPRS	1.0 E-3	1.0 E-3	1.0	1.0 E-3	70.42
5	ESW-PSF-LF-5	1.8 E-4	1.8 E-4	1.0	1.8 E-4	12.68
7	ESF-XHE-FO-LPSAT	1.0 E-1	1.01 E-1	0.99	1.0 E-4	6.97
	ESF-XHE-MC-LEVEL	1.0 E-3		0.01		0.070
	ESF-ASP-NOHDPLT	1.0				
60	LCS-PSF-HW-INJ13	3.8 E-3	7.6 E-3	0.5	9.0 E-6	0.317
	LCS-PSF-HW-INJ27	3.8 E-3		0.5		0.317
72	LCS-PSF-HW-INJ13	3.8 E-3	6.3 E-3	0.60	6.0 E-6	0.254
	ESF-PWR-FC-4160B	2.5 E-3		0.40		0.169
73	LCS-PSF-HW-INJ13	3.8 E-3	7.03 E-3	0.54	6.0 E-6	0.228
	LCS-MDP-FS-2DP37	3.23 E-3		0.46		0.194
74	LCS-PSF-HW-INJ27	3.8 E-3	6.3 E-3	0.60	6.0 E-6	0.254
	ESF-PWR-FC-4160C	2.5 E-3		0.40		0.169
75	SLCS-PSF-HW-INJ27	3.8 E-3	7.03 E-3	0.54	6.0 E-6	0.228
	LCS-MDP-FS-2CP37	3.23 E-3		0.46		0.194
76	LCS-PSF-HW-INJ13	3.8 E-3	6.3 E-3	0.60	6.0 E-6	0.254
	ESF-PWR-FC-4160D	2.5 E-3		0.40		0.169
77	LCS-PSF-HW-INJ13	3.8 E-3	7.03 E-3	0.54	6.0 E-6	0.228
	LCS-MDP-FS-2BP37	3.23 E-3		0.46		0.194
78	LCS-PSF-HW-INJ27	3.8 E-3	6.3 E-3	0.60	6.0 E-6	0.254
	ESF-PWR-FC-4160A	2.5 E-3		0.40		0.169
79	LCS-PSF-HW-INJ27	3.8 E-3	7.03 E-3	0.54	6.0 E-6	0.228
	LCS-MDP-FS-2AP37	3.23 E-3		0.46		0.194
98	ESW-PSF-LF-8	3.0 E-5	1.0 E-1	0.0003	4.2 E-6	0.0001
	ESW-XHE-FO-EHS	1.0 E-1		0.9997		0.296
99	LCS-MDP-FS-2BP37	3.23 E-3	5.73 E-3	0.56	4.0 E-6	0.158
	ESF-PWR-FC-4160C	2.5 E-3		0.44		0.124

* The SETS fault tree level cutsets as provided to BNL were calcuated assuming loss of offsite power occurred with a probability of 1.0. Therefore, LOSP appeared predominantly throughout the cutsets. Since this distorted the results, those cutsets containing LOSP as a basic event were ignored.

† $\overline{A}$ = 1.4eE-3 failures/demand

PEACH BOTTOM UNIT 2

SYSTEM: LOW PRESSURE CORE SPRAY SYSTEM

C.S. #	Cutset Events	Event Probabilities	Sum of Event Probabilities	Fraction of Cutsets	Cutset Probability	% of System $\overline{A}$
100	ESF-PWR-FC-4160D	2.5 E-3	5.0 E-3	0.5	4.0 E-6	0.141
	ESF-PWR-FC-4160A	2.5 E-3		0.5		0.141
						(95.014%)
101	LCS-MDP-FS-2CP37	3.23 E-3	5.73 E-3	0.56	4.0 E-6	0.158
	ESF-PWR-FC-4160D	2.5 E-3		0.44		0.124
102	ESF-ASP-FC-PL52C	2.5 E-3	5.0 E-3	0.5	4.0 E-6	0.141
	ESF-ASP-FC-PL52D	2.5 E-3		0.5		0.141
103	ESF-PWR-FC-4160C	2.5 E-3	5.0 E-3	0.5	4.0 E-6	0.141
	ESF-PWR-FC-4160B	2.5 E-3		0.5		0.141
104	ESF-PWR-FC-4160C	2.5 E-3	5.0 E-3	0.5	4.0 E-6	0.141
	ESF-PWR-FC-4160D	2.5 E-3		0.5		0.141
105	LCS-MDP-FS-2AP37	3.23 E-3	5.73 E-3	0.56	4.0 E-6	
	ESF-PWR-FC-4160D	2.5 E-3		0.44		0.124
106	ESF-ASP-FC-PL52A	2.5 E-3	5.0 E-3	0.5	4.0 E-6	0.141
	ESF-ASP-FC-PL52B	2.5 E-3		0.5		0.141
107	LCS-MDP-FS-2DP37	2.5 E-3	5.73 E-3	0.56	4.0 E-6	0.158
	ESF-PWR-FC-4160C	2.5 E-3		0.44		0.124
148	ESW-PSF-LF-8	3.0 E-5	1.103 E-2	0.003	2.5 E-6	0.005
	ESW-PSF-LF-19	1.1 E-2		0.997		0.174
153	LCS-PSF-HW-INJ27	3.8 E-3	7.03 E-3	0.54	2.4 E-6	0.091
	LCS-MDP-FR-2AP37	3.23 E-3		0.46		0.078
						(97.839%)
154	LCS-PSF-HW-INJ13	3.8 E-3	7.03 E-3	0.54	2.4 E-6	0.091
	LCS-MDP-FR-2BP37	3.23 E-3		0.46		0.078
155	LCS-PSF-HW-INJ27	3.8 E-3	7.03 E-3	0.54	2.4 E-6	0.091
	LCS-MDP-FR-2CP37	3.23 E-3		0.46		0.078
156	LCS-PSF-HW-INJ13	3.8 E-3	7.03 E-3	0.54	2.4 E-6	0.091
	LCS-MDP-FR-2DP37	3.23 E-3		0.46		0.078
160	ESW-MDP-FS-ESWA	5.3 E-3	7.4 E-3	0.72	2.24 E-6	0.114
	ESW-PSF-LF-02	2.1 E-3		0.28		0.044
	ECW-XHE-FO-ECWPP	1.0				
161	ESW-PSF-LF-01	2.1 E-3	7.4 E-3	0.28	2.24 E-6	0.044
	ESW-MDP-FS-ESWB	5.3 E-3		0.72		0.114
	ECW-XHE-FO-ECWPP	1.0				

PEACH BOTTOM UNIT 2

SYSTEM: LOW PRESSURE CORE SPRAY SYSTEM

C.S. #	Cutset Events	Event Probabilities	Sum of Event Probabilities	Fraction of Cutsets	Cutset Probability	% of System $\bar{A}$
162	LCS-PSF-HW-INJ27 LCS-MDP-MA-2AP37	3.8 E-3 1.86 E-3	5.66 E-3	0.67 0.33	2.1 E-6	0.099 0.049
163	LCS-PSF-HW-INJ13 LCS-MDP-MA-2BP37	3.8 E-3 1.86 E-3	5.66 E-3	0.67 0.33	2.1 E-6	0.099 0.049
164	LCS-PSF-HW-INJ27 LCS-MDP-MA-2CP37	3.8 E-3 1.86 E-3	5.66 E-3	0.67 0.33	2.1 E-6	0.099 0.049
165	LCS-PSF-HW-INJ13 LCS-MDP-MA-2DP37	3.8 E-3 1.86 E-3	5.66 E-3	0.67 0.33	2.1 E-6	0.099 0.049
166	ESW-PSF-LF-8 ESW-ACT-FA-EHS	3.0 E-5 6.25 E-3	6.28 E-3	0.005 0.995	2.1 E-6	0.0007 0.147
177	LCS-MDP-FR2DP37 ESF-PWR-FC-4160C	2.13 E-3 2.5 E-3	4.63 E-3	0.46 0.54	1.6 E-6	0.052 0.062
178	LCS-MDP-FR-2BP37 ESF-PWR-FC-4160C	2.13 E-3 2.5 E-3	4.63 E-3	0.46 0.54	1.6 E-6	0.052 0.062
179	LCS-MDP-FR-2AP37 ESF-PWR-FC-4160D	2.13 E-3 2.5 E-3	4.63 E-3	0.46 0.54	1.6 E-6	0.052 0.062
180	LCS-MDP-FR-2CP37 ESF-PWR-FC-4160D	2.13 E-3 2.5 E-3	4.63 E-3	0.46 0.57	1.6 E-6	0.052 0.062
190	LCS-MDP-MA-2AP37 ESF-PWR-FC-4160C	1.86 E-3 2.5 E-3	4.36 E-3	0.43 0.57	1.4 E-6	0.042 0.056
191	LCS-MDP-MA-2DP37 ESF-PWR-FC-4160C	1.86 E-3 2.5 E-3	4.36 E-3	0.43 0.57	1.4 E-6	0.042 0.056
192	LCS-MDP-MA-2BP37 ESF-PWR-FC-4160C	1.86 E-3 2.5 E-3	4.36 E-3	0.43 0.57	1.4 E-6	0.042 0.056
193	LCS-MDP-MA-2CP37 ESF-PWR-FC-4160D	1.86 E-3 2.5 E-3	4.36 E-3	0.43 0.57	1.4 E-6	0.042 0.056
206	ESW-PSF-LF-8 ESW-MOV-FT-0498	3.0 E-5 3.75 E-3	3.78 E-3	0.008 0.992	1.26 E-6	0.0007 0.088
234	LCS-PSF-HW-INJ13 LCS-MOV-MA-MV5D	3.8 E-3 7.99 E-4	4.6 E-3	0.826 0.174	9.0 E-7	0.052 0.011
235	LCS-PSF-HW-INJ27 LCS-MOV-MA-MV5C	3.8 E-3 7.99 E-4	4.6 E-3	0.826 0.174	9.0 E-7	0.052 0.011

PEACH BOTTOM UNIT 2

SYSTEM: LOW PRESSURE CORE SPRAY SYSTEM

C.S. #	Cutset Events	Event Probabilities	Sum of Event Probabilities	Fraction of Cutsets	Cutset Probability	% of System $\bar{A}$
236	LCS-PSF-HW-INJ13 LCS-MOV-MA-MV11B	3.8 E-3 7.99 E-4	4.6 E-3	0.826 0.174	9.0 E-7	0.052 0.011
237	LCS-PSF-HW-INJ13 LCS-MOV-MA-MV5B	3.8 E-3 7.99 E-4	4.6 E-3	0.826 0.174	9.0 E-7	0.052 0.011
238	LCS-PSF-HW-INJ27 LCS-MOV-MA-MV5A	3.8 E-3 7.99 E-4	4.6 E-3	0.826 0.174	9.0 E-7	0.052 0.011
239	LCS-MOV-MA-MV11A LCS-PSF-HW-INJ27	7.99 E-4 3.8 E-3	4.6 E-3	0.174 0.826	9.0 E-7	0.011 0.052

MULTIPLE OCCURRENCES

Event	Cutset #	% of System $\overline{A}$
LCS-PSF-HW-INJ13	60	0.317
	72	0.254
	73	0.228
	76	0.254
	77	0.228
	154	0.091
	156	0.091
	163	0.099
	165	0.099
	234	0.052
	236	0.052
	237	0.052
		1.187
LCS-PSF-HW-INJ27	60	0.317
	74	0.254
	75	0.228
	78	0.254
	79	0.228
	153	0.091
	155	0.091
	162	0.099
	164	0.099
	235	0.052
	238	0.052
	239	0.052
		1.187

Event	Cutset #	% of System $\overline{A}$
ESF-PWR-FC-4160B	72	0.169
	103	0.141
		0.310
LCS-MDP-FS-2DP37	73	0.194
	107	0.158
		0.352
ESF-PWR-FC-4160C	74	0.169
	99	0.124
	103	0.141
	104	0.141
	107	0.124
	177	0.062
	178	0.062
	191	0.056
	192	0.056
		0.935
LCS-MDP-FS-2CP37	75	0.194
	101	0.158
		0.352
ESF-PWR-FC-4160D	76	0.169
	100	0.141
	101	0.124
	104	0.141
	105	0.124
	179	0.062
	180	0.062
	190	0.056
	193	0.056
		0.935
ESF-PWR-FC-4160A	78	0.169
	100	0.141
		0.310
LCS-MODP-FS-2B37	77(79)	0.194
(2AP37)	99(105)	0.158
		0.352 each

Event	Cutset #	% of System $\bar{A}$
ESW-PSF-LF-8	98	0.0001
	148	0.0005
	166	0.0007
	206	0.0007
		0.0020 each
LCS-MDP-FR-2AP37	153(154)[155](156)	0.078
[2BP37]	179(178)[180](177)	0.052
[2CP37]		
(2DP37)		
		0.130 each
LCS-MDP-MA-2AP37	162(163)[164](165)	0.049
(1BP37)	190(192)[193](191)	0.042
[2CP37]		
(2DP37)		
		0.091 each