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

December 19, 1995

MEMORANDUM TO: Suzanne C. Black, Chief
Quality Assurance and Maintenance Branch
Division of Reactor Controls and Human Factors

FROM: Joseph J. Petrosino
Quality Assurance Section
Quality Assurance and Maintenance Branch

SUBJECT: TRIP REPORT - NOVEMBER 17, 1995, OBSERVATION OF THE CONDUCT OF THE GGNS GRADED QUALITY ASSURANCE PROGRAM EXPERT PANEL THAT WAS CONVENED TO CATEGORIZE SAFETY RELATED COMPONENT HANDLING (TAC M92450)

On November 17, 1995, members of the Office of Nuclear Reactor Regulation (NRR), and Region IV observed the conduct of the sixth meeting (list of attendees attached) of the Grand Gulf Nuclear Generating Station's (GGNS) expert panel that discussed and reached agreement on certain elements of GGNS' QA criteria for the "procurement of low safety significance components," copy of which is attached.

The expert panel had previously completed the development of the technical criteria to identify plant systems and components that are safety-significant. The panel had approved for use an October 11, 1995, EPRI paper titled "Criteria for Determining the Safety Significance of Plant Structures, Systems, and Components for the Grand Gulf Graded QA Program," (attached). The NRC staff has not reviewed or endorsed the criteria in that document; however, the staff is currently reviewing that document and will respond to GGNS subsequently by letter.

During this meeting, the NRC staff observed the expert panel as they discussed and modified the QA criteria which is to be applied to components that GGNS determines to be safety-related, low safety-significance components (LSSCs). The expert panel discussed numerous elements of the QA criteria for the procurement of low safety significance components, made some modifications and agreed upon each element before they went on to the next issue.

For example, the first agreement that the panel reached was the elimination of the term "important to safety" that was used in the attached "Quality Assurance Criteria." The expert panel discussed how to handle situations where non-safety related components have been designated as safety significant. The panel was concerned about whether existing warehouse stocks should and could be utilized for those applications. To solve the concern about existing stocks, the panel concluded that unless a deficiency arose that would trigger a corrective action review, that the existing stock would be continued to be used and engineering evaluations to evaluate the adequacy of the existing stock would be performed as necessary. In addition the panel ascertained

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× L-4-1, Pt. 50, QA

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that future procurement of non-safety related safety significant items should be in accordance with the Appendix B procurement program or the site component dedication program.

The expert panel also discussed the specific modifications to the GGNS Quality Assurance Criteria. For example, in the area of Criterion III, "Design Control," the panel determined that it may need to do some source verification depending upon which critical characteristics would be verified on-site. In those cases where the item was procured as commercial-grade, GGNS will assume 10 CFR Part 21 responsibility. In the area of Criterion VII, "Control of Purchased Material, Equipment and Services," the expert panel also agreed to change site practices for LSSC receipt inspection to allow the use of "certified inspectors" that are qualified in accordance with GGNS requirements for training and certification in lieu of using GGNS "quality inspectors," that are certified in accordance with the requirements of ANSI N45.2.6, "Qualification of Inspection, Examination, and Testing Personnel for Nuclear Power Plants." In the area of Criterion XVI, "Corrective Action," the expert panel clarified that generic implications of both safety related and non-safety related corrective actions will be considered in low safety-significant component applications. In the area of Criterion XVIII, "Audits," the panel conceptually discussed how the LSSC assessments would be conducted to evaluate whether any cumulative safety-impact existed.

During the conduct of the expert panel, the NRC staff questioned how the LSSC controls would be reflected in the QA manual, the GGNS personnel indicated that their plan was to designate the LSSC criteria in the position statement section of the Operational Quality Assurance Manual (OQAM). However, GGNS indicated that the position statement of the OQAM would not be submitted to the NRC staff.

The staff observed that satisfactory interaction was apparent between the expert panel members during discussion on different issues and believed that it was due in part to the organizational diversity and expert panel member knowledge and experience.

Attachments:

1. November 17, 1995 Meeting Attendee List
2. GGNS Quality Assurance Criteria and View Graphs
3. October 11, 1995 EPRI Paper

cc: See next page

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Attachment 1

GGNS EXPERT PANEL MEETING LIST OF ATTENDEES
November 17, 1995

ENTERGY:	ORGANIZATION:
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J. Booth	OPS
C. Brooks	NS & RA
S. Davis	Materials
L. Daughtery	NS & RA
R. Dubey	NPE
R. Ingram	GG-P & SE
G. Lantz	NPE-Elec. & I&C
R. Logan	CDE
M. Meisner	NSRA
G. Smith	NPE
K. Smith	PP&S
B. Warren	NPE

OTHER UTILITIES AND INDUSTRY GROUPS:

M. Burnett	Houston Lighting and Power (STP)
S. Rosen	Houston Lighting and Power (STP)
C. Rogers	Arizona Public Service Co. (PVNGS)
S. Floyd	NEI
A. Neymer	NEI

US NRC OBSERVERS:

W. Ang	Region IV
S. Black	NRR
R. Gramm	NRR
J. Lynch	NRR Consultant
J. Petrosino	NRR

NOTE: Identification on Attendee List does not indicate full time attendance.

Quality Assurance Criteria

Procurement of Low Safety Significance Components

Introduction

Implementation of graded QA at Grand Gulf will be accomplished in a phased manner. It is expected that various aspects of the program will change as experience is gained with graded QA and as graded QA concepts are applied to new areas of site operation.

In its initial stages, the Grand Gulf implementation of graded QA focuses on a graded procurement process. To implement graded procurement two major objectives must be met:

- Development and application of technical criteria to identify those systems and components that are important to safety, and
- Development of quality assurance criteria to be applied to components that are determined to not be important to safety (i.e., LSSCs - low safety significance components).

The first objective was completed through expert panel revision to and concurrence with the EPRI report [later].

The second objective is addressed by this position paper.

Objective of Graded Procurement

The purpose of graded procurement is to restore flexibility in the allocation of resources by eliminating the "quality assurance premium" associated with purchasing LSSCs. In other words, the cost of components purchased "Q" is often several times the cost of an identical component without the "Q" pedigree. Since the cost differential for "Q" components is largely due to the application of a vendor's Appendix B program, the basic tenet for graded procurement of LSSCs is the elimination of the requirement for a vendor to have an Appendix B program.

Quality Assurance Criteria for LSSCs - Overview

The elimination of Appendix B vendor requirements for LSSCs is the only substantive reduction in quality assurance controls for LSSCs. Since the LSSC is not important to safety, its procurement pedigree may be downgraded in compliance with Appendix B's directive to apply quality assurance consistent with an SSC's safety importance. With one exception, all other Appendix B criteria will remain unchanged or increase, as discussed below.

It should also be noted that Appendix B "pedigree" for LSSCs will often be replaced by other quality standards as a natural result of the engineering design process. Although not necessary, specifying that components be purchased to standards such as B31.1 or UL certified, confers added confidence in manufacturing/materials processes for LSSCs.

Application of Appendix B Criteria to LSSCs for Graded Procurement

Few changes in Appendix B applications are necessary to implement a graded procurement program:

- Criterion IV (Procurement Document Control) and Criterion VII (Control of Purchased Material, Equipment and Services) will result in reduced levels of quality assurance oversight (although, not a reduction in commitment as defined by 10CFR50.54) for LSSCs compared to SSCs important to safety,

- Criterion XV (Nonconforming Materials, Parts or Components), Criterion XVI (Corrective Action) and Criterion XVIII (Audits) will result in additional quality assurance oversight for LSSCs compared to SSCs important to safety, and

- The remainder of the Appendix B criteria will continue to be applied in the same fashion as for SSCs important to safety[1].

The application of each Appendix B criterion in the Grand Gulf quality assurance program is discussed below for LSSCs.

<u>Criterion I - Organization</u>

No change.

<u>Criterion II - Quality Assurance Program</u>

No change.

This criterion requires grading.

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[1] As Grand Gulf applies graded QA to processes other than procurement, it is expected that additional quality assurance criteria for LSSCs will be developed. For instance, Criterion VI (Document Control) may be addressed to allow variation in the procedure change process depending upon whether a component is important to safety or an LSSC. These changes, however, are not being pursued as part of the graded procurement effort.

Criterion III - Design Control

No change.

Upon request, the design organization will specify the functional attributes necessary to satisfy the safety classification, regulatory requirements, commitments and economic performance characteristics for any SSC. Such specifications are part of the standard PERR (Procurement Engineering Request/Response) process, which will require no change for graded procurement.

From a Design Control viewpoint, it should be noted that the only effect of graded procurement will be elimination of the need to specify purchase from a vendor with an Appendix B program. All design requirements and commitments (e.g., EQ, seismic, ASME classes, 10CFR21, etc.) remain unaffected by graded QA and must be complied with.

Criterion IV - Procurement Document Control

LSSCs will be designated in appropriate databases as not important to safety. This designation will be understood to allow the purchase of the LSSC from a vendor without an Appendix B program. Such designation only refers to quality assurance procurement controls - it has no effect on other requirements/commitments that apply to the LSSC and their resulting specification by the design authority.

Criterion V - Instructions, Procedures and Drawings

No change.

Criterion VI - Document Control

No change.

Criterion VII - Control of Purchased Material, Equipment and Services

Appropriate procedures will be changed to allow the use of "certified inspectors" rather than "quality inspectors" for the receipt inspection of LSSCs that are safety-related. For this purpose, "certified inspectors" are individuals capable and qualified (via training, qual cards, etc.) to perform the receipt inspection rather than "quality inspectors" certified to ANSI 45.2.6.

The implementation of other portions of Criterion VII is unchanged.

Criterion VIII - Identification and Control of Materials, Parts and Components

No change.

For components that are identical except for pedigree, creation of a new stock code is automatic, and such components are physically segregated.

Criterion IX - Control of Special Processes

No change.

Criterion X - Inspection

No change.

Criterion XI - Test Control

No change.

Criterion XII - Control of Measuring and Test Equipment

No change.

Criterion XIII - Handling, Storage and Shipping

No change.

Criterion XIV - Inspection, Test and Operating Status

No change.

Criterion XV - Nonconforming Materials, Parts or Components

and

Criterion XVI - Corrective Action

Quality assurance controls will be increased.

For some time after implementation of graded procurement, Grand Gulf will have identical components in both important to safety and LSSC applications. If failures of LSSCs occur, the quality assurance program must be able to identify when failure modes may be significant for identical (including pedigree) components in applications important to safety. In other words, if the failure mode could be generic to such components, the corrective action program must ensure that necessary corrective action is applied to the important to safety components.

Appropriate deficiency procedures and forms will be changed to include a question to determine if the component failure mode could be generic and, if so, to apply corrective action to identical components serving important to safety functions. In support of enhancements to Criterion XVIII below, the same procedures will also be changed to include a means to identify when deficiencies occurred on LSSCs.

Criterion XVII - Quality Assurance Records

No change.

Criterion XVIII - Audits

Quality assurance controls will be increased.

The failure of an LSSC, by definition, should have no perceptible adverse impact on safety. However, since graded procurement will result in numerous components being purchased from vendors who do not have an Appendix B program, some additional care should be taken in ensuring that the cumulative safety impact due to graded procurement is minimal. As a prudent measure, Grand Gulf intends to conduct a periodic assessment of LSSC failures to determine if the cumulative effect of such measures results in a perceptible decrease in safety. Should such a situation be discovered, it would constitute a significant condition adverse to quality to be resolved appropriately in accordance with Criterion XVI.

The Quality Programs organization will conduct an assessment in conjunction with appropriate technical personnel every two years to determine if a cumulative safety impact results from not requiring a vendor Appendix B program when purchasing LSSCs. Assessments may be discontinued when it is apparent that no cumulative safety impact results from graded procurement.

To facilitate document retrievability for the assessment, appropriate deficiency procedures and forms will be changed to include a means of identifying which deficiencies are associated with LSSC failures.

QA Criteria
Low Safety Significant Components

- Bottom-up approach
  - Individual failures of low safety significant components (LSSCs) should, by definition, have no adverse effect on a function important to safety
  - Quality assurance controls that minimize such component failure rates should, therefore, add little value to safety
  - While the above is true for the vast majority of LSSCs, there are some valid (but narrow) concerns which should be addressed:
    - Mis-classified LSSCs (i.e., should really be safety significant)
    - LSSCs identical to safety significant components
    - Cumulative effect of LSSC failure

QA Criteria
Mis-Classified LSSCs

- Mistakes in classification will be rare due to overly conservative system and component classification criteria
- Changes in function (either through physical modification or procedure change) which cause the component to be safety significant will require a feedback loop into the Q-list
- In the unlikely case of a failure of a mis-classified component, the corrective action program must ensure the mis-classification is rectified

QA Criteria
Identical LSSCs/Safety Significant Components

- Concern is similar to common-cause or common-mode failure
- Will the corrective action for the LSSC failure be recognized as applicable to identical safety significant components?
- Corrective action program must ensure that generic applicability is considered
- Much of the concern is limited to initial period following graded QA implementation - as low safety significant components are replaced, their pedigree will no longer be identical to that of safety significant components

QA Criteria
Cumulative Effect of LSSC Failure

- While the cumulative safety effect of LSSC failure should be negligible if properly classified, it is prudent to confirm

- The quality assurance program should provide for periodic confirmation that reduced quality assurance for LSSCs has not resulted in an adverse effect on safety

Graded Procurement QA Criteria Changes for LSSCs

- Reduced Scope QA
  - Elimination of vendor QA program requirements
  - Receipt inspector certification (via training, qual cards, etc.) rather than certification to ANSI 45.2.6
- Enhanced scope QA
  - Enhanced controls to ensure generic implications of LSSC failures are applied to identical SSCs
  - Periodic assessment of cumulative effect of increased LSSC failures and implementation of corrective action commensurate with safety importance of the cumulative effect

Graded Procurement QA Criteria Changes for NS-Rs

- Apply changes in a forward looking manner

- As components come up for replacement (and warehouse stock is depleted) NS-R components classified as safety significant will be procured in compliance with Appendix B

10CFR21 and Graded QA

Purpose of 10CFR21:

Identify and disseminate information about basic component defects

Defect:

A departure from the technical requirements included in a procurement document that could create a substantial safety hazard

Relationship to Graded QA:

Assuming correct component categorization, deviations from procurement technical requirements for low safety significant components cannot create a substantial safety hazard

Application of 10CFR21

For identical components in safety significant vs. low safety significant applications:

- The number of critical characteristics may vary (more critical characteristics for safety significant application)
- The level of control exerted over a single characteristic will vary (more stringent controls for safety significant application)

Criteria for Determining the Safety Significance of Plant Structures, Systems, and Components for the Grand Gulf Graded QA Program

October 11, 1995

Prepared by:

Science Applications International Corporation

for

Electric Power Research Institute

ENTERGY Operations Inc. – Grand Gulf Nuclear Station

Prepared by: 10/11/95 Jim Howarth 10/20/95
Entergy Date SAIC Date

10/17/95
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Operations Member Date

10/17/95
PRA Group Member Date

Charles C. Abbott 10/12/95
Quality Programs Member Date

Robert 10/11/95
Central Design Engineering Member Date

Approved by R. K. 10/17/95
Graded QA Expert Panel Chairman Date

Table of Contents

1. Introduction

In 1993, the Electric Power Research Institute (EPRI) began a project to apply PSA to several programmatic areas of nuclear plant operation, maintenance, and regulation. This EPRI project began by identifying the major nuclear power production costs that are amenable to reduction through PSA applications.

One of the cost drivers is quality assurance (QA). QA requirements affect several areas of plant operation, one of which is procurement of replacement parts. A typical nuclear plant may spend $2-5M per year on replacement parts, and up to 70% of that cost may be attributed to the QA "pedigree" of those parts. Many of these parts have a negligible role in preventing core damage and large radionuclide releases, the principal safety functions of the plant.[1] If the plant reduces or eliminates the QA requirements for the non-safety significant parts, it might achieve immediate, substantial savings in procurement costs.

The concept of applying QA criteria commensurate with an item's safety significance is known as *graded QA*. In 1993, the US Nuclear Regulatory Commission (NRC) identified graded QA as an important subject in a workshop on the elimination of requirements marginal to safety [Reference 1]. Later, the NRC described the potential for using PSA technology as a basis for a "graded QA" program [Reference 2]. Various authors have also suggested ways to apply PSA results to graded QA applications [Reference 3].

This report describes a practical application of PSA technology to a nuclear plant's graded QA program. The Electric Power Research Institute (EPRI) and ENTERGY Operations, Inc., jointly sponsored this project. The Grand Gulf Nuclear Station is the pilot plant.

This report is a product of Grand Gulf's expert panel on Graded QA. In a series of meetings held during July 1995, this panel reviewed and revised criteria for assigning QASS Classifications. The expert panel's discussions focused on two inputs from the EPRI project:

- a draft set of QASS Classification criteria
- a set of preliminary QASS Classifications for Grand Gulf systems and components, based on the draft criteria.

This report describes the final criteria produced by consensus of the expert panel members.

Grand Gulf's objective is to have a set of criteria that is:

- easy to understand
- technically defensible

[1] Throughout this report, plant safety is associated with the concept of *dominant public risk contributors*. In the Grand Gulf PSA, the plant's dominant public risks come from potential accidents that exceed the plant's design basis. Two important characteristics of these accidents are (i) the chance of core damage, and (ii) the chance of a large radionuclide release following soon after core damage. If plant equipment significantly affects either of these characteristics, it is considered "safety significant."

- sufficient to identify candidates for reduced QA requirements with a minimum of effort
- detailed enough to allow an independent reviewer to reproduce the major results.[1]

The final set of criteria from the expert panel include both probabilistic and deterministic criteria. The probabilistic criteria use risk importance calculations from Grand Gulf's probabilistic safety assessment (PSA) and related studies. The deterministic criteria arise from insights about the limitations of PSA models and risk importance calculations. These deterministic criteria are intended to have a conservative bias, and substitute for more elaborate probabilistic modeling.

The term *QA safety significant* (QASS) appears throughout this report. It refers to a combination of deterministic and probabilistic criteria. This term has important differences with other related terms, including:

- *Safety Related*. The term "safety related" describes systems, structures, and components bound by 10 CFR 50, 10 CFR 100, and other legal requirements. A fundamental groundrule for Grand Gulf's graded QA program is that an item's safety significance is a property independent of its legal status as a safety or non-safety related. Some safety related items may be non-QASS. Likewise, some non-safety related items may be QASS.
- *Maintenance Rule Risk Significant*. Previously, Grand Gulf evaluated plant systems under the scope of the Maintenance Rule (10 CFR 50.65). This evaluation followed the guidelines given in NUMARC 93-01 [Reference 4]. These NUMARC guidelines limit the scope of plant risks under consideration. They focus the evaluation team on preventive maintenance activities that affect power operation. Because the Maintenance Rule has a limited scope, it allows the evaluation team to exclude "external" sources of risk such as fires, floods, and seismic events. In contrast, the scope of Grand Gulf's graded QA program is broad enough to include other modes of operation, and other sources of plant risk. Thus, it might decide to classify some equipment that is not risk significant under the Maintenance Rule's scope as QASS.
- *Risk Importance*. A PSA model can calculate risk importance measures for systems and components. Several risk importance measures exist [Reference 5]. The two most common are *Risk Achievement Worth (RAW)* and *Risk Reduction Worth (RRW)*. Grand Gulf uses risk importance data as only one of several inputs into its system level evaluations for both the Maintenance Rule and its graded QA program.

[1] The goal for reproducability is to enable an independent reviewer to arrive at the same classifications for safety significance. As long as the final classifications are the same, small differences in the rationale used to assign a QA classifications are considered acceptable.

Grand Gulf will use the criteria described in this report to assign equipment to one of the following two QASS Classifications.

- QA Safety significant systems, structures, and components (SSCs).
- Non-QA safety significant SSCs.

These two classifications are henceforth referred to as "QA Safety Significant" (QASS) or "Non-QA Safety Significant" (Non-QASS).

QASS Classifications are only the first step toward changing GGNS processes. The "non-QASS" label does not remove any other design requirements.

The following two sections describe the criteria for QASS items. Section 2 describes criteria for classifying plant systems. If a system is non-QASS, then all the components in the system are classified as non-QASS unless determined otherwise by the expert panel. If the system is QASS, then some components within the system may still be non-QASS Section 3 describes criteria classifying components within QASS systems.

Appendix A provides details about the supporting analyses that provide the basis for QASS Classifications.

Appendix B describes a second set of criteria for systems and components. These are criteria that can be used to confirm an item's status as non-QASS.

Reference 1 Workshop on Program for Elimination of Requirements Marginal to Safety, U. S. Nuclear Regulatory Commission, NUREG/CP-0129, April 1993.

Reference 2 F. Gillespi, et. al., Draft Report, Regulatory Review Group, Risk Technology Applications, U. S. Nuclear Regulatory Commission, May 1993.

Reference 3 H. Specter, et. al., Technical Session (seven pages), The Future:Risk-Based Methods/Regulation, Transactions of the American Nuclear Society, 68 pages 309-314, 1993.

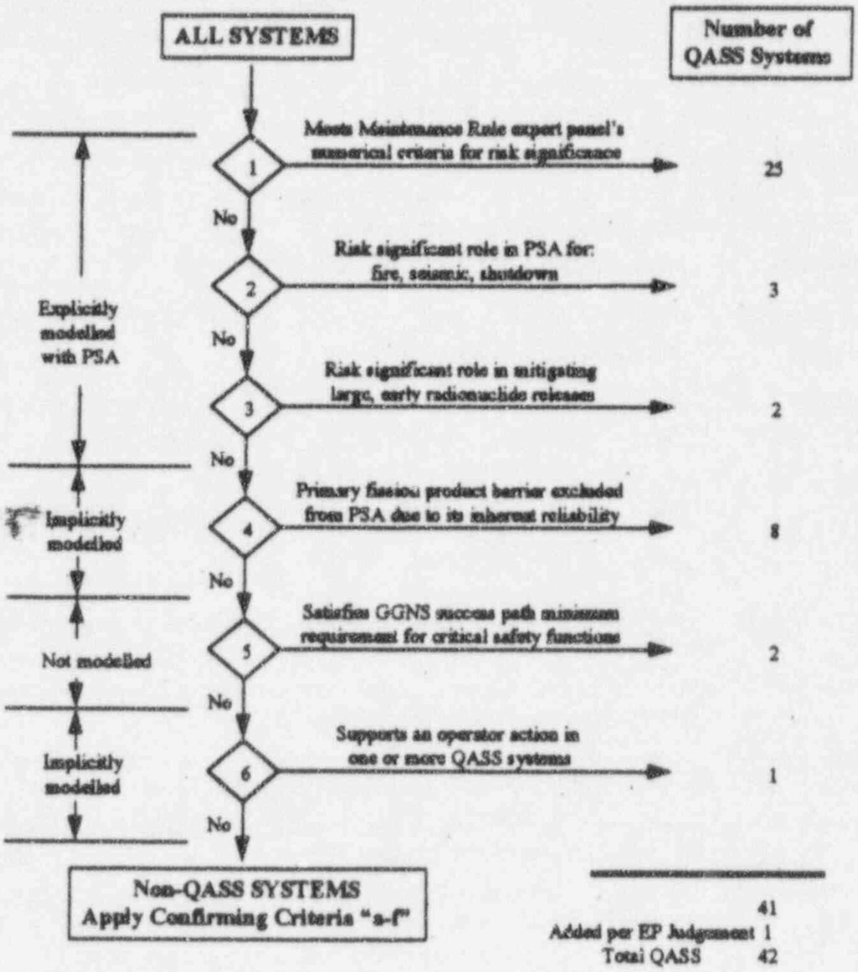
Reference 4 Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants, Nuclear Energy Institute, NUMARC-93-01, May 1993.

Reference 5 W. Vesely, et. al., Measures of Risk Importance and their Applications, U. S. Nuclear Regulatory Commission, NUREG-3385, July 1983.

2. Criteria for Assigning QASS Classifications to Plant Systems

Grand Gulf's expert panel defined six criteria as a basis for classifying systems as QASS. Figure 2-1 shows the sequence in which GGNS applied these criteria, and the number of systems originally identified by each criterion.[1] The numbers are taken from the expert panel meetings held July 11-14 1995.

Figure 2-1 Summary of System-level QASS Criteria[2]



Criteria 1-3 use risk importance measures to classify systems as QASS. Criteria 4-6 identify various ways that systems might be risk significant, but because they are not explicitly modeled in the GGNS PSA, their risk importance measures are unquantified.

Figure 2-1 refers to a step for documenting a decision to classify a system as non-QASS. This step involves a second set of criteria called "confirming criteria." Figure 2-1 refers to these as criteria "a-f," and Appendix B-1 describes these criteria in detail.

[1] Note that this chart does not address which systems satisfy <u>multiple</u> criteria.

[2] The numbers for the criteria are consistent with: (a) Grand Gulf's system summary table, published separately, and (b) the subsections to follow.

Although Figure 2-1 suggests a sequential process, it is important to apply *each* criterion to *each* plant system. Some systems satisfy multiple QASS criteria. Other systems help satisfy multiple plant safety functions. As a result, there may be several reasons for classifying a given system as QASS. A thorough understanding of all the reasons a system is QASS is an important prerequisite to assigning component-level QASS Classifications. Therefore, to make component-level classification possible, it is necessary to evaluate all the reasons why a system is QASS.

The rest of this section describes the QASS criteria shown in Figure 2-1.

2.1 Criterion 1:

This criterion identifies systems that have high numerical risk importance values. In this case, risk importance is measured relative to the chance of core damage, using the plant's Level 1 PSA model.

Appendix A.1 describes the process Grand Gulf used to determine risk significance for the Maintenance Rule. The initial step in that process was to measure the importance of systems:

- Modeled in the PRA, and
- Within the Scope of the Maintenance Rule

Grand Gulf quantified risk importance measures (known as Methods 1, 1A, 2, and 3) for basic events in the PSA model. Then, for each importance measure, Grand Gulf chose a numerical threshold as the boundary between basic events classified as "risk significant" and "not risk significant." Finally, Grand Gulf determined which systems were represented by basic events in the "risk significant" category. If any basic event in the risk significant group could be linkod with a system, then Grand Gulf identified the system as risk significant.

Criterion: **If the system satisfies any numerical screening criteria from Grand Gulf's Maintenance Rule Program Position Statement, the system is classified as QASS.**

Table 2-1 lists additional GGNS systems that satisfy this criterion.

Table 2-1 QASS Systems by Criterion 1

System	System Description
B21	Nuclear Boiler
C11	CRD System
E12	RHR
E21	LPCS
E22	HPCS
E30	Suppression Pool Make-up
E51	RCIC
L11	125v Batteries
L21	125v Swgr & Dist
P11	Condensate & Refueling Water Storage & Transfer
P41	Standby Service Water
P42	CCW
P43	Turb Bldg Cooling Water
P44	Plant Service Water

Table 2-1 QASS Systems by Criterion 1

System	System Description
P53	Instrument Air
P64	Fire Protection
P75	Standby Diesel
P81	HPCS Diesel
R12	6.9 kv xformers
R20	480v Load & MCCs
R21	4.16 kv Swgr & Load Shedding & Sequence Panels
R27	500 kv Ckt Bkrs
T51	Emerg Pump Room Ventilation
X77	Diesel Gen Bldg Ventilation
Y47	Standby Service Water Pumphouse Ventilation

2.2 Criterion 2:

This criterion identifies systems that have high numerical risk importance values in risk studies beyond the scope of the Level 1 PSA. As in Criterion 1, risk importance is measured relative to the chance of core damage.

Grand Gulf has two types of core damage risk studies that produce quantitative measures of risk importance: (i) fire risk analysis and (ii) shutdown risk analyses.

Appendix sections A.2 and A.3 summarize the methods and findings of the fire risk and shutdown risk analyses, respectively.

Grand Gulf is a "reduced-scope" plant for seismic IPEEE. As a result, the IPEEE seismic study is not a quantitative model, and does not include quantitative measures of risk importance. Instead, it uses a deterministic "success path" method that evaluates plant safety functions that prevent core damage. As a result, the seismic evaluation found no new risk significant systems.

Criterion: **If a system is found to be risk significant in any of Grand Gulf's risk studies (i.e., IPEEE and shutdown), the system is classified as QASS.**

Table 2-2 lists additional GGNS systems that satisfy this criterion.

Table 2-2 Additional QASS Systems by Criterion 2

System	System Description
C61	Remote Shutdown
P47	Service Water Radial Well
P65	Fire Detection

2.3 Criterion 3:

This criterion identifies systems that have high numerical risk importance with respect to large early radionuclide releases. In this case, risk importance is measured using the Grand Gulf's Level 2 PSA model.

The chance of radionuclide release is another end state that correlates with public risk. Some systems help prevent radionuclide releases, but have no role in preventing core damage.

These systems help maintain containment integrity, and force radionuclide releases through the suppression pool and other attenuating pathways. These systems may be safety significant, but would not appear as such when focusing strictly on the core damage end state.

Appendix A.4 describes a set of sensitivity calculations with Grand Gulf's Level 2 PSA (a Containment Event Tree). These calculations identify the following functions as risk significant.

- ECCS injection
- Suppression pool cooling
- Drywell structure
- Drywell vacuum breakers

Criterion: **If the system is risk significant for preventing large early radionuclide releases from core damage accidents, it is classified as QASS.**

Table 2-3 lists additional GGNS systems that satisfy this criterion.

Table 2-3 Additional QASS Systems by Criterion 3

System	System Description
E61	Combustible Gas Control
M24	DW/SP/Upper Cont Pool

2.4 Criterion 4:

This criterion identifies systems that:

- Act as primary barriers to fission product release (fuel rods, reactor, and containment), and
- Are not modeled in the PSA.

These systems are not modeled in the PSA because the components are highly reliable in their role as a fission product boundary. This is a standard PSA modeling practice. Failure probabilities for pipe or valve ruptures are so low that the component failures do not appear in any of the PSA's dominant cutsets. As a result, these components have no numerical measures of risk importance.

By inspection, one can ascertain that components in fission product barriers would (if the PSA included very low probability cutsets) score high on one risk importance scale: risk achievement worth (RAW).

Similar to the Mxx (containment integrity) system, Grand Gulf defined a system Bxx (Reactor Coolant Pressure Boundary and Containment Penetration Piping). This system includes components that form part of the reactor coolant pressure boundary, other than those already included in systems B13 and B33. This will include:

- All ASME Class I piping out to the outboard reactor coolant pressure boundary isolation valve in any system connected to the RCS.
- Piping in containment penetrations.

Criterion: **If a fission product barrier is not explicitly modeled in the IPE because of its inherent high reliability, it is classified as QASS.**

Table 2-4 lists additional GGNS systems that satisfy this criterion.

Table 2-4 Additional QASS Systems by Criterion 4

System	System Description
B13	Reactor System
B33	Reactor Recirculation
Bxx	Reactor Coolant Pressure Boundary and Containment Penetration Piping
J11	Fuel
M10	Containment
M23	Hatches & Locks
Mxx	Containment Isolation
R60	Penetrations

2.5 Criterion 5:

This criterion identifies a minimum acceptable complement of systems needed to perform the safety functions that prevent core damage. Appendix A.5 describes these safety functions and the systems capable of providing those functions.

Some systems may have low (or unmeasured) numerical risk importance values because several diverse, reliable systems perform the same safety function. For Grand Gulf, this specifically applies to systems that provide the reactivity control function. There are several diverse ways of achieving reactor shutdown, and therefore all the systems that perform this function have low numerical risk significance.[1] This forms a baseline assumption in the PSA, and as a result, the model does not represent these systems in detail. Consequently, the PSA does not produce risk importance measures for reactivity control system components.

Changing the QA requirements for one system can influence the risk importance of other systems performing the same function. If you reduce QA requirements for a system, then theoretically, you can expect some loss of system reliability (however small). In addition, this change elevates the risk importance of components in other systems performing the same function. Ideally, a PSA model would be able to ensure (by measuring numerical changes in risk importance) that QA changes do not reduce safety function reliability to an unacceptable level.

Unfortunately, given the PSA limitations discussed above, the model is unable to measure the impact of QA changes on the reliability of all safety functions (e.g., reactivity control). It is impractical for Grand Gulf to modify its PSA to make more detailed importance calculations. Instead, this deterministic criterion serves as an alternative way of ensuring that QA changes do not reduce safety function reliability to an unacceptable level.

[1] The CRD system is a partial exception. Its risk importance value is high because of its ability to provide an alternate means of reactor inventory control—not because of its reactivity control function.

This criterion establishes a minimum QA requirement for each safety function. Using this approach, Grand Gulf's Graded QA expert panel established a requirement that at least one means of providing every safety function remain QASS.[1]

Grand Gulf's safety functions include:

- Reactivity control
- Reactor pressure control
- Reactor inventory control
- Decay heat removal

The expert panel has the prerogative of choosing which among several alternatives to maintain as QASS, after considering the following factors:

- Relative effectiveness of QA for each alternative (QA may be more effective for a standby systems, since active systems tend to be "self-correcting")
- The impact of plant configuration changes during normal operations
- Relative system reliability
- The system's ease of use (or, the potential for errors of commission)
- The electrical and mechanical diversity the system adds to the collection of other QASS systems.

Criterion: **Ensure that at least one system or set of systems necessary to complete each critical safety function is classified as QASS.**

Table 2-5 lists additionalf GGNS systems that satisfy this criterion.

Table 2-5 Additional QASS Systems by Criterion 5

Function	System	System Description
Reactivity Control	C51	Neutron Monitoring
	C71	Reactor Protection Sys.

2.6 Criterion 6:

The PSA model includes several operator actions associated with plant systems. Of these, the actions associated with QASS systems (identified by preceding criteria) are potentially risk significant. This criterion identifies any other systems that have equipment needed to perform those potentially risk significant actions.

The following table lists some of the operator actions modeled in the PSA that are among QASS systems.

Basic Event Name	Description
B21-FO-HEADS-I	Operator fails to manually initiate ADS at subchannel Level
B21-FO-HEDEP-I	Human error fail to depressurize with ADS valves
E12-FO-HEECCS-G	Operator fails to initiate ECCS
HVC-FO-HEMOD-U	Operator fails to open dampers

[1] Expert panel meeting minutes, July 12, 1995.

Basic Event Name	Description
P75-FO-HE-DG11-I	Operator fails to manually initiate DG11
E51-FO-HEISOL8-G	Operator fails to manually isolate RCIC system
E51-FO-HESTNIS-G	Operator fails to bypass high steam tunnel temp isolation
E51-FO-HESYACT-G	Operator fails to manually initiate RCIC
E22-FO-HEF015-I	Operator fails to open SP suction valve
E22-FO-HEHPCS-I	Operator fails to manually actuate HPCS
P81-FO-HE-DG13-I	Operator fails to manually initiate DG13
E12-FO-HESPC-M	Operator fails to manually align for suppression pool cooling
B21-FO-HEDEP2-I	Operator fails to manually depressurize vessel with Non-ADS valves
P75-FO-HE-DG12-I	Operator fails to manually initiate DG12
E12-FO-HESDC-O	Operator fails to properly align for shutdown cooling

Criterion: **If instrumentation or actuation equipment in remaining[1] non-QASS systems is necessary for the operator to perform an operator action modeled in a QASS system, then ensure that at least one system or set of systems (sufficient to support the actions) is classified as QASS**.

Table 2-6 lists additional GGNS systems that satisfy this criterion.

Table 2-6 Additional QASS Systems by Criterion 6

System	System Description
L62	Inverters

2.7 Collective Expert Judgement of the Panel

After applying the preceding criteria, Grand Gulf's expert panel deliberated on the remaining plant systems to see if any others should be classified QASS. By majority vote, the panel classified the systems listed in Table 2-7 as QASS.

Table 2-7 Additional QASS Systems by Collective Judgement of the Panel

System	System Description
E61	Combustible Gas Control
Z51	Control Room HVAC

[1] This criterion should be applied after Criteria 1-5. It applies only to the non-QASS systems remaining.

3. Criteria for Assigning QASS Classifications to Components in QASS Systems

All components in non-QASS systems are classified as non-QASS.

This section describes five criteria for classifying components as QASS, given that they are members of a QASS system. Appendix B.2 describes six criteria that can be used to confirm a component's status as non-QASS.

To support the overall goal of reproducibility, many of these criteria have been translated into computerized rules that can be used with a database management program.

3.1 Criterion H1:

An engineer familiar with the PSA can determine which components are modeled in the PSA. In the system evaluations published to date, this determination is recorded in the "S" column of the Graded QA Analysis Table. This table contains a row for every component listed in the Grand Gulf SIMS database. A value of "D" in the "S" column indicates the SIMS component is directly modeled in the PSA.

Criterion: **If an engineer familiar with the PSA determines that a component is explicitly modeled as either a component or supercomponent, the component is classified as QASS.**

3.2 Criterion H2:

As described in Section 3.1 an engineer familiar with the PSA can determine which components are modeled in the PSA. In the system evaluations published to date, this determination is recorded in the "S" column of the Graded QA Analysis Table. This table contains a row for every component listed in the Grand Gulf SIMS database. A value of "S" in the "S" column indicates the SIMS component is required to support an item modeled in the PSA.

Criterion: **If an engineer familiar with the PSA determines that a component is needed to support another component or supercomponent modeled in the PSA, then the component is classified as QASS.**

3.3 Criterion H3:

To remain conservative, Grand Gulf will continue to classify components as QASS until it establishes a specific basis for classifying components as non-QASS.

Criterion: **If a component has not been evaluated for its safety significance against other component-level criteria, then the component is classified as QASS.**

3.4 Criterion H4:

Components that support operator actions have a potentially important role in minimizing risk. The following list shows some of the operator actions that appear in the PSA.

Basic Event Name	Description
B21-FO-HEADS-I	Operator fails to manually initiate ADS at subchannel level
B21-FO-HEDEP-I	Human error fail to depressurize with ADS valves
E12-FO-HEECCS-G	Operator fails to initiate ECCS
HVC-FO-HEMOD-U	Operator fails to open dampers
P75-FO-HE-DG11-I	Operator fails to manually initiate DG11
E51-FO-HEISOL8-G	Operator fails to manually isolate RCIC system
E51-FO-HESTNIS-G	Operator fails to bypass high steam tunnel temp isolation
E51-FO-HESYACT-G	Operator fails to manually initiate RCIC
E22-FO-HEF015-I	Operator fails to open SP suction valve
E22-FO-HEHPCS-I	Operator fails to manually actuate HPCS
P81-FO-HE-DG13-I	Operator fails to manually initiate DG13
E12-FO-HESPC-M	Operator fails to manually align for suppression poool cooling
B21-FO-HEDEP2-I	Operator fails to manually depressurize vessel with ADS Valves
P75-FO-HE-DG12-I	Operator fails to manually initiate DG12
E12-FO-HESDC-O	Operator fails to properly align for shutdown cooling

Where these actions apply to QASS systems, then components that support these actions are considered QASS. Examples of the types of components that would meet this criterion include:

- Frequency meter/transducer (for diesel generators only)
- Control room indicators
- Control room meters
- Transducers (if in the loop for control room indications)

Criterion: **If a component provides instrumentation or an actuation device that operators need to perform a PSA-modeled operator action for a QASS system, then the component is classified as QASS.**

3.5 Criterion H5:

The criteria discussed in Section 3.1 and 3. 2 require an engineer to determine which components are modeled in the PSA (either directly or in a support role). Some components may not be modeled in the PSA, but nevertheless have a role in one of the extensions to the PSA--either the IPEEE analyses or the shutdown risk analyses.

An example of a component that meets this criterion is a seismic snubber.

Criterion: **If a component is not modeled in the PSA, but is nevertheless required to perform a risk significant function in other plant risk studies (IPEEE and shutdown), then the component is classified as QASS.**

4. Summary of Grand Gulf QASS Systems

Table 4-1 shows a summary of all the QASS systems identified by the criteria described in Section 2.

Table 4-1 does not necessarily identify all criteria that each system meets. Prior to deleting a system from Table 4-1, it is necessary to evaluate the system aganist all other criteria.

Table 4-1 Summary of Grand Gulf QASS Systems

System	System Description	System includes Safety Related Components	Components Modeled in the PRA	Number of Safety Related Components	Total Components	QASS Criterion 1	QASS Criterion 2	QASS Criterion 3	QASS Criterion 4	QASS Criterion 5	QASS Criterion 6	Other QASS	Q/ Non-Q
B13	Reactor System	Y	N	413	610				X				Q
B21	Nuclear Boiler	Y	Y	1843	2241	X							Q
B33	Reactor Recirculation	Y	Y	436	1445				X				Q
Bxx	Reactor Coolant Pressure Boundary and Containment Penetration Piping	Y	N						X				Q
C11	CRD System	Y	Y	3629	5318	X - Coolant Injection				X			Q
C51	Neutron Monitoring	Y	N	279	389					X			Q
C61	Remote Shutdown	Y	N	219	219		X - per Fire IPEEE						Q
C71	Reactor Protection Sys.	Y	Y	322	385					X			Q
E12	RHR	Y	Y	1244	1384	X	X - ADHR (shutdown)	X - ECCS Injection					Q
E21	LPCS	Y	Y	182	197	X		X - ECCS Injection					Q
E22	HPCS	Y	Y	402	444	X		X - ECCS Injection					Q
E30	Suppression Pool Make-up	Y	Y	147	163	X							Q
E51	RCIC	Y	Y	464	534	X							Q
E61	Combustible Gas Control	Y	Y	763	865			X - DW vacuum breakers				X - H2 Igniters	Q
J11	Fuel	Y	N	2	2				X				Q

System	System Description	System Includes Safety Related Components	Components Modeled in the PRA	Number of Safety Related Components	Total Components	QASS Criterion 1	QASS Criterion 2	QASS Criterion 3	QASS Criterion 4	QASS Criterion 5	QASS Criterion 6	Other QASS	Q/ Non-Q
L11	125v Batteries	Y	Y	2	10	X							Q
L21	125v Swgr & Dist	Y	Y	84	297	X							Q
L62	Inverters	Y	N	56	126						X		Q
M10	Containment	N	N		11				X				Q
M23	Hatches & Locks	Y	N	37	59				X				Q
M24	DW/SP/Upper Cont Pool	N	N		8			X - Drywell structure					Q
Mxx	Containment Isolation	Y	Y						X				Q
P11	Condensate & Refueling Water Storage & Transfer	Y	Y	99	378	X							Q
P41	Standby Service Water	Y	Y	1178	1311	X							Q
P42	Component Cooling Water	Y	Y	164	492	X							Non-Q
P43	Turb Bldg Cooling Water	Y	Y	1	865	X							NonQ
P44	Plant Service Water	Y	Y	116	778	X	X - ADHR Support (Shutdown)						Non-Q
P47	Service Water Radial Well	N	Y		681		X - ADHR Support (Shutdown)						Non-Q
P53	Instrument Air	Y	Y	67	1983	X	X - ADHR Support (Shutdown)						Non-Q
P64	Fire Protection	Y	Y	40	2201	X	X - per Fire IPEEE						Non-Q
P65	Fire Detection	N	N		800		X - per Fire IPEEE						Non-Q

System	System Description	System Includes Safety Related Components	Components Modeled in the PRA	Number of Safety Related Components	Total Components	QASS Criterion 1	QASS Criterion 2	QASS Criterion 3	QASS Criterion 4	QASS Criterion 5	QASS Criterion 6	Other QASS	Q/ Non-Q
P75	Standby Diesel	Y	Y	856	1307	X							Q
P81	HPCS Diesel	Y	Y	462	570	X							Q
R12	6.9 kv xformers	N	Y		52	X							Non-Q
R20	480v Load & MCCs	Y	Y	529	1406	X							Q
R21	4.16 kv Swgr & Load Shedding & Sequence Panels	Y	Y	267	615	X							Q
R27	500 kv Ckt Bkrs	N	Y		31	X							Non-Q
R60	Penetrations	Y	N	49	151				X				Q
T51	Emerg Pump Room Ventilation	Y	Y	46	60	X							Q
X77	Diesel Gen Bldg Ventilation	Y	Y	37	82	X							Q
Y47	Standby Service Water Pumphouse Ventilation	Y	Y	47	75	X							Q
Z51	Control Room HVAC	Y	N	423	500							X	Q

Appendix A

This appendix provides additional details about other analyses that provide a basis for determining the safety significance of plant structures, systems, and components for the Grand Gulf Graded QA Program. It includes:

- A.1: an excerpt from Grand Gulf's report on implementing the Maintenance Rule. It describes the Maintenance Rule's expert panel's procedure and results.
- A.2: a summary of insights about safety significant systems identified in the GGNS IPEEE fire risk analysis.
- A.3: a summary of findings from an analysis of risk important systems during outages.
- A.4: a summary of findings from an analysis of the risk importance of systems designed to mitigate radionuclide release following core damage accidents.
- A.5: an excerpt from Grand Gulf's Seismic IPEEE report. It describes the plant's critical safety functions and the systems success criteria for each function.

A.1 Maintenance Rule Criteria for Risk Significance

[This subsection is an excerpt from Grand Gulf's documentation of the criteria and methods used to determine the risk significance of systems for the Maintenance Rule Program Position Statement]

9.0 Establishing Risk and Performance Criteria/Goal Setting and Monitoring

9.1 Reference

10 CFR 50.65 (a)(1)

Each holder of an operating license under §§ 50.21 (b) or 50.22 shall monitor the performance or condition of structures, systems, and components against licensee established goals, in a manner sufficient to provide reasonable assurance that such structures, systems, and components, as defined in paragraph (b), are capable of fulfilling their intended functions. Such goals shall be established commensurate with safety and, where practical, take into account industry-wide operating experience. When the performance or condition of a structure, system, or component does not meet established goals, appropriate corrective action shall be taken.

9.2 Guidance

Once the selection of those SSCs determined to be within the scope of the Maintenance Rule (Section 8.0) has been completed, it is then necessary to establish risk significant and performance criteria to initially determine which SSCs must have goals established and monitoring activities performed in accordance with (a)(1). For SSCs that do not meet performance criteria, goals are established commensurate with an SSCs safety significance and performance. Monitoring the performance of the SSCs against established goals is intended to provide reasonable assurance that the SSCs are proceeding to acceptable performance.

All SSCs determined to be within the scope of the Maintenance Rule are subject to an effective PM program as indicated by (a)(2) (see Section 10.0). SSCs that are within the scope of (a)(2) could be included in the formal PM program, be inherently reliable (e.g., visual inspection during walkdowns to meet licensee requirements that already exist), or be allowed to run to failure (provide little or no contribution to system safety function). When SSCs in (a)(2) do not perform acceptably, they are evaluated to determine the need for goal setting and monitoring under the requirements of (a)(1).

9.3 Determining the SSCs Covered by (a)(1)

This section explains how to determine which SSCs under the scope of the Maintenance Rule will have goals and monitoring established in accordance with (a)(1). Establishing both risk significant criteria (Section 9.3.1) and performance criteria (Section 9.3.2) is necessary to provide a standard to measure the performance of SSCs (Section 9.3.3).

9.3.1 Establishing Risk Significant Criteria

In the establishment of risk significant criteria, the PRA/IPE plant evaluation, submitted in December 1992, was used to determine risk significant systems. The following criteria determination methods were evaluated to choose the most appropriate listing to be used as input to an expert panel.

The criteria for determining risk significance was given the PRA coordinator for GGNS to develop the risk significant list for each method. A Senior Engineer and a Maintenance Specialist reviewed the lists developed from the methodology. The more comprehensive and conservative list developed was given as input to the expert panel for consideration of risk significance. The seven members of the expert panel consisted of:

- 6 SRO, shift experienced personnel
    * 1 of 6 SROs also experienced in EOP program
    * 1 of 6 SROs trained in PRA methodology
    * 2 of 6 SROs have engineering backgrounds
- 1 PRA expert

---

7 TOTAL

9.3.1.1 Method #1, Risk Reduction Worth - A

This method summed all the Vesely importance values to get a 100% value. Each Vesely value was divided by the 100 percent value. All cut sets which summed to a total of 99 percent of the previously determined 100 percent are the systems identified by this method to be Risk Significant. Risk Reduction Worth assumes the SSC has perfect reliability for all failure modes.

Method #1A, Risk Reduction Worth - B

This method was derived from Method #1. It was suggested that Method #1 did not focus on the proper criteria for determining Risk Significance. To determine Risk Significant systems with this method, total all the Vesely importance values except for those which are human error related. Again as in #1, divide each cut set by this total. The cut off for this method is 99.9 percent of the Vesely importance total instead of the previous 99 percent. Risk Reduction Worth assumes the SSC has perfect reliability for all failure modes.

9.3.1.2 Method #2, Core Damage Frequency Contribution

This method uses a 90 percent cut off of all of the cut sets that are used to determine overall core damage frequency for GGNS. In this 90 percent, all system(s) that were used were considered Risk Significant.

9.3.1.3 Method #3, Risk Achievement Worth

This method assumed that the system or component would fail. The result would be the effect on the overall CDF. The cut off point for this method is - if the CDF doubled then the SSC would be considered Risk Significant. Risk Achievement Worth is the increase in risk if the SSC is assumed to be failed for all failure modes.

9.3.1.4 PRA Methodology Results

Method 1, 1A, 2 & 3 Results

	#1	#1A	#2	#3
1. B21 - Nuc. Boiler (SRVs)	X	X	X	X
2. C11 - CRD System	X	X	X	
3. C41 - Stby Liquid Control				
4. E12 - RHR System	X	X	X	X
5. E21 - LPCS System			X	
6. E22 - HPCS System	X	X	X	X
7. E30 - S/P Makeup System	X	X		X
8. E51 - RCIC System	X	X	X	
9. L11 - 125V Batteries	X	X	X	X
10. L21 - 125V Bat. Swgr & Panels	X	X	X	X
11. P11 - CST & RW Storage/Xfr			X	
12. P41 - SSW System	X	X	X	X
13. P42 - CCW System			X	

14. P43 - TBCW System	X	X	X	X
15. P44 - PSW System		X	X	X
16. P53 - Instrument Air	X		X	
17. P64 - Fire Protection	X	X	X	X
18. P75 - Stby Diesel/Gen System	X	X	X	X
19. P81 - HPCS Diesel/Gen System	X	X	X	X
20. R12 - 6.9KV Power Xfmrs			X	
21. R20 - 480V LC or MCCs		X	X	X
22. R21 - 4.16KV Swgr & LSS Panels	X	X	X	X
23. R27 - 500KV CB&115KV MODs			X	
24. T51 - Emerg. Pmp Rm Vent	X	X	X	X
25. X77 - D/G Bldg Vent System	X	X	X	X
26. Y47 - SSW Pmp Hse Vent System	X	X	X	X
27. Mxx - Containment Integrity	X	X	X	
TOTALS	19	20	25	17

Criteria method #2 gave the more comprehensive listing of systems that are risk significant. This list was used as input to the Expert Panel to make the final determination on risk significant.

9.3.1.5 Expert Panel Methodology and Results

This group was selected based on their prior training and experience. The input from this panel was help to determine what systems at GGNS (using PRA input) would be classified as Risk Significant relative to the Maintenance Rule.

The Delphi method for eliciting expert opinion was be used. The following functions were taken into consideration when ranking these systems.

A. Accident Mitigation Functions

1. Reactor Level, Inventory Control

2. Reactivity Control

3. Decay Heat Removal

The expert panel ranked the systems from 1 to 10 based on the relative significance to the functions listed above. Also, the panel reviewed the list to determine which systems should be deleted or added. All plant systems were considered for inclusion.

(1 - being not important ... 10 - being very important)

The following are the results of all panel member submittals. (Mxx - Containment Integrity was initially left out of the system list given to the Expert Panel. Later it was decided to add Mxx to the risk significant list due to emphasis placed on this function per GGNS PRA/IPE submittal.

Numerical Ranking of Results

SYSTEM	NUMERICAL RANKING	
1. Stby Diesel Gen System	6.80	
2. 4.16 KV Swgr and LSS	6.66	
3. RHR System	6.52	
4. 125V DC Batteries	6.19	
5. 125V DC Swgr and Panels	6.19	
6. Nuclear Boiler System	6.04	
7. Stby Service Wtr System	6.00	
8. 480V Load Centers or MCCs	5.85	
9. CRD Hydraulic System	5.42	
10. LPCS System	5.09	
11. HPCS System	4.95	
12. RCIC System	4.71	
13. HPCS Diesel Gen System	4.47	
14. D/G Bldg Vent System	3.61	
15. Instrument Air System	3.57	
16. 500KV C-B and 115KV Disc	3.57	
17. Emerg Pmp Rm Vent System	3.52	
18. SSW Pmp Hse Vent System	3.38	
19. Plant Service Water	2.95	
20. Fire Protection System	2.90	
21. Cond & Refuel Wtr Storage	2.61	
22. CCW System	2.47	(Cutoff of 2.00 used)
23. 6.9KV Power Transformers	1.80	
24. TBCW System	1.14	

Additions and/or Deletions:

A. C51 - Neutron Monitoring System	1/7 member(s) said to add
B. C41 - Stby Liquid Control	4/7 member(s) said to add
C. B21 - Vessel Level Instrumentation	1/7 member(s) said to add
D. R27 - 500KV C-B and 115KV disc	1/7 member(s) said to delete
E. P43 - TBCW System	2/7 member(s) said to delete
F. R12 - 6.9KV Power Transformers	1/7 member(s) said to delete
G. G33 - RWCU System	1/7 member(s) said to add
H. E30 - SPMU System	1/7 member(s) said to add
I. N21/N19 - Condensate System(s)	1/7 member(s) said to add
J. C71 - Reactor Protection System	1/7 member(s) said to add

Analysis of Additions and Deletions

Following completion of the expert panel evaluation, it was determined that a cutoff value was needed. The cutoff of 2.0 was used due to R12 system (ranked 1.8) is a system which was not scoped within the Rule for GGNS. This was an obvious cutoff limit. The two systems which fell below 2.0 were R12-6.9 kv Power Transformers and P43 - Turbine Building Cooling Water system. The recommendation from the expert panel to remove the P43 system from the list of risk reinforced the legitimacy of the 2.0 numeric cutoff.

All 25 systems (using Method #2) will be identified as Risk Significant except for the following:

1. R12- 6.9 kv Power Transformers will be deleted from the list due to ranking very low (numerically) with one member recommending deletion. There are 5 categories for systems to be included in the scope of the Maintenance Rule. R12 does not fall under any of these categories. Risk Significant is a sub-list of systems within the scope of the Maintenance Rule and it stands to reason that any Risk Significant system would fall within the initial scoping criteria.

2. C41 - SLC System will be added to the list due to the majority of panel members recommending its addition to Risk Significance. This system does not show up in the list of systems in a 90% cut of all of the cut sets scoped in the PRA. This is due to no credit taken for the manual initiation for this system and power requirements which cannot be assumed to be available for system function.

3. P43 will be deleted from the list due to 2 out of 7 panel members recommending deletion and a low numerical ranking. Also, the loads from this system will be picked up by the SSW - Standby Service Water system in an accident situation. This system is considered lost and not recoverable for the duration of the accident.

From the previous methodology, 24 systems were identified as Risk Significant.

Containment Integrity (Mxx) is a system that was created to combine the function of containment isolation from all systems which would simplify system evaluations. In review of the systems identified by the previous methodology, along with Expert Panel results, it was decided to include Containment Integrity into the risk significant list due to this function mentioned in cut sets of risk significant systems.

The following is the final listing for the 24 risk significant systems for GGNS:

<u>Method 1, 1A, 2, 3 and Expert Panel Results</u>

	#1	#1A	#2	#3	Expert Panel	Risk/Sig
1. B21 - Nuc. Boiler (SRVs)	X	X	X	X	X	X
2. C11 - CRD System	X	X	X		X	X
3. C41 - Stby Liquid Control					X	X
4. E12 - RHR System	X	X	X	X	X	X
5. E21 - LPCS System			X		X	X
6. E22 - HPCS System	X	X	X	X	X	X
7. E30 - S/P Makeup System	X	X		X		
8. E51 - RCIC System	X	X	X		X	X
9. L11 - 125V Batteries	X	X	X	X	X	X
10. L21 - 125V Bat. Swgr	X	X	X	X	X	X
11. P11 - CST & RW Storage/Xfr			X		X	X
12. P41 - SSW System	X	X	X	X	X	X
13. P42 - CCW System			X		X	X
14. P43 - TBCW System	X	X	X	X		
15. P44 - PSW System		X	X	X	X	X
16. P53 - Instrument Air	X		X		X	X
17. P64 - Fire Protection	X	X	X	X	X	X
18. P75 - Stby D/G System	X	X	X	X	X	X
19. P81 - HPCS D/G System	X	X	X	X	X	X
20. R12 - 6.9KV Power Xfmrs			X			
21. R20 - 480V LC or MCCs		X	X	X	X	X
22. R21 - 4.16KV Swgr & LSS	X	X	X	X	X	X
23. R27 - 500KV CB&115KV MODs			X		X	X
24. T51 - Emerg. Pmp Rm Vt	X	X	X	X	X	X
25. X77 - D/G Bldg Vent Sys.	X	X	X	X	X	X
26. Y47 - SSW P/H Vent Sys.	X	X	X	X	X	X
27. Mxx - Cont. Integrity	X	X	X			X
TOTALS	19	20	25	17	23	**24**

A.2 A Summary IPEEE Fire Risk Analysis Insights

Some fire scenarios provide measurable contributions to plant risk. In Grand Gulf's IPEEE fire risk evaluation, the most important fire scenarios occur in plant areas with risk significant equipment. These plant areas have fire detection and suppression systems used to mitigate the consequences of a fire.

According to the IPEEE study, the risk significant fire scenarios are those that involve failures of the detection and suppression systems. In turn, this means that the equipment in the fire detection and suppression systems has a high risk importance. The IPEEE fire risk analyses identified risk significant fire areas in the Auxiliary Building, the Diesel Generator Building, and the Control Building.Grand Gulf's Fire Protection, Fire Detection, and CO2 Storage systems (P64 and P65) have equipment that can mitigate risk significant fires.

Another way to cope with fire risk is to provide redundant system capability. In the case of a Control Room fire, Grand Gulf's redundant system is the Remote Shutdown Panel (C61). This system is also risk significant, because its reliability is a significant factor affecting the risk of Control Room fires.

A.3 Safety Significant Systems During Outages

During shutdown operations, equipment outages tend to be relatively long and occur more often. Shutdown operations involve continuous changes in the plant configuration and equipment availability. These changes produce changes in the level of plant safety, and changes in the ranking of risk important equipment. Depending on outage-specific conditions, one or more low capacity, non-ESF systems may have a high risk importance. Consider two examples drawn from Grand Gulf's shutdown risk models.

- The Alternate Decay Heat Removal (ADHR) system (which is part of system E12) influences safety early in an outage, when one or more RHR trains are out of service.
- As decay heat levels decline, the fire water system (P64) becomes increasingly capable of mitigating loss of RCS inventory events, elevating its risk importance. If enough ECCS are out of service, the fire water system takes on a high risk importance.

The fact that a system's risk importance is outage-specific raises an important question for a graded QA program:

> *Is there a reasonable way to identify safety significant systems during outages in general?*

There is a "conservative" answer to this question. The most conservative approach is to consider all the physically possible outage scenarios, and identify the set of systems that appear risk significant in *any* scenario.

This conservative approach is impractical. The set of "all possible" outage scenarios includes many that fail to pass muster with outage planners.[1] The only way to account for all these "physically possible" outage scenarios is to classify every system as risk significant, at which point the Graded QA concept becomes academic. A more practical approach is discussed in the following sections.

A.3.1 A Practical Approach to Implementing Graded QA

A practical approach to Graded QA begins by limiting the evaluation of shutdown risk importance to a set of "likely" outage schedules. These are outages that conform to Technical Specifications, NUMARC 91-06, and other constraints on outage activities. GGNS's administrative procedures integrate these requirements into a prescription for designing an outage schedule. These requirements standardize the role of systems during outages and provide a credible, practical basis for making judgments about safety significance.

GGNS has outage risk models for its four recent refueling outages: RF04, RF05, RF06, and RF07. This collection of outage risk models can serve as a basis for measuring system risk significance.

The GGNS outage risk models evaluate only two safety functions: decay heat removal and inventory control. Table A-1 lists the systems referenced by the model. The following section describes how the outage risk model can be used to quantify the risk importance of these systems.

Table A-1

Safety Function	System Description	System Code
Decay Heat Removal	- RHR shutdown cooling - ADHR - RWCU cooling through non-regenerative heat exchangers - FPCCU	E12 E12 G33 G41
RCS Inventory Control	- HPCS - LPCS - CRD hydraulic system - Fire water system - Condensate and Refueling water storage and transfer system - LPCI Mode of RHR	E22 E21 C11 P64 P11 E12

A.3.2 Outage Risk Importance Calculations

Currently, GGNS uses EPRI's ORAM software for its outage risk models. The ORAM software does not perform standard risk importance calculations, and the ORAM model does not portray individual components.

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[1] An example might be an outage scenario involving simultaneous work on all three diesel generators, with fuel in the reactor, and a high decay heat level. It is physically possible, but would never occur within the bounds of GGNS' outage planning guidelines.

Hence, it is not possible to use the ORAM model to produce risk importance data that are directly comparable to the risk importance data derived from the GGNS PRA.

Still, it is possible to use the ORAM model to compute an "outage-wide average" risk importance measure. The procedure involves a series of sensitivity analyses with the ORAM model, using the following steps:

- Solve the model once for a "typical" outage, using best estimates for system failure probabilities.
- Record the resulting outage-wide average risk measure, and use it as a "benchmark" for subsequent calculations.
- Change an input value to the ORAM model. Choose a parameter that affects the chance that a system can respond to an accident, such as:
    - the system's conditional failure probability, or
    - the system's scheduled unavailability.
- Solve the model again.
- Record the new outage-wide average risk measure, and its ratio to the benchmark value. This ratio is a measure of a system's outage risk importance.

A.3.3 Risk Importance Calculations Using the GGNS ORAM Model

The sixth GGNS refueling outage (RF06) was selected as a reference outage for determining outage risk importance. The GGNS ORAM model was modified to evaluate the importance of low-flow injection systems.

Based on the RF06 outage schedule, the reference point for GGNS' outage risk importance calculations--the outage-wide average risk of core damage--is lower than plant risk during power operation. On an annual basis, the chance of core damage due to shutdown accidents is about 1E-6 events per calendar year. This is about a factor of 30 below the core damage frequency during power operation.

The ORAM event tree sequence model includes the event OPL, the chance operators fail to inject water into the RCS with alternate injection systems. The plant emergency procedures refer to the following low-flow systems:

- Condensate transfer pumps
- Refueling water storage and transfer pumps
- CRD pumps
- Fire water pumps

The original ORAM model assumed that a common operator error dominates the chance that all these systems fail to inject. Based on that assumption, the original model had no explicit reference to hardware failures within these systems.

The revised ORAM model includes explicit references to hardware failures within these systems. The revision includes:

- Additions to the list of ORAM variables
- Changes to the ORAM event logic formula for the event OPL

A term representing the chance each system fails on demand was added to the list of ORAM variables. Calculations with CAFTA provided the basis for these values used in the ORAM model. The results appear in Table A-2.

Table A-2

System	P(failure)
Condensate Transfer	0.1
Refueling Water Storage and Transfer	0.1
CRD	0.129
Fire Water	0.0426

Next, the ORAM formulae for event OPL were modified to represent the joint failure probability of these systems. These formulae consider system maintenance states, (which may vary during an outage), and support system dependencies.

Table A-3 summarizes the results of the sensitivity runs with the revised ORAM model.

Table A-3

Case	Avg. CDF (per outage-hour)	Outage RAW	System Code
Base Case	1.26E-9	N/A	
P(ADHR fails) = 1	3.19E-9	2.53	E12
P(RWCU fails) = 1	1.55E-9	1.23	G33
P(FPCCU fails) = 1	1.26E-9	1	G41
P(both Cnd and RF xfer fails) = 1	1.26E-9	1	P11
P(CRD fails) = 1	1.26E-9	1	C11
P(Fire water fails) = 1	1.26E-9	1	P64
P(HPCS Unavailable) = 1	1.28E-9	1.02	E22
P(LPCS Unavailable) = 1	1.27E-9	1.01	E21
P(LPCI/RHR Fails) = 1	2.17E-8	17.22	E12

These calculations suggest that only system E12, which contains both RHR and ADHR, has a significant outage RAW.

A.4 Evaluating Safety Significance for Systems that Mitigate Radionuclide Releases

Several Grand Gulf systems have a role in mitigating radionuclide releases. Grand Gulf's Level 2 IPE model represents these systems and their functions in a Containment Event Tree (CET). The CET logic model assumes core damage as a starting point, and maps out sequences that represent various radionuclide release categories.

Grand Gulf used its CET model to rank the risk importance of systems that can mitigate large radionuclide releases. The calculation procedure involved the following steps.

- Perform four risk importance calculations for each system. One pair of calculations assumes operator actions follow Grand Gulf's current EOPs. The other pair assumes operator actions follow an expected revision to the EOPs. One calculation in each pair measures a standard RAW. The other measures a modified-RAW by assuming the chance of system failure increases by 10%. Depending on the system, these numerical assumptions may apply to one or more places in the model (i.e., as answers to CET "questions").
- Apply weighting factors to various release categories for CET sequences. These weights approximate the relative public risk of each release category, and substitute for detailed off-site consequence calculations.
- Solve the CET model, and record the weighted sum for the frequency of large, early releases for each of the four cases, for each system.
- Compare the results to the baseline risk level. Assign the system that causes the greatest risk increase a rank of 1, the next highest risk increase a rank of 2, and so on.
- Tabulate the results, and identify the systems that have the highest risk importance over all four cases.

Table A-4 summarizes the results of these ranking calculations. It highlights four functions, representing six systems that are risk significant across all four cases. This risk ranking is therefore insensitive to either: (1) the chance of changing Grand Gulf's EOPs, and (2) the method of calculating RAW.

Table A-4 Importance Ranking for Systems Mitigating Radionuclide Releases

[Ranking: 1= Highest Increase in Eqvt. Releases]

System	SSC Sensitivity Case	Original EOP		New EOP: Revised MSIV Venting	
		RAW	10% Increase	RAW	10% Increase
E12	SPC	11	10	11	10
E30	SPMU	9	4	9	4
E12	Contmt Spray	4	9	7	9
P64	Fire Water	6	4	4	4
E12,E21,E22	ECCS	1	1	1	2
E61	DW Vac Bkrs	2	3	2	3
M24	Drywell	3	2	3	1
M10,M23	Containment	10	11	10	11
M41	Contmt Venting	5	4	8	4
E61	CGCS - Igniters	6	4	4	4
	BASE CASE	6	4	4	4

indicates SSC is relatively more important

indicates SSC is relatively less important

A.5 Safety Function Success Criteria

[This subsection is an excerpt from Grand Gulf's seismic IPEEE report.[1] It describes the plants critical safety functions and the system success criteria for those functions.]

The four principal safety functions that are required to achieve and maintain a safe shutdown condition are:

- Reactivity Control
- Reactor Pressure Control
- Reactor Inventory Control
- Decay Heat Removal

In addition, the containment function must also be reviewed for a seismic IPEEE. The primary functions include containment integrity, containment isolation, and prevention of bypass. As such, containment isolation valves, any other components which could cause an early containment bypass, and the components which actuate them will be included in the safe shutdown equipment list and will be walked down. The anchorages and spatial systems interactions for cabinets/panels housing components on the safe shutdown equipment list will also be reviewed.

2. Safe Shutdown Success Paths

Safe shutdown success paths are developed to identify the systems that must function to successfully shutdown and cool the reactor following the occurrence of a review level earthquake. A safe shutdown success path is a string of systems which is used to accomplish all of the required safe shutdown functions. This success path can be depicted in a Shutdown Path Logic Diagram (SPLD).

2.1 Principal Safety Functions

The first step in the development of the SPLDs is to define the safety functions that must be accomplished to achieve and maintain a stable shutdown. The four functions listed above were identified in EPRI NP-6041. These functions are reactivity control, reactor coolant system pressure control, reactor coolant inventory control and decay heat removal. The GGNS IPE [Reference 3] also identified safety functions that must be accomplished to successfully mitigate the events analyzed in the IPE. The initiating events of interest for this evaluation are Loss of Offsite Power and small LOCA as indicated in Section 1. The success criteria for these initiating events are listed in Tables 2.1 and 2.2. The safety functions are listed across the top of the page in these two tables and are equivalent to the safety functions from EPRI NP-6041 except for the early containment over pressure protection function for the small LOCA initiating event. This function is accomplished by the successful operation of the vapor suppression system. The vapor suppression system consists of the weir wall inside the drywell, the drywell to suppression pool vents and the suppression pool. These

[1] Grand Gulf Nuclear Station Engineering Report for Selection of Safe Shutdown Paths and Equipment for the GGNS Seasmic IPEEE.

components are all passive in nature and need only maintain their structural integrity in order to accomplish the function. These will be reviewed as part of the containment structural review. Therefore, the four primary safety functions identified above will form the basis for the identification of the frontline systems for the safe shutdown paths.

2.2 Safety Function Success Criteria

The systems that can accomplish each of the primary safety functions must now be defined. During the performance of the GGNS IPE, combinations of systems that are required to successfully function to successfully accomplish each safety function were identified. Tables 2.1 and 2.2 provides a listing of these systems for the loss of offsite power and the small LOCA initiators. The systems that can successfully perform each safety function are summarized below.

Reactivity Control	Reactor Protection System and Control Rod Drive System Alternate Rod Insertion (and CRD System) and Reactor Pump Trip Manual Rod Insertion (and CRD System) and Reactor Pump Trip Standby Liquid Control System and Reactor Pump Trip
Reactor Pressure Control	Steam Line Safety Relief Valves Power Conversion System (MSIVs and Condenser)*
Reactor Inventory Control	Feedwater* High Pressure Core Spray System# Reactor Core Isolation Cooling System# Control Rod Drive System (injection mode) Low Pressure Core Spray and Depressurization with at least 4 SRVs# Low Pressure Core Injection and Depressurization with at least 4 SRVs# Condensate and Depressurization with at least 4 SRVs* Standby Service Water Crosstie to LPCI and Depressurization with at least 4 SRVs Firewater and Depressurization with at least 4 SRVs Suppression Pool Makeup
Decay Heat Removal	Power Conversion System* Suppression Pool Cooling Mode of RHR Containment Spray Cooling Mode of RHR Shutdown Cooling Mode of RHR Containment Venting

* Not available with Loss of Offsite Power

\# Suppression Pool Makeup required for these systems if LOCA

The above systems also require various support systems for successful operation. The front line to support system dependencies are identified in Tables 2.4 and 2.5.

2.3 Overall Success Path Logic Diagram

With the identification of the primary safety functions and the systems that can accomplish those safety functions an overall success path logic diagram (SPLD) can be developed for GGNS. A SPLD is a graphic representation that shows the combinations of systems whose successful operation will result in long term shutdown following the seismic margin earthquake. It can be envisioned as a simple electrical circuit diagram constructed in a series-parallel fashion. The Seismic margin earthquake is depicted as the node on the left, and the desired long-term safe shutdown condition is depicted as the node on the right. Between these two nodes are a number of system blocks arranged in a series-parallel manner, showing alternate paths of achieving the safe shutdown condition. The selected paths must represent paths that the control room operators will use based upon their training and procedures. Adequate instrumentation must also be available to the operators. An overall SPLD for GGNS is provided in Figure 2.1. This SPLD was constructed using the success criteria for a Loss of Offsite Power and Small LOCA initiators as discussed above. Note that some of the systems that are capable of being utilized are not included in the GGNS overall SPLD since they are not the systems that operators would preferentially use. It should be noted that support systems are also not included in the tables or Figure 2.1. Dependencies between front line systems and support systems are identified in Table 2.4.

The first node on the Overall SPLD is the function of Reactivity Control. This node consists of two parallel paths. One path is made up of the control rod drive system which works in conjunction with the reactor protection system. This path represents the insertion of control rods into the core in response to an automatic scram signal generated by RPS to shutdown the reactor. An automatic scram signal can be generated by any one of several results of a seismic event including the loss of offsite power. The second parallel path is made up of the standby liquid control system (SLCS) block. This block represents the injection of sodium pentaborate into the reactor coolant system such that the reactor is shutdown. The SLCS is only actuated manually and must be actuated quickly. EPRI NP-6041 recommends that SLCS not be relied upon because of assumed stress levels on the operators during a seismic event. Even though GGNS does not completely agree with this assumption because of the strength of the emergency operating procedures (EOPs), the high degree of operator training and the culture of adherence to procedure, this assumption will be maintained for this analysis. Therefore, no credit will be taken in the preferred and alternate path selections. The other methods of reactivity control are not included since they rely on the CRD system also.

The second node in the SPLD represents the function of Reactor Pressure Control. This block represents the opening and closing of the SRVs to control reactor pressure.

Because of the assumption of loss of offsite power no credit is taken for the power conversion system as it will lead to the isolation of the main steam isolation valves and prevent use of the condenser to control pressure.

The third node of the SPLD represents the function of Reactor Inventory Control. This node consists of two primary parallel paths. One path represents inventory control with high pressure systems and the other represents inventory control with low pressure systems. The high pressure path consists of two blocks in parallel. The top block represents the injection of water into the core at high pressure by the RCIC system. The lower block represents the injection of water into the core at high pressure by the HPCS system. Other high pressure inventory control methods (feedwater and CRD injection) are not included because of unavailability because of the basic assumptions or low capacity. The second path represents inventory control with low pressure systems and consists of two parts. The first part consists of the Automatic Depressurization System (ADS) block. This block represents depressurization of the reactor using the SRVs. Note that the EOPs direct the operators to inhibit automatic depressurization. Therefore, depressurization is only performed manually since the operators will follow the EOPs. Depressurization can be accomplish through the use of any of the SRVs but only the ADS valves are credited in this analysis. Depressurization is always required for use of the low pressure systems for inventory control. This is true even with the assumption that a small LOCA exists since the assumed break size is too small to depressurize the reactor in sufficient time to allow low pressure systems to inject prior to core damage. The top block represents the low pressure coolant injection (LPCI) mode of RHR. The bottom parallel path represents the low pressure core spray (LPCS) system. Other methods of low pressure inventory control are not included on the SPLD. Condensate would not be available because of the assumption of loss of offsite power. SSW crosstie to LPCI and firewater could be used but SSW crosstie is a lower priority system for injection in the EOPs and firewater would only be effective after inventory had been maintained by some other system for a period of several hours. The final block, which is in series with both parallel paths for reactor inventory control represents the suppression pool makeup (SPMU) system. This system is required <u>only</u> if a LOCA is assumed. It is necessary because of the loss of inventory from the suppression pool to the drywell with a LOCA inside the drywell. Systems such as LPCI and LPCS which take suction on the suppression pool could loose net positive head if this inventory loss if not made up. Both RCIC and HPCS can take suction from the condensate storage tank in addition to the suppression pool. However, the condensate storage tank is not credited for this analysis.

The fourth and final node of the SPLD represents the function of Decay Heat Removal. This node consists of two parallel success paths. The top path represents the removal of decay heat using the suppression pool cooling (SPC) mode of the RHR system.

The lower block represents the decay removal with the shutdown cooling (SDC) mode of the RHR system. Both of these modes utilize the RHR heat exchangers.

Note that for the SPC mode to work, decay heat from the core must be rejected to the suppression pool through the SRVs. This also requires a continued means of making up inventory to the reactor vessel. Other methods of decay heat removal not included on the SPLD are the containment spray mode (CS) of RHR and containment venting. Containment venting is not included since this method requires instrument air which may not be available during a loss of offsite power. CS mode is not included because its use is lower in priority to SPC and SDC in the EOPs.

Table 2.1
(Table 3.1-6 of IPE Summary Report)
Success Criteria for a Loss of Offsite Power Transient

INITIATOR	REACTOR SUBCRITICALITY	RCS OVERPRESSURE PROTECTION	EMERGENCY CORE COOLING	LATE CONTAINMENT OVERPRESSURE PROTECTION
TI	RPS or ARI & RPT or Manual Rods & RPT or Timely SLC & RPT	SRVs Open & Close	HPCS or RCIC or CRD (Maximized Flow) [See Note A] or {DEP W/4 SRVs and LPCS or 1 of 3 LPCI or SSW Crosstie or Firewater [See Note B]}	{1 of 2 RHR (SPC, SDC, or CS) and Associated SSW} or Containment Venting

NOTES :

A) Operation of both CRD (i.e., maximized flow) pumps is only successful when the vessel is at high pressure. One CRD pump is sufficient if CRD is only used in the long term (i.e., when coolant makeup has been provided for a period of time).

B) Firewater is only capable of providing sufficient injection in the long term (i.e., when coolant makeup has been established for a period of time).

Table 2.2
(Table 3.1-13 of IPE Summary Report)
Success Criteria for Small LOCAs

INITIATOR	REACTOR SUBCRITICALITY	EARLY CONTAINMENT OVERPRESSURE PROTECTION	EMERGENCY CORE COOLING	LATE CONTAINMENT OVERPRESSURE PROTECTION
S2	RPS or ARI & RPT or Manual Rods & RPT or Timely SLC & RPT (For Steam Line Break) [See Note A]	PCS and 1 of 2 SPMU	1FW or {HPCS or RCIC or [DEP W/4 SRVs and LPCS or 1 of 3 LPCI] and 1 of 2 SPMU} or {DEP W/4 SRVs and Condensate or SSW Crosstie or Firewater [See Note B and C]}	PCS or {1 of 2 RHR (SPC or CS) and Associated SSW} or Containment Venting

NOTES :

A) Assumes SLC is ineffective for a liquid break.

B) Depressurization with SRVs necessary in order to use low pressure systems. Three SRVs needed to depressurize based on the sizing of a Large LOCA (i.e., equivalent area of approximately 4 stuck open relief valves).

C) Firewater is only capable of providing sufficient injection in the long term (i.e., when coolant makeup has been established for a period of time).

Table 2.3 Front-line Systems

FUNCTION	PREFERRED PATH	ALTERNATE PATH*
Reactivity Control	CRD	CRD
Pressure Control	SRVs in relief mode for initial transient (Div1)	SRVs in relief mode for initial transient (Div2)
Inventory Control	RCIC HPCS SPMU A	ADS (Div 2) LPCI C
Decay Heat Removal	RHR A in SPC (Hot Shutdown)	RHR B in SDC (Cold Shutdown)

* LOCA is not assumed for the Alternate Path.

Table 2.4 FRONT LINE TO SUPPORT SYSTEM DEPENDENCY MATRIX*

	HPCS	RCIC	CRD	LPCS	LPCI			SSW/ RHR X-TIE	FIRE WATER INJ	ADS	RHR/SDC		RHR/SPC		RHR/CS		PCS	CTMT VENT	SPMU		SLC	
					A	B	C				A	B	A	B	A	B			A	B	A	B
ESF AC DIV I			X	X	X				X		X	X	X		X			X	X		X	
ESF AC DIV II			X			X	X	X	X		X	X		X		X		X		X		X
ESF AC DIV III	X																					
BOP AC			X						X								X					
ESF DC DIV I		X	X	X	X					X	X		X		X			X	X			
ESF DC DIV II		X[9]	X			X	X			X		X		X		X		X		X		
ESF DC DIV III	X																					
BOP DC																	X					
SSW TRAIN A					X						X		X		X							
SSW TRAIN B						X	X	X				X		X		X						
CCW			X																			
TBCW																	X					
PSW																						
CHILLED WTR																						
CIRC WTR																	X					
INST AIR			X[10]						X	X[8]							X	X				
ECCS Rm HVAC	X			X[2]	X	X	X'	X			X	X	X	X	X	X						
Steam Tunnel HVAC		X[1]															X					

*Dependencies are indicated for the systems in the column header.

Table 2.5 SUPPORT SYSTEM TO SUPPORT SYSTEM DEPENDENCY MATRIX*

	DG			DGX		SSW			CCW	TBCW	PSW	CHLD WTR	INST AIR	CIRC WTR	DG Rm HVAC	SSW Pump House Vent		ECCS Rm HVAC	STM Tnl HVAC	AC Power			DC Powe		
	I	II	III	I	II	A	B	C								A	B			I	II	III	I	II	III
ESF AC DIV I						X					X		X		X	X		X							
ESF AC DIV II							X		X[3]		X		X		X	X	X	X							
ESF AC DIV III				X	X			X							X			X							
BOP AC									X	X	X	X	X	X					X						
ESF DC DIV I	X					X					X		X												
ESF DC DIV II		X					X		X		X		X												
ESF DC DIV III			X	X	X			X																	
BOP DC									X[11]	X[12]	X[11]	X[12]	X[13]	X											
SSW TRAIN A	X																	X							
SSW TRAIN B		X							X[4]				X					X							
SSW TRAIN C			X															X							
TBCW													X												
PSW									X	X		X													
CHILLED WTR																			X						
INST AIR											X	X													
DG Rm HVAC	X	X	X																						
SSW Pump House Vent A						X[5]		X[6]																	
SSW Pump House Vent B							X[5]																		
Switchgerr & Batt Rm Cooling Div I																				X			X		
Switchgerr & Batt Rm Cooling Div 2																					X			X	

*Dependencies are indicated for the systems in the column header.

Notes for Tables 2.4 and 2.5

1. Delayed time dependency. The RCIC pump will operate for 30 minutes after the steam leak detection signal is initiated. No isolation occurs during SBO due to loss of power to the timer.

2. Delayed time dependency. LPCS Pump will operate approximately 10 to 12 hours without room cooling.

3. Train B pump.

4. SSW Train B is alternate source of cooling water under certain conditions.

5. Delayed time dependency. SSW pumps will fail approximately 2.5 hours after loss of HVAC.

6. Delayed time dependency. SSW pumps will fail approximately 2.5 hours after loss of HVAC. No dependency if SSW A pump is not operating.

7. Deleted
8. Backup by accumulators.

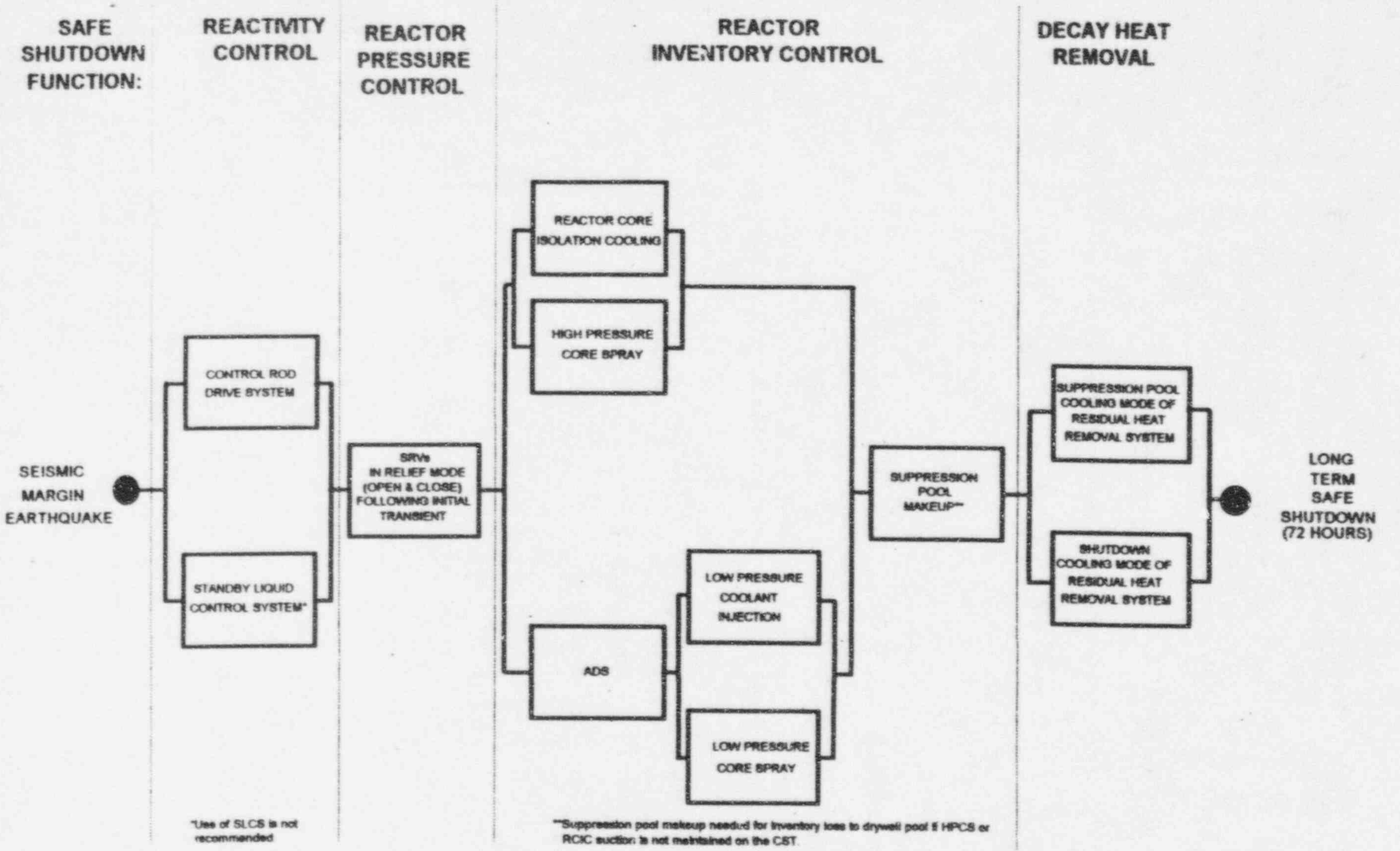
9. Required for redundant actuation logic and Level 8 protection instrumentation.

10. Required for enhanced flow mode only.

11. DC power is required to start the pumps. However, the pumps are normally operating and DC power is not modeled.

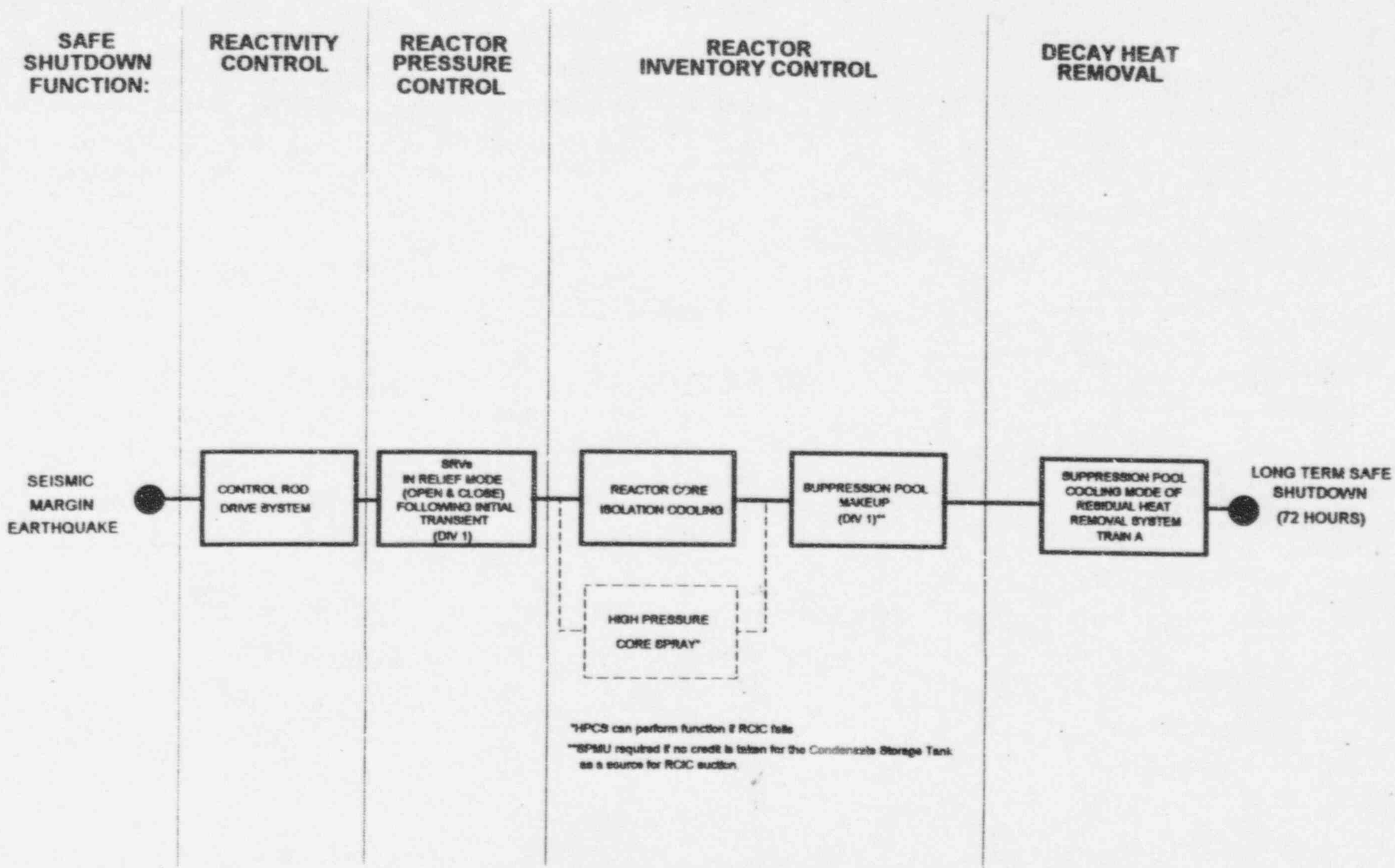
12. DC power is required to start standby pump.

13. DC power is required to start the normally operating compressor. Therefore, DC power is not modeled.



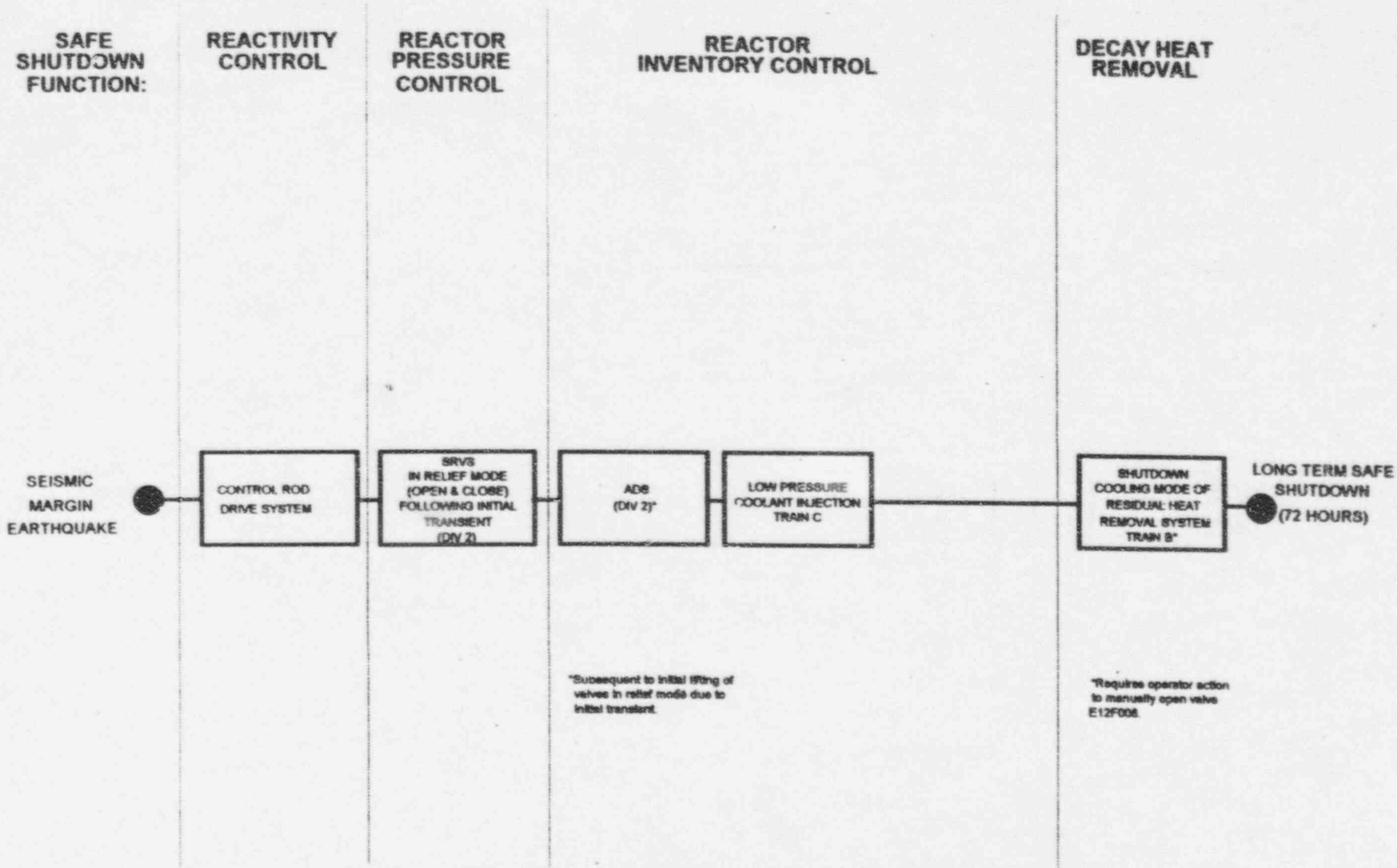
GGNS PROPOSED SUCCESS PATH LOGIC DIAGRAM

Figure 2.1 GGNS Proposed Success Path Logic Diagram



GGNS IPEEE SEISMIC MARGIN ASSESSMENT SUCCESS PATH--PREFERRED

Figure 2.2 GGNS Preferred Success Path



GGNS IPEEE SEISMIC MARGIN ASSESSMENT SUCCESS PATH–ALTERNATE (ASSUMES NO LOCA)

Figure 2.3 GGNS Alternate Success Path

Appendix B

B.1 Confirming Criteria for Non-QASS Systems

The criteria described below are used to confirm a system as non-QASS. These criteria are *designed for use only after concluding that a system meets none of the criteria in Section 2.*[1]

B.1.1 Confirming Criterion (a)

If a system is not modeled in the PSA, then it can be said that the system "has no functional relationship" to either core damage risk or the risk of large early radionuclide releases.

The Grand Gulf PSA includes those systems shown in Table B-1.

Table B-1 Systems in the Grand Gulf PSA

System	System Description
B21	Nuclear Boiler-MSRVs
C11	CRD System
C41	Standby Liquid Control
C71	Reactor Protection Sys.
E12	LPCI & SDC
E21	LPCS
E22	HPCS
E30	Suppression Pool Make-up
E31	Leak Detection System
E51	RCIC
E61	Combustible Gas Control
G33	RWCU
L11	125v Batteries
L21	125v Swgr & Dist
L51	125v Battery Chargers
M41	Containment Cooling Sys
N11	Main & Reheat Steam
N19	Condensate
N21	Feedwater
N71	Circulating Water
P11	Condensate & Refueling Water Storage & Transfer
P41	Standby Service Water
P42	CCW
P43	Turb Bldg Cooling Water
P44	Plant Service Water
P52	Service Air
P53	Instrument Air
P64	Fire Protection
P71	Plant Chilled Water
P75	Standby Diesel
P81	HPCS Diesel
R11	4.16kv xformers

[1] One might read every criterion in this section with the preface "Systems that do not meet the criteria in Section 2.1 *and* are..."

Table B-1 Systems in the Grand Gulf PSA

System	System Description
R12	6.9 kv xformers
R20	480v Load & MCCs
R21	4.16 kv Swgr & Load Shedding & Sequence Panels
R27	500 kv Ckt Bkrs
R28	120/280 Dist & Lighting Panels
T41	Aux Bldg Ventilation
T46	ESF Elec Swgr Room Cooling
T51	Emerg Pump Room Ventilation
X77	Diesel Gen Bldg Ventilation
Y47	Standby Service Water Pumphouse Ventilation

If a system appears in Table B-1, then it can not satisfy this criterion. Of Grand Gulf's *other* systems, those that are non-QASS satisfy this criterion.

Criterion: **If a system has no functional relationship to either core damage risk or the risk of large, early radionuclide releases, then it can be classified as non-QASS.**

B.1.2 Confirming Criterion (b)

Some Grand Gulf's system numbers apply to items that are physically remote from the power plant, or to maintenance programs that do not comprise any specific equipment. A list of examples appear in Table B-2.

Table B-2 GGNS Systems that Do Not Affect Plant Operation

System	System Description
G50	CRD Maintenance
X37	Warehouse Ventilation
Y70	Sewage Treatment Plant

Criterion: **If a system is either: (i) a physical feature located away from the power block and having no relationship to plant operation, or (ii) not a plant physical feature, then it can be classified as non-QASS.**

B.1.3 Confirming Criterion (c)

Table B-3 is a list of systems that are modeled in the PSA, but do not satisfy the numerical screening criteria for the Maintenance Rule. Appendix A.1 describes the numerical criteria for risk importance that the Maintenance Rule expert panel applied.

Table B-3 Systems Modeled in the PSA But With Low Safety Significance

System	System Description
E31	Leak Detection System
G33	RWCU
L51	125v Battery Chargers
M41	Containment Cooling Sys
N11	Main & Reheat Steam

Table B-3 Systems Modeled in the PSA But With Low Safety Significance

System	System Description
N19	Condensate
N21	Feedwater
N71	Circulating Water
P52	Service Air
P71	Plant Chilled Water
R11	4.16kv xformers
R28	120/280 Dist & Lighting Panels
T41	Aux Bldg Ventilation

Criterion: **If a system is modeled in the PSA and does not meet the Maintenance Rule's screening criteria for risk significance, and meets no other expert panel test for safety significance, then it can be classified as non-QASS.**

B.1.4 Confirming Criterion (d)

Some systems are not modeled in the PSA because their chance of failure is considered to be very low. Some reasons why a system's failure probability might be considered low include:

- There is a technical analysis showing that the plant can tolerate a loss of the system's function for up to 24 hours without affecting risk. (This applies to system T46)
- Other requirements besides QA requirements provide an adequate level of confidence that the system can perform as designed. (This applies to system Y21.)
- Selected components within the system are also included within the Mxx system, and the plant can tolerate a loss of other system functions without affecting risk.

Table B-4 contains examples of systems that might have a role in minimizing risk, but are considered to have a low failure probability.

Table B-4 Systems With a Potential Role in Risk Management, but With a Very Low Failure Probability

System	System Description
C82	Plant Annunciators
Y21	Yard Substructure
Y60	Fire Water Pumphouse
Z77	Emerg Swgr & Battery Rooms Ventilation
T46	ESF Elec. Swgr & Battery Room Ventilation

Criterion: **If a system has a potential role in minimizing risk, but its failure probability is considered very low, then it can be classified as non-QASS.**

B.1.5 Confirming Criterion (e)

Some systems are not modeled in the PSA because the functional relationship between the system and plant risk measures is too weak to quantify. This often occurs when the system's role is one of supporting risk significant operator actions for accident mitigation, and there are numerous, redundant cues and alternatives available to operators. Some examples include the systems listed in Table B-5.

Table B-5 Systems that may be used in accident mitigation, but whose failure probability is too Weak to Quanify

System	System Description
C84	Meteorological Monitoring
C93	Emergency Response Facilities
M92	Lighting, Communication & Fire Alarm
R61	Public Address & Intercommunication
T92	Lighting, Communication & Fire Alarms
U92	Lighting, Communication & Fire Alarms
V92	Lighting, Communication & Fire Alarms
X40	EOF Ventil Rad Monitors
X41	EOF Area Rad Monitors
X45	EOF Doors
X46	EOF Emerg Diesel Gen
X47	Emerg Ops Facility HVAC
Y92	Lighting, Communication & Fire Alarms
Z17	Control Bldg HVAC
Z92	Lighting, Communication & Fire Alarms

Criterion: **If a system is not modeled in the PSA because of the functional relationship between the system and the plant risk measure is too weak to quantify, then it can be classified as non-QASS.**

B.1.6 Confirming Criterion (f)

Some systems are not modeled in the PSA because they are highly reliable or "passive" systems. Passive items include cables, raceways, underground piping, and structures.

These systems might be considered risk significant in the context of external events such as fires, floods, or seismic events. For some of these potentially risk significant systems, other design requirements besides those imposed by 10 CFR 50 Appendix B are sufficient to assure that their chance of failure remains low.

Table B-6 Highly Reliable Structures or Passive Systems with other Requirements besides QA that are sufficient to assure reliability during accidents

System	System Description
R30	Specialty Cable
R31	Instrument and Computer Cable
R33	Coaxial and Triaxial Cable
R34	Thermocouple Extension Cable
R35	600 V Multiconductor Control Cable
T10	Auxiliary Bldg
V21	Radwaste Substructure
V22	Radwaste Superstructure
X70	Diesel Gen Bldg Structure
X72	DG Bld Substructure
X73	DG Bld Superstructure
Y21	Yard Substructure
Y22	Yard Superstructure
Y35	Underground Piping
Y40	Standby Water Basis Substructure
Y41	Standby Water Basin Superstructure
Y91	Raceway
Z10	Control Bldg

Criterion: **<u>If a system is either: (i) a highly reliable structure or (ii) a passive system with other requirements besides QA that are sufficient to assure reliability during accidents, then it can be classified as non-QASS.</u>**

B.2 Confirming Criteria for Non-QASS Components

The criteria described below are used to confirm a component as non-QASS. These criteria are *<u>designed for use only after concluding that a component meets none of the criteria in Section 3.1 (except for the "catch-all" criterion described in Section 3.1.3</u>*.[1]

B.2.1 Criterion L1:

Criterion: **<u>If a component is not modeled in any of the Grand Gulf risk studies, and it is non-Q, then it can be classified as non-QASS.</u>**

[1] One might read every criterion in this section with the preface "Components that do not meet the criteria in Section 3.1 *<u>and</u>* ..."

B.2.2 Criterion L2:

Some components can fail without causing a loss of the safety function modeled in the PSA. These components do not have a significant safety impact.

Criterion: **If a component is not required for the system function modeled in the PSA, then it can be classified as non-QASS (or NSS).**

B.2.3 Criterion L3:

Piping systems can generally tolerate small flow diversions. The PSA model generally assumes that flow diversions in piping less than 1/3 the diameter of the main system flow path will not prevent system success.

Examples of the types of components identified as satisfying this criteria include:

- manual valves
- relief valves

Criterion: **If a component is in a flow path that could create only a small flow diversion, then it can be classified as non-QASS (or NSS).**

B.2.4 Criterion L4:

The failure rates for passive components are usually 100 times lower than failure rates for active components. This disparity in failure rates is due to several reasons, not necessarily tied to QA requirements.

Examples of the types of components identified as satisfying this criteria include:

- Piping
- Normally locked (or administratively controlled) isolation valves
- Tanks and vessels
- Piping orifices and flow elements
- Cables and Wiring
- Hand switches in auto (with a "spring return to auto" feature), where the auto position does not affect a safety function
- Other components, instruments, and valves, whose only function is to maintain pressure boundary integrity

Criterion: **If a passive non-active component is considered highly reliable regardless of its QA status, then it can be classified as non-QASS (or NSS).**