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CE NPSD-1208/WCAP-16125
Project Number 694

Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Attention: Chief, Information Management Branch,
Division of Program Management

Subject: Westinghouse Owners Group Transmittal of WCAP-16125, Rev. 0, "Justification for Risk-Informed Modifications to Selected Technical Specifications for Conditions Leading to Exigent Plant Shutdown"

This letter transmits four copies of WCAP-16125-NP, Rev. 0 (non-proprietary), "*Justification for Risk-Informed Modifications to Selected Technical Specifications for Conditions Leading to Exigent Plant Shutdown.*" WCAP-16125-NP is a revision to CE NPSD-1208 that was originally submitted to the NRC for review through NEI on January 24, 2001. This transmittal is for the purpose of assisting the staff in completing the review of CE NPSD-1208/WCAP-16125.

WCAP-16125 provides a risk-informed justification to extend the allowed outage time or completion time for various Technical Specification 3.0.3 action statements. An informal Request for Additional Information (RAI) was issued on May 21, 2001 and responses to these RAIs issued on February 20, 2002. The enclosed Revision 1 to CE NPSD-1208/WCAP-16125, Rev. 0 incorporates these RAI responses along with the results of confirmatory supplemental risk calculations and recently approved changes to the ISTS associated with an inoperable radiological boundary.

Staff review of CE NSPD-1208 has been performed on a fee waived basis in support of the Risk Informed Technical Specification improvement program Initiative 6B, "Extend Time in LCO 3.0.3 to Initiate Shutdown and Develop Risk Informed Actions Based on Plant Configurations." The WOG expects that the staff will complete their review of CE NPSD-1208/WCAP-16125 on a fee waived basis.

Westinghouse is prepared to discuss WCAP-16125-NP, if needed, in order to facilitate the staff's review. If you require further information, feel free to contact Mr. Paul Hijeck, Owners Group Project Office at 860-731-6240.

Very truly yours,

Robert H Bryan

Robert H. Bryan, Chairman
Westinghouse Owners Group

Westinghouse Non-Proprietary Class 3

WCAP-16125-NP, Revision 0
(CE NPSD-1208, Revision 01)

September 2003

Justification for Risk-Informed Modifications to Selected Technical Specifications for Conditions Leading to Exigent Plant Shutdown

Tasks 1115, 2035

Westinghouse

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Westinghouse Electric Company LLC
2000 Day Hill Road
Windsor, Connecticut 06095-0500

WCAP-16125-NP, Rev. 00

(CE NPSD-1208, Rev. 01)

Justification for Risk-Informed Modifications to Selected Technical Specifications for Conditions Leading to Exigent Plant Shutdown

Tasks 1115, 2035
Final Report

September 2003

Author: ______________________
Raymond E. Schneider
Reliability & Risk Assessment

Reviewer: ______________________
Rupert A. Weston
Reliability & Risk Assessment

Approved: ______________________
Gary A. Brassart
Reliability & Risk Assessment

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EXECUTIVE SUMMARY

This report addresses one of several industry based initiatives to support the development of Risk-Informed Plant Technical Specifications.

Specifically, this report justifies modifications to various Technical Specification (TS) Action Statements for the conditions that result in a loss of safety function related to a system or component included within the scope of the plant TSs. It is proposed that the current Required Action be changed from either a default or explicit 3.0.3 entry (or equivalent action) to a risk-informed action based on the system's risk significance. In most instances, an Allowed Outage Time (AOT)/Completion Time (CT) of 24 hours is proposed.

The proposed TS changes discussed in this report are summarized in Table 2-1. These changes are risk-informed and are in conformance with RG 1.174 and RG 1.177, as appropriate. Risk assessments performed to support these modifications are based on bounding analyses and are applicable to Combustion Engineering (CE) designed Nuclear Steam Supply Systems (NSSSs) operated in the United States. Furthermore, the risk associated with the implementation of these TS changes will be managed in accordance with paragraph (a) (4) of 10 CFR 50.65 (Maintenance Rule).

The benefit derived from these changes is that the proposed AOT/CT extensions provides needed flexibility in the performance of corrective maintenance of these components during power operation. These actions will avert the costs and risks associated with plant shutdowns and ensure that the public health and safety is preserved.

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ACRONYMS

ADV	-	Atmospheric Dump Valves
AFW	-	Auxiliary Feedwater
ANO	-	Arkansas Nuclear One
AOT	-	Allowed Outage Time
ATWS	-	Anticipated Transient without Scram
BAMU	-	Boric Acid Makeup
BAT	-	Boric Acid Tank
BV	-	Block Valve
CARC	-	Containment Air Recirculation Cooling
CC	-	Containment Cooling
CCFP	-	Conditional Containment Failure Probability
CCDP	-	Conditional Core Damage Probability
CDF	-	Core Damage Frequency
CDP	-	Core Damage Probability
CE	-	Combustion Engineering
CEA	-	Control Element Assembly
CEOG	-	Combustion Engineering Owners Group
CIAS	-	Containment Isolation Actuation Signal
CIV	-	Containment Isolation Valve
CLERP	-	Conditional Large Early Release Probability
CR	-	Control Room
CRC	-	Control Room Cooling
CREACUS	-	Control Room Emergency Air Cleanup System
CREATCS	-	Control Room Emergency Air Temperature Control System
CRV	-	Control Room Ventilation
CS	-	Containment Spray
CTMT	-	Containment
DBA	-	Design Basis Accident
DCH	-	Direct Containment Heating
EACS	-	Emergency Air Cleanup System
EATCS	-	Emergency Air Temperature Control System
ECCS	-	Emergency Core Cooling System
ECCS PREACS		ECCS Pump Room Exhaust Air Clean Up System
ECW	-	Emergency Chilled Water
ESF	-	Engineered Safety Feature
FCS	-	Fort Calhoun Station
GDC	-	General Design Criterion
HEPA	-	High Efficiency Particulate Air
HPME	-	High Pressure Melt Ejection
HPSI	-	High Pressure Safety Injection
hrs	-	Hours
ICCDP	-	Incremental Conditional Core Damage Probability

ACRONYMS CONTINUED

ICLERP	-	Incremental Conditional Large Early Release Probability
ICS	-	Iodine Cleanup System
IEF	-	Initiating Event Frequency
ISTS	-	Improved Standard Technical Specifications
LBLOCA	-	Large Break Loss of Coolant Accident
LCO	-	Limiting Conditions for Operation
LERF	-	Large Early Release Frequency
LER	-	Large Early Release
LERP	-	Large Early Release Probability
LOCA	-	Loss of Coolant Accident
LOFW	-	Loss of Feedwater
LOOP	-	Loss of Offsite Power
LPSI	-	Low Pressure Safety Injection
LTOP	-	Low Temperature Overpressure Protection
MBLOCA	-	Medium Break Loss of Coolant Accident
MFW	-	Main Feedwater
MHA	-	Maximum Hypothetical Accident
MP2	-	Millstone Plant Unit 2
MSLB	-	Main Steam Line Break
MSSV	-	Main Steam Safety Valves
MTC	-	Moderator Temperature Coefficient
NPSH	-	Net Positive Suction Head
NRC	-	Nuclear Regulatory Commission
PORV	-	Power Operated Relief Valve
PRA	-	Probabilistic Risk Assessments
PREACS	-	Penetration Room Emergency Air Cleanup System
PSA	-	Probabilistic Safety Analysis
PSV	-	Pressurizer Safety Valve
PVNGS	-	Palo Verde Nuclear Generating Station
PWR	-	Pressurized Water Reactor
RCS	-	Reactor Coolant System
RG	-	Regulatory Guide
RPS	-	Reactor Protection System
RPV	-	Reactor Pressure Vessel
RWST	-	Refueling Water Storage Tank
SB	-	Shield Building
SBLOCA	-	Small Break Loss of Coolant Accident
SBEACS	-	Shield Building Exhaust Air Cleanup System
SDC	-	Shutdown Cooling
SDM	-	Shutdown Margin
SG	-	Steam Generator

ACRONYMS CONTINUED

SGD	-	Steam Generator Depressurized
SGTR	-	Steam Generator Tube Rupture
SIAS	-	Safety Injection Actuation Signal
SIT	-	Safety Injection Tank
SONGS	-	San Onofre Nuclear Generating Station
SL	-	St. Lucie
TS	-	Technical Specification
WSES	-	Waterford Steam Electric Station

1.0 PURPOSE

This report provides the technical justification for proposed risk-informed modifications to Technical Specifications (TSs) such that unnecessary exigent plant shutdowns resulting from entry into Limiting Condition for Operation (LCO) 3.0.3 (or equivalent ACTION STATEMENTS) may be avoided. The proposed modifications are typically associated with plant conditions when two redundant trains of a system are inoperable resulting in the loss of a safety function, and there is either no Action for the condition (requiring a default LCO 3.0.3 entry) or conditions exist where the specific Action includes a 1 hour shutdown requirement (explicit LCO 3.0.3 entry). The intent of these modifications is to provide a risk-informed alternative to the current LCO 3.0.3 requirements such that the plant staff has adequate time to resolve a significant loss of function while the plant remains operating. Resolving the issue while the plant is at power is often the lowest risk state. In those rare instances where a repair at power is attempted but is unsuccessful, and a delayed shutdown is still required, the additional planning time will reduce risks during plant transition while incurring negligible incremental risks to the public health and safety. The net impact of these proposed modifications is considered risk neutral.

The risk-informed assessment provided in this report follows the general guidance of Regulatory Guide (RG) 1.174 and RG 1.177 (References 1 and 2, respectively). The modifications proposed in this report are applicable to all domestic Combustion Engineering (CE) designed NSSSs. Plant specific assessments are provided where plant uniqueness results in a variation from the risk assessment.

This report, WCAP-16125, updates and supersedes CD NPSD-1208 in its entirety. Also, WCAP-16125 incorporates responses to NRC staff request for additional information on CE NPSD-1208. WCAP-16125 is submitted for staff review in support of Risk-Informed Technical Specification initiatives as embodied in TSTF-426.

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2.0 SCOPE OF PROPOSED CHANGES TO TECHNICAL SPECIFICATIONS

This report justifies modifications to various Technical Specification (TS) Actions for the conditions that result in a loss of function related to a system or component included within the scope of the plant TSs. It is proposed that the current Required Actions be changed from either a default or explicit 3.0.3 entry (or equivalent action) to a risk-informed action based on the system's risk significance. In most instances, a 24 hours AOT/CT is proposed. In specific instances, shorter or longer CTs are proposed, as appropriate. Risk-informed Allowed Outage Times (AOTs) for these TS systems and components are established in Section 4. Table 2-1 summarizes the proposed TS changes to NUREG-1432 (Reference 3) and their associated risk impact. The technical evaluation is also applicable to US fleet of CE designed NSSS with plant specific TS. For purposes of illustration, cross-comparisons of the associated TS LCOs used throughout the US fleet of CE designed NSSSs to NUREG-1432 are presented in Appendix A.

The benefit from these changes is that the proposed AOT extensions provide needed flexibility in the performance of corrective maintenance of these components during power operation. These actions will avert the risks associated with plant shutdowns while ensuring that the public health and safety is preserved.

The methodology for assessing the risk impact of the proposed modifications is presented in Section 4. Section 5 provides the results of the risk-informed evaluation for the various TSs under consideration.

The proposed actions provide a risk-informed process for establishing shutdown priorities and therefore provide adequate protection of the public health and safety. Furthermore, by averting unnecessary plant shutdowns the overall risk of plant operation is reduced.

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Table 2-1: Summary of Risk Impacts Resulting from Proposed Modifications to Technical Specifications							
NUREG-1432	**SYSTEM/ COMPONENT**	**CONDITION**	**CURRENT ACTION / AOT**	**PROPOSED TIME TO RESTORE ONE TRAIN AOT/COMPLETION FOR CONDITION**	**PROPOSED END STATE IF ACTION NOT MET** (See Note 4, 5, 8)	**CCDP** (See Notes 1 & 2)	**CLERP** (See Note 1)
3.1.9 (NA-ISTS)	Boration System	System Inoperable	No Condition defined. Default 3.0.3 entry.	24 hrs	Mode 3 in 6 hrs	4.7E-8	3.4E-9
3.4.9	Pressurizer Heaters	Two Groups of Class 1E Heaters Inoperable	No Condition defined. Default 3.0.3 entry.	24 hrs	Mode 4 in 12 hrs (See Note 9)	3.0E-7 (See Note 10)	1.1E-8
3.4.11	PORVs	Inability of two PORVs to Open, or Inability of both PORVs to close and block valves to be closed	Separate Condition Entry Allowed for each PORV Default to 3.0.3 condition Mode 4 In 12 hrs	8 hrs for conditions in which both PORVs are unable to open or unable to close once challenged, but may be isolated. Extension does not apply to PORVs that are leaking and that cannot be isolated via block valves, or are not expected to be isolable following a demand.	Unchanged	9.2E-7	6.7E-8
3.5.1	SITs	Two or More SITs Inoperable	Explicit 3.0.3 entry	24 hrs	Unchanged	< 1.4E-8	4.1E-11
3.5.2	LPSI (See Note 3)	Two Trains Inoperable	Defined 1 hr shutdown	24 hrs	Unchanged	1.2E-7	3.7E-10
3.5.2	HPSI	Two Trains Inoperable	Defined 1 hr shutdown	4 hrs	Unchanged	< 3.0E-6	< 4.0E-8
3.6.1	CTMT	Inoperable	Restore in 1 hr Shutdown. Mode 5 Entry in 36 hrs.	8 hrs	Unchanged	NA	1.0E-7
3.6.6.1	CSS (See Note 4)	Two Trains Inoperable	Defined 1 hr shutdown	12 hrs if CARC not available 72 hrs if CARC available (reciprocity with TS 3.6.6.8)	Mode 4 in 12 hrs (See Note 9)	7.5E-7 (when CARC not available) Insignificant impact for PWRs with diverse containment cooling systems[6]	(See Note 6)
3.6.10	ICS	Two Trains Inoperable	No Condition defined Default 3.0.3 entry	24 hrs	Mode 4 in 12 hrs (See Note 9)	NA	< 1.0E-7

Table 2-1: Summary of Risk Impacts Resulting from Proposed Modifications to Technical Specifications							
NUREG -1432	SYSTEM/ COMPONENT	CONDITION	CURRENT ACTION / AOT	PROPOSED TIME TO RESTORE ONE TRAIN AOT/COMPLETION FOR CONDITION	PROPOSED END STATE IF ACTION NOT MET (See Note 5)	CCDP (See Notes 1 & 2)	CLERP (See Note 1)
3.6.13	SBEACS	Two Trains Inoperable	No condition defined. Default 3.0.3 entry.	24 hrs If CC Available and Containment Intact Default to 3.6.1 otherwise	Mode 4 in 12 hrs (See Note 9)	NA (See Note 7)	NA (See Note 7)
3.7.11	CREACS	Two Trains Inoperable*	Explicit 3.0.3 entry	24 hrs Nuclear Hazard Only, otherwise [plant specific] hrs	Mode 4 in 12 hrs (See Note 9)	NA (See Note 7)	NA (See Note 7)
3.7.12	CREATCS	Two Trains Inoperable	Explicit 3.0.3 entry	24 hrs	Mode 4 in 12 hrs (See Note 9)	NA (See Note 7)	NA (See Note 7)
3.7.13	ECCS PREACS	Two Trains Inoperable*	No condition defined. Default 3.0.3 entry.	24 hrs	Mode 4 in 12 hrs (See Note 9)	NA (See Note 7)	NA (See Note 7)
3.7.15	PREACS	Two Trains Inoperable*	No condition defined. Default 3.0.3 entry.	24 hrs	Mode 4 in 12 hrs (See Note 9)	NA (See Note 7)	NA (See Note 7)

NA – Not applicable

Notes for Table 2-1:
1 Based on continued "at power" operation for full AOT (for ICCDPs and ICLERPs crediting the current one hour, See Tables 4.1-2 and 4.2-1a, respectively).
2 See Section 4.
3 Mode 5 end state not desirable as SDC is compromised. Mode 4 is low risk end state.
4 CSS proposed AOT applies to both containment cooling TSs.
5 Mode 3 - hot standby; Mode 4 - hot shutdown; Mode 5 - cold shutdown.
6 For plants with non-diverse containment cooling systems, unavailability of CSs is assumed to prevent the establishment of ECCS recirculation and result in core damage (See Table 4.2-1a).
7 AOT based on controlling system challenge probability to < 10^{-6} (See Section 4.4).
8 End state consistent with Reference 4.
9 Current 3.0.3 entry requires Mode 5 end state.
10 Assumes probability of manual RCS pressure control is high. If plant trip is considered likely a controlled shutdown should be initiated.

* Two trains inoperable for reasons other than inoperable boundary.

3.0 BACKGROUND

In response to the Nuclear Regulatory Commission (NRC's) initiative to improve plant safety by developing risk-informed TSs, the WOG has undertaken a program for defining and obtaining risk-informed TS modifications. As part of this program, several technical specification modifications involving Allowed Outage Time (AOTs) and specific ACTIONS were identified for joint application.

This report provides technical justification for the modification of various TSs to define and/or modify Action to extend the time required to initiate a plant shutdown from 1 hour (e.g. TS 3.0.3) to a risk-informed time varying from 4 hours to 72 hours, dependent upon the TS system/component and plant design features. In addition, the report proposes, consistent with Reference 4, the modification of many of the CT/AOT TS Actions to allow a Mode 4 end state when the time cannot be met.

The intent of the proposed modifications to the plant TS is to enhance overall plant safety by:

(a) Avoiding unnecessary plant shutdowns.

(b) Minimizing plant transitions and associated transition and realignment risks.

(c) Providing for increased flexibility in scheduling and performing maintenance and surveillance activities.

(d) Providing explicit guidance where none currently exists.

This report covers a diverse range of components with essentially four separate impacts on plant risk.

1) Accident Prevention

2) Accident Mitigation

3) Large Early Release Prevention

4) Control of Delayed Radiation Releases to the Environment

The first category of components contains those which are used during plant operation and whose removal from service may increase the plant risk by creating an increased potential for plant upsets. A typical TS component within this category is the pressurizer heaters. Under certain circumstances (e.g. inadequate emergency power) extended outage of these systems could complicate plant operations by increasing the complexity of plant pressure control. The incremental risk associated with the outage of these components is primarily associated with the increased potential for event initiation (i.e. plant trip).

The second category is comprised of components designed to support accident mitigation. These systems typically impact both the core damage and large early release probabilities. These systems/components are typically highly reliable, and normally available in a standby mode. Systems/components in this category are intended to function during rare, but high consequence, events. This category includes the components of the Emergency Core Cooling System (ECCS) and the pressurizer Power Operated Relief

Valves (PORVs)[1]. In some instances, functions of the containment cooling systems may also be grouped in this category.

The third category of components includes those that have a primary role in minimizing large early releases of radioactive materials. The only component included in this assessment is the containment.

The last category includes those components that impact the plant design basis and may affect offsite exposure following design basis and severe accidents, but have no direct impact on the surrogate risk metrics associated with core damage and large early releases. Typically these systems may contribute to controlling the magnitude of the releases or provide another design basis function. Components in this category include the control room, penetration rooms and Emergency Core Cooling System (ECCS) room ventilation systems, containment Iodine Cleanup Systems (ICS) and the containment sprays when used for fission product removal.

Risk assessments performed within the scope of this task are consistent with the general guidance of RGs 1.174 and 1.177. Where possible, risk-informed assessments of the proposed TS modifications are established based on bounding assumptions. In instances where plant-specific or generic plant-class risk assessments are performed, results are based on a current Probabilistic Safety Analysis (PSA) plant model. All WOG members with CE designed NSSSs consider the supporting analytical material contained within the document to be applicable to their respective member utilities, regardless of the format of their plant TSs.

[1] The design basis of the PORV is to provide protection against Pressurizer Safety Valve (PSV) challenges. This function has minimal impact on plant risk. A non-design basis function which may have a more significant impact on plant risk utilizes the PORV to support feed and bleed cooling to the core during total loss of feedwater events.

4.0 RISK-INFORMED EVALUATION OF ALLOWED OUTAGE TIMES

This section presents the methodology for a risk-informed assessment of AOTs when a design system or function is unavailable. The general methods used to support the risk-informed evaluations are based on RGs 1.174 and 1.177. In performing the evaluation, two conditions were tacitly assumed:

1) A condition resulting in the inoperability of a system or component which currently results in the need for an immediate shutdown is an infrequent event. This is evidenced by the fact that plant shutdowns due to entries into these TSs are rare. Furthermore, when this condition does arise, the actual cause of the inoperability is often due to an incomplete OPERABILITY "paper trail" or a partial system failures rather than a deleterious common-cause failure of critical components leading to a functional failure of the entire system.

and,

2) The risk incurred by increasing the required shutdown action time may be controlled to acceptable levels using a risk-informed approach that considers the component risk worth and offsetting benefits of avoiding plant transitions.

The extended time intervals sought to replace the one hour Action Statement are relatively short (generally, one day or less), non-repetitive and infrequently entered. Therefore, since a change to this aspect of the TS represents a temporary plant condition, it is considered to be in the nature of a pre-assessed Notice of Enforcement Discretion.

The criteria for the risk-informed assessment of the AOTs were selected based on RG 1.174. Regulatory Guide 1.174 indicates that for plant changes which would result in an increase in Core Damage Frequency (CDF) of less than 1.0E-6 per year and an increase in Large Early Release Frequency (LERF) of less than 1.0E-7 per year, the incremental change is considered small. Furthermore, the change may be considered regardless of the plants' total CDF. Since these proposed TS changes would be rare, (i.e. infrequent events due to emergent conditions) an effective surrogate single entry metric is appropriate. Assuming that plants enter one of the evaluated system unavailability conditions once every 5 years, the associated single entry CDP and Large Early Release Probability (LERP) consistent with the RG 1.174 guidance would be 5.0E-6 and 5.0E-7, respectively. Even more restrictive CDP/LERP guidelines were employed in this evaluation. These are:

- Incremental Conditional Core Damage Probability (ICCDP) < 1.0E-6
- Incremental Conditional Large Early Release Probability (ICLERP) < 1.0E-7

The above risk goals/guidelines were selected in preference to that of RG 1.177, since (1) RG 1.177 guidance is intended to apply to recurring maintenance entries and (2) the above guidelines ensure that the risks associated with implementing the proposed changes are small. As will be discussed later, for most of the extension requests defined in this document, the difference is academic for most systems as the requested AOT extension is consistent with either guideline. In a few instances (i.e., HPSI and PORV TS), the absolute maximum incremental risk exceeds the regulatory guidelines. The extended incremental AOT for these conditions is small and is recommended as a means of allowing a prudent "at power" assessment and minor repairs, so that shutdown risk may be averted.

Several systems contained within the TSs have no contribution, or a relatively indirect contribution, to either core damage or large early release. Such systems include those associated with the control room ventilation envelope, containment ventilation envelope, containment negative pressure protection and containment radionuclide control. While, in some instances, these systems may contribute to long-term public doses, their "risk " impact as assessed via Level 1 and 2 PSAs has consistently proven to be negligible. However, these systems do support the important design objective of helping to control the magnitude of radiological releases following an accident. The risk "worth" of these systems is established by ensuring that the allowed duration of system or component inoperability is limited and commensurate with its function. For the purpose of this assessment, recommended AOTs for these systems have been established, such that the probability of system challenge[2] during the AOT would be less than 1.0E-6. This is a conservative guideline as system challenge is not necessarily associated with core damage or significant radiation releases.

The following sub-sections provide a description of the methodology and the associated risk-informed assessments for the applicable TSs. An assessment of the specific recommended TS changes is provided in Section 5.

These TS modifications are intended to provide additional time for the plant staff to respond to conditions when a plant system or function within the scope of the TS is declared inoperable. As a consequence of the low expected frequency of the associated challenge, the short interval of the proposed AOT and the risk impact of the system unavailability, the redundancy and diversity typically associated with ensuring the deterministic aspect of defense-in-depth was not always possible. In these cases, defense-in-depth is considered via controlling the outage time for related equipment, restricting activities which may challenge these systems, and where possible, using contingency actions to limit concurrent unavailabilities and evaluating repair activities and alternatives. Such activities will be performed in accordance with 10 CFR 50.65(a)(4) and associated guidance documents.

4.1 ASSESSMENT OF CORE DAMAGE PROBABILITIES

This section describes the two methodologies used for calculating the core damage probability associated with extending the allowed pre-shutdown time interval from one hour to the proposed risk-informed AOTs. The first methodology focuses on the impact of removing accident mitigation components from service. The second methodology addresses those systems whose core damage contribution is due to initiation of accidents. The appropriate methodology to use in the core damage assessment is based on the function of the unavailable component. (Note that TS components that do not directly influence the initiation or mitigation of a core damage event are assumed to have an incremental Core Damage Probability (CDP) of zero.)

4.1.1 Methodology for Estimating Conditional CDP of the unavailability of Standby Mitigation Equipment

The present methodology provides a bounding generic approach for evaluating the incremental Conditional Core Damage Probability (CCDP) where possible. This approach can be implemented for evaluating the risks associated with the unavailability of standby mitigating systems. (A variant of this approach is applied to components whose unavailability impacts the plant trip probability, see Section 4.1.2.) Typical "at power" systems/components that can be grouped in the standby mitigating systems category include the Safety Injection Tanks (SITs), Low Pressure Safety Injection (LPSI), High Pressure

[2] System challenge implies a challenge where the operation of the system would mitigate the consequence of an event.

Safety Injection (HPSI) and Power Operated Relief Valves (PORVs). In this bounding risk approach, all events to which the mitigating system is a contributor are identified and the event frequency associated with the event is quantified. It is then assumed that any unavailability of the system will result in the inability of the event to be mitigated. Consequently, the events are conservatively assumed to go directly to core damage. Table 4.1-1 identifies the relationship of the mitigating systems to the initiating event frequencies against which they are designed to protect. Initiating frequencies are established from Reference 7. Detailed table notes provide additional information pertaining to the Initiating Event Frequency (IEF) assessment. In general, it is assumed that the unavailability of the affected system will lead to all associated events progressing towards core damage. Potential mitigating strategies not credited in this analysis and other associated conservatisms are summarized in response to request for additional information question 4 (Reference 21).

The general expression used for estimating the duration that a mitigating component/system may be removed from service (and be non-functional) is as follows:

$$ICCDP_{goal} = \sum_{i = events} [(CCDP_i) \times (IEF_i)] \times \left(\frac{\Delta T}{8760}\right) \qquad \text{(Eqn: 4-1)}$$

where:

$ICCDP_{goal}$ = 1.0E-6

$CCDP_i$ = Conditional core damage probability given event (i), with system unavailable, (assumed to be 1)

IEF_i = Initiating event frequency (per year) of event (i) occurring

ΔT = Time (in hours) to reach $ICCDP_{goal}$

The summation implies that all events where the component has a mitigation role in the success criteria are included.

The change in core damage frequency (ΔCDF) for each system/component is obtained by multiplying the respective ICCDP value with the yearly frequency that the system/component is expected to be declared inoperable. The general expression used for estimating ΔCDF is as follows:

$$\Delta CDF = (ICCDP) \times (f) \qquad \text{(Eqn: 4-2)}$$

where:

ΔCDF = Change in core damage frequency (per year)
ICCDP = Incremental core damage probability associated with the proposed extension
f = Frequency (per year) of system/component declared inoperable

4.1.1.1 Assessment of AOTs for the Unavailability of Mitigating Systems and Components

Using Equation 4-1, with IEF established in Table 4.1-1, one can relate the risk criteria with unavailable system hours. These results are compiled in Table 4.1-2.

	Table 4.1-1: Mapping of Mitigating Components and Frequency of Events Mitigated (a)								
System / Component Unavailable	Event Frequency (per year)								Component Challenge Frequency (g)
	LBLOCA	MBLOCA	SBLOCA	SGTR	Stuck Open PORV	Stuck Open PSV	Events Leading to F&B	ATWS	
SIT	5.0E-06	(b)	(b)	(b)	(b)	(b)	(b)	(b)	5.0E-06
LPSI	5.0E-06	4.0E-05	(d)	(d)	(b)	(b)	(b)	NA	4.5E-05
HPSI	5.0E-06	4.0E-05	5.0E-04	7.0E-05 (e)	1.0E-03	2.5E-03 (l)	1.0E-03 (c)	NA	(h)
CS (No CARCS available)	5.0E-06	4.0E-05	5.0E-04	(j)	(j)	(j)	(j)	(j)	5.5E-04
PORV	(b)	(b)	(b)	(b)	NA	(b)	1.0E-03 (c)	8.4E-06 (f)	1.0E-03 (f)
Pressurizer Heaters	NA	NA	NA	NA	NA	NA	NA	NA	NA
Boration System	NA	NA	NA	NA	NA	NA	NA	1.7E-05 (k)	1.7E-05 (k)

Notes for Table 4.1-1

(a) Data extracted from Table 3-1 and 3-8 of Reference 7.
(b) System/Component is not required to avert core damage for this event.
(c) The frequency of challenging F&B is estimated as the product of the frequency of events that lead directly or indirectly to a loss of Main Feedwater (MFW) and the probability of failing Auxiliary Feedwater (AFW). Events that lead directly or indirectly to a loss of MFW include a total loss of MFW flow, excessive or partial loss of MFW flow, and Loss of Offsite Power (LOOP). Based on information provided in Tables 3-1, 4-7 and 4-1 of Reference 2, the estimated frequency for the loss of MFW events defined herein is 5.03E-01 per year. The AFW failure probability is 2.0E-03, which is a bounding value for CE designed NSSSs. (See Table D-6 of Reference 19.) The estimated frequency of challenging F&B becomes 1.01E-03 per year.
(d) Components may be used as a backup mitigating component, however its risk importance is low in these sequences due to the high reliability of the primary component and the common dependencies.
(e) Not all SGTR events require HPSI for event mitigation. Following SGTR, cooldown procedures will allow event mitigation via two charging pumps. The probability that two charging pumps will be available for event mitigation is 0.99 (0.01 failure probability). Thus, the frequency of occurrence of an SGTR event requiring HPSI mitigation can be estimated as (SGTR initiating event frequency) multiplied by (charging pump failure probability) = (0.007 per year) x (0.01) = 7.0E-05 per year.
(f) This is taken as the product of the initiating event frequency based on the limited set of transients for ATWS and the failure probability of the RPS. The initiating event frequency is 1.4 per year. Using a generic RPS failure probability of 1.2E-5 per demand, the ATWS initiating event frequency becomes 1.68E-5 per year. This frequency is rounded up to 1.7E-5 per year. PORVs may be used to mitigate ATWS events and in a proceduralized manner to effect feed and bleed following a loss of FW events. Assume 50% of ATWS events require PORVs for event mitigation. ATWS events that occur in MOC/EOC do not require PORVs.
(g) Based on the total of applicable initiating event frequencies.
(h) 5.1E-03 per year for plants with PORVs; 3.1E-03 per year for plants without PORVs.
(i) NA – Not applicable.
(j) Containment heat removal is required to ensure sump cooling. Sump cooling is not required with these events as they may be mitigated using injection resources.
(k) The ATWS values from Table 3-8 of Reference 7 represent CDF due ATWS, rather than the initiating event frequency for ATWS. ATWS frequency is calculated as follows: $ATWS_f = I_T \times RPS = 1.4 \times 1.2E\text{-}5 = 1.68E\text{-}05$ per year (value rounded up to 1.7E-05 per year).
(l) Based on one event for the operating period considered in Reference 7.

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Table 4.1-2: Time (hrs) [(a)] for an Unavailable System to Accumulate to an Incremental CDP of 1.0E-6

System/Component Unavailable	Mean Challenge Frequency/(yr^{-1})	Time (hours) to reach CDP = 10^{-6} (b)	Proposed AOT (hours)	CDP Risk for Proposed AOT	ICCDP (a)
SIT	5.0E-06	1752	24	1.37E-08	1.31E-08
LPSI	4.5E-05	195	24	1.23E-07	1.18E-07
HPSI: PWR w/ PORVs	5.1E-03	2	4	2.33E-06	1.75E-06
HPSI: PWR w/o PORVs	3.1E-03	3	4	1.42E-06	1.06E-06
CS (no CARC available)	5.5E-04	16	12	7.53E-07	6.91E-07
PORV	1.0E-03	9	8	9.22E-07	8.07E-07
Boration Systems	1.7E-05	516	24	4.66E-08	4.46E-08

Notes for Table 4.1-2

(a) Based on incremental time (AOT - 1 hr)
(b) The time is rounded up to the nearest hour.

The above table suggests that the SITs, LPSI, and boration systems are clear candidates for having alternative Required Actions in the Technical Specification. Changes to the HPSI, CS and PORV TSs are also proposed. The proposed incremental AOT risk for HPSI is greater than the nominal goal of 1.0E-6. However, the infrequent entry into this condition (~ once in a plant operating life) supports these extensions as providing a low yearly risk increase of less than 5.0E-8, well within the guidelines of RG 1.174. The above changes will allow time for the operating staff to resolve the inoperabilities and hence avert the risk associated with a unit shutdown.

The inability of a PORV to open can impact the outcome of the total loss of FW events and to a lesser extent (assuming a 40 year residual operating life), Anticipated Transient without Scram (ATWS) events. From Table 4.1-1 the likelihood of an event requiring feed and bleed action is on the order of 1.0E-3 per year. The likelihood of ATWS events requiring PORVs for event mitigation is much lower (~ 8.4E-6). Thus, the risk of core damage resulting from total unavailability of the PORVs becomes 1.0E-3 per year.

This table also considers an AOT extension for the CSS when the CS is the only design basis heat removal system. Without availability of the CS, long term pressure and temperature control cannot be established. Furthermore, for CE designed NSSSs, sump cooling is accomplished via the use of heat exchangers in the spray line. The inability to inject subcooled water into the containment could result in a delayed failure of the ECCS system during its recirculation mode of operation and ultimately core damage. This condition was conservatively assumed to apply to all LOCAs.

The unavailability of the boration system affects post trip cooldown and ATWS mitigation. The insertion of the control rods will typically ensure reactor shutdown. The boration systems provide shutdown margin in the event of a stuck rod or failure of all CEAs to fully insert. Thus an inoperable boration system may interfere with being able to maintain the reactor shutdown and plant cooldown to cold shutdown. From an accident mitigative perspective, high pressure boration pathways impact ATWS events. In this assessment, the relationship is conservatively treated by assuming that the incremental core damage risk is the same as the ATWS initiating event frequency. This significantly over estimates the risk, since a portion of the ATWS events will proceed to core damage regardless of the availability of this system.

The change in core damage frequency is estimated using Equation 4-2. It is assumed that the inoperability for the above system/components would be an infrequent event that ranges between once every three years to once every five years for any one rand system. This assumption is reasonable given the occurrence of these typical events. These frequencies are not intended to be a prohibition on the use of the proposed actions, but rather are cognitive of the infrequent nature of such failures. Using this assumption and the ICCDP values from Table 4.1-2, the estimated ΔCDF for each system/component is shown in Table 4.1-3.

Table 4.1-3: Potential Risk Impact of Proposed AOT on Change in Core Damage Frequencies

System/Component Unavailable	ICCDP	ΔCDF (per year)	
		1-In-3 yr. Entry	1-In-5 yr. Entry
SIT	1.31E-08	4.38E-09	2.63E-09
LPSI	1.18E-07	3.94E-08	2.36E-08
HPSI: PWR w/ PORVs	1.75E-06	5.82E-07	3.49E-07
HPSI: PWR w/o PORVs	1.06E-06	3.54E-07	2.12E-07
CS (no CARC available)	6.91E-07	2.30E-07	1.38E-07
PORV	8.07E-07	2.66E-07	1.60E-07
Boration Systems	4.46E-08	1.49E-08	8.93E-09

4.1.2 CDP estimates the unavailability of plant control equipment: Assessment of Risk Contribution of the unavailability of Class 1E Pressurizer Heaters

The pressurizer Technical Specification (3.4.9) includes requirements for two banks to have a minimum pressurizer heater power and emergency power supply capability. It is the primary intent of the inclusion of pressurizer heater requirements within the TS to ensure that long term subcooling will be maintained during a loss of offsite power event. Pressurizer heaters are not considered in design basis accident analyses and are not required to effect a post-accident plant cooldown (however, the cooldown will be less controlled.)

Consequently pressurizer heaters do not have a significant role in the mitigation of core damage events. However, these heaters are necessary to adequately control the RCS pressure during normal power operation. In this assessment, it is assumed that the unavailability of the pressurizer heaters will increase the potential for plant trip. The risk associated with this component unavailability was evaluated by assuming that without pressurizer heaters, the RCS pressure will be controlled manually by other means (i.e. charging and letdown, HPSI or RCS Heat Removal). The current methodology assumes that the incremental risk of the unavailability of these systems is approximately:

$$ICCDP \cong \Delta IE \times CDP|_{trip} \times \frac{AOT}{8760}$$

Where ΔIE is the increase in reactor trip frequency due to the unavailability of the pressurizer heaters, $CDP|_{trip}$ is the core damage probability for an associated trip, and AOT is the outage time for the heaters.

In this case, the unavailability of the Class 1E pressurizer heaters is assumed to increase the plant trip potential by 0.05 per day (a typical plant trip probability is normally about 1.5 per year or 0.004 per day).

This is considered a conservative estimate in that many potential TS entries may not involve normal pressurizer heater capability (e.g. some entries may be influenced by the status of the emergency power supply) and situations which result in increased difficulty in maintaining and controlling pressure would directly result in plant shutdown. Given the availability of AFW and Emergency Diesel Generators, the conditional core damage probability following a normal plant high/low pressure trip is ≈ 6.0E-6 for a representative CE designed PWR (Reference 18). Substituting a value of 5.0E-2 per day (18.3 per year) for the assumed increase in plant trip potential and a value of 6.0E-6 for CDP/trip in the above expression, the probability of the loss of all pressurizer heaters causing a core damage event is approximately 3.0E-7 over a 24 hour period. Based on the incremental time of 23 hours (i.e., AOT-1), the ICCDP value becomes 2.9E-07. The associated changes in core damage frequency for losing all pressurizer heaters once-in-5 years and once-in-3 years are 5.8E-08 per year and 9.6E-08 per year, respectively. Therefore, as RCS pressure can be controlled manually, the risk of extending the AOT to 24 hours is acceptably small. Such a condition might be expected if non-Class 1E heaters are operational. If plant pressure cannot be manually controlled, an orderly plant shutdown should be initiated.

4.1.3 Comment on Uncertainty in CDPs

The preceding assessments utilized mean values of IEFs with a conservative assumption that system challenges proceeded to core damage. That is, operator recovery and/or actions and the availability of alternative mitigative systems are not credited. Overall, using the upper bound 95th percentile value for IEFs, as shown in Table 4.1-4, would increase the risk values presented in Table 4.1-2 by a factor of approximately four or less.

Table 4.1-4 Initiating Event 95th % Upper Bound Frequencies

Initiating Event	Mean IEF (per yr)	95th % Upper Bound
Large LOCA	5.0E-06	1.0E-5
Medium LOCA	4.0E-05	1.0E-4
Small LOCA	5.0E-04	1.0E-3
Steam Generator Tube Rupture	7.0E-03	1.4E-2
Anticipated Transient w/o Scram	1.7E-05	2.5E-5
Stuck Open PORV	1.0E-3	3.9E-3
Stuck Open PSV	2.5E-3	1.1E-2

A review of the above table indicates that the error factors for more risk significant initiating events are on the order of 2 to 4. The impact of these uncertainties on the plant risk, (see Table 4.2-4), demonstrates that even at the upper bound IEF, the proposed AOT does not introduce a significant increase in plant risk for the AOTs. This conclusion is further supported by the fact that system inoperability entries are infrequent events and that capabilities to restore operability while "at power" will avert the risk of plant shutdown [(which is generally equivalent to the risk associated with AOT entry (see Section 4.5)].

4.2 ASSESSMENT OF INCREMENTAL LARGE EARLY RELEASE PROBABILITY RESULTING FROM AN INCREMENTAL INCREASE IN CORE DAMAGE

This section considers the impact of the recommended TS modifications in terms of their effect on the Incremental Conditional Large Early Release Probability (ICLERP). The Large Early Release Frequency (LERF) is defined as the frequency of those accidents leading to significant, unmitigated release of radioactivity from containment in a time frame prior to effective evacuation of the close-in population, such that there is a potential for early health effects. This includes events which lead to early containment failure at or shortly after vessel breach, containment bypass events and loss of containment isolation. A review of CE designed NSSSs indicates that early releases arise as a result of one of the following classes of scenarios:

1. Containment Bypass Events

   These events include interfacing system Loss of Coolant Accidents (LOCAs) and Steam Generator Tube Ruptures (SGTRs) with a simultaneous loss of Steam Generator (SG) isolation [e.g. stuck open Main Steam Safety Valves (MSSVs or ADVs)].

2. Severe Accidents Accompanied by Loss of Containment Isolation

   These events include any severe accident in conjunction with an initially unisolated containment.

3. Containment Failure Associated with Energetic Events in the Containment

   Events causing containment failure include those associated with the High-Pressure Melt Ejection (HPME) phenomena (including Direct Containment Heating (DCH)) and hydrogen conflagrations/detonations.

Of the three release categories, Category 1 tends to represent a large, early release of direct, unscrubbed fission products to the environment. Category 2 events encompass a range of releases varying from early to late. These releases may, or may not, be scrubbed. Category 3 events may result in a high-pressure failure of the containment immediately upon, or a short time after, reactor vessel failure.

Level 2 analyses for CE designed NSSS plants indicate that post-accident operation of one containment fan cooler or one containment spray train is sufficient to ensure containment integrity (Reference 8). Thus, the design of the typical CE PWR has diverse and redundant components for use in post-accident containment cooling.

The calculation of the ICLERP due to the limited duration unavailability of safety equipment may be estimated by relating the role of the unavailable component with reference to its role in mitigating one or more of the three categories of contributors to the large early release.

4.2.1 Discussion of the ICLERP Model

Incremental Conditional Large Early Release Probability (ICLERP) is a measure of the incremental risk of significant radiation exposure associated with the specific system out of service for a period of time. The ICLERP estimate consists of three parts: (1) challenge frequency (or core damage frequency), (2) conditional probability of Large Early Release (LER) and (3) the exposure time.

The contribution of incremental core damage frequency is established from Section 4.1. Bounding estimates for ICLERP were developed by using a simplified LER event tree presented in Figure 4.2-1. The LER event tree sums the incremental contributions from (a) containment bypass events (including Inter-System LOCAs and induced SGTRs), (b) loss of containment isolation events, and (c) energetic containment failures.

LERF assessments are provided for at-power operation only. The simplified LER event tree (See Figure 4.2-1) focuses on causes for, and interrelationships of, the containment large early release contributors following an event which is adversely impacted by the unavailability of an accident mitigation system. As discussed previously, the input into the LER event tree is the ICCDP. The fraction of ICCDP that propagates into a large early release event is established based on responses to the following events:

- Containment isolation
- High RCS pressure
- Secondary side depressurization of the steam generator(s).
- Occurrence of thermally-induced SGTR.
- Containment failure due to RPV lower head failure.

In evaluating the LERF increases, it was conservatively assumed that all incremental core damage events lead to high pressure Reactor Coolant System (RCS) core damage states. It was also assumed that no operator actions were performed to depressurize the RCS prior to failure of the reactor vessel lower head. The top events in the LER tree are described and modeled as follows:

Containment Isolated (ICI)

This top event defines the state of containment integrity prior to the event. Large early fission product releases could occur when a severe accident occurs in conjunction with an initially unisolated containment. Typically, these events are very small contributors to the total containment failure probability. The probability of containment isolation failure used in the PSAs for the CE designed NSSS plants varies from 1.0E-4 to approximately 3.0E-3. The upper limit of 3.0E-3 was selected as a bounding value.

RCS Pressure – High (RCSH)

In this assessment, incremental core damage events leading to high RCS pressure are associated with the inability to establish Feed & Bleed cooling to the RCS. This affects a fraction of the Loss of Feedwater (LOFW) and related initiating events and all ATWS events. Events where the mitigating equipment is only used to respond to a LOCA will not have any incremental high pressure sequences, since LOCA events are low and moderate pressure events and ECCS equipment cannot discharge into the high pressure RCS. In this assessment, all core damage events associated with inoperability of PORVs or the unavailability of the boron system are assumed to result in a high pressure core damage sequence (RCSH = 1). Analogously, contributions to the LOCA CDP increment LOCAs are not assumed to result in high RCS measures (RCSH = 0).

Steam Generator Depressurized (SGD)

It is conservatively assumed that incremental core damage events that do not arise as a result of a LOCA lead to a core melt condition at high RCS pressure. Therefore, the potential for these events becoming a large early release is dependent upon the ability to maintain the steam generator tubes intact and the secondary side isolated. Both of these factors are reflected in the response to this query. Steam generator depressurization is assumed to occur either via prior operator action or failure of a Main Steam Safety Valve (MSSV) to close. The combined probability of Steam Generator (SG) depressurization has been estimated for a typical CE designed PWR (see Reference 5) to be less than 0.1. Therefore this parameter is set equal to 0.10.

Thermally-Induced SGTR Occurs (TI-SGTR)

Given a steam generator depressurization event, it is conservatively assumed that the probability that a steam generator tube will fail prior to failure of another RCS component is 0.5. (This factor is a conservative representation of the failure probability and will be dependent on the SG design, age, operating history, and time in cycle.) The assessment is bounding provided SG tubes meet their design limits. Studies conducted by many researchers (see for example Reference 20) indicate that the probability of steam generator tube failure reduces significantly if the SGs remain pressurized. For this condition, the probability of thermally-induced SGTR is conservatively assumed to be 0.01.

Additional conservatism taken in the thermally-induced SGTR assessment includes neglecting the potential for the challenged PSV/PORV to stick open and the neglect of any operator actions to depressurize the RCS. Both of these factors can result in a significant reduction to the LERP. For example, NRC assessments of PSV/PORV challenges during station blackout scenarios indicate a large number (~35 water/two phase) challenges of the PSVs prior to core uncovery. Such challenges have a high (~14%) probability of failing the PSV, resulting in a potentially open valve (Reference 5).

RPV Lower Head Failure Results in Containment Failure (DCH)

Failure of the Reactor Pressure Vessel (RPV) lower head releases an energetic discharge of molten core materials into the containment. A recent assessment of Direct Containment Heating (DCH) induced containment threats performed by Sandia National Laboratories (Reference 6) concluded that the Conditional Containment Failure Probability (CCFP) is less than 0.01 for Ft. Calhoun Station (FCS), Palo Verde Nuclear Generating Station (PVNGS) 1, 2 & 3, St. Lucie (SL) 1 & 2 and Waterford Steam Electric Station (WSES) 3. These calculations were based on an assessment of DCH induced pressure loading and the plant specific fragility curves. Arkansas Nuclear One, Unit 2 (ANO-2), Millstone Point, Unit 2 (MP2), Palisades and San Onofre Nuclear Generating Station (SONGS) Units 2 & 3 were assessed to have CCFPs between 0.01 and 0.1. One plant failed the screening criterion established by the Reference 6 methodology. This plant required additional analyses to resolve the DCH issue. After considering the High Pressure Melt Ejection (HPME) probabilities given core damage for these plants, the Sandia assessment concluded that the CCFPs for all CE designed NSSSs would be approximately 0.01 or less, when considering thermally induced failure of RCS piping in advance of reactor vessel lower head failure. Therefore, a CCFP of 0.01 due to HPME is selected and used as a bounding value for the combined effects of RCS piping failure and HPME induced containment failure for all CE designed NSSS plants.

Low pressure vessel failures and early hydrogen deflagration induced containment failures have been neglected in this assessment as their conditional LERF impact is not significant for events where the inoperability results in increased high pressure CD sequences and is < 1% for low pressure sequences.

4.2.2 Supporting ICLERP Assumptions for ICLERP Quantification

Based on the above discussions, the following assumptions are made with respect to the ICLERP model:

1. The probability of containment isolation failure used in the PSAs for CE designed NSSS plants varies from 1.0E-4 to approximately 3.0E-3. The upper limit (3.0E-3) was selected and used as a bounding value in this report.

2. It is assumed that all the incremental core damage events arising from PORV or Boration system unavailabilities result in a high RCS pressure plant damage state (RCS_HIGH = 1). Therefore, the potential for these events becoming a large early release is dependent upon the ability of the RCS to maintain the steam generator tubes intact and for the secondary side to be isolated.

3. Incremental core damage events resulting from LPSI or SIT unavailability results only in the RCS pressure events (RCS_HIGH = 0).

4. The High Pressure Safety Injection (HPSI) system is primarily used to mitigate moderate and low pressure events. It is conservatively assumed that for plants with PORVs, 20% of the incremental plant damage state categorized with HPSI system unavailability will be at high RCS pressure.

5. It is assumed that 50% of the incremental core damage events resulting from a reactor trip induced by the unavailability of the pressurizer heaters leads to high pressure plant damage.

6. When exposed to high-pressure core damage states, the probability of a steam generator tube failing prior to failure of the RCS is conservatively assumed to be indeterminate (0.5). It is also assumed that all TI-SGTRs are classified as large early releases.

7. A Conditional Containment Failure Probability (CCFP) of 0.01 due to High Pressure Melt Ejection (HPME) is selected and used as a bounding value for the combined effects of RCS piping failure and HPME induced containment failure for all CE designed NSSS plants. This is based on a recent assessment performed by Sandia National Laboratories (Reference 6).

8. With the exception of a potential TI-SGTR event, it is assumed that no new bypass events are created.

4.2.3 ICLERP Quantification

Estimates for ICLERPs were developed based on the conservative approach described above. This approach sums the incremental LER contributors identified in the simplified LER event tree shown in Figure 4.2-1 (System/Component specific trees are included in Appendix B). Accordingly, the ICLERP is estimated by multiplying the incremental contributors to large early release with the associated ICCDP for the proposed AOT. The incremental contributors to a large early release are identified in Figure 4.2-1 as event tree scenarios LERP-1 through LERP-5. A summary description for each of these scenarios is:

LERP- 1: This incremental contributor to large early release involves incremental core damage probability followed by an isolated containment, a depressurized steam generator due to stuck open MSSV and TI-SGTR.

LERP- 2: This incremental contributor to large early release involves incremental core damage probability followed by an isolated containment, a depressurized steam generator due to a stuck open MSSV, steam generator tubes intact and HPME failure of the containment.

LERP- 3: This incremental contributor to large early release involves incremental core damage probability followed by an isolated containment, pressurized steam generators and TI-SGTR.

LERP- 4: This incremental contributor to large early release involves incremental core damage probability followed by an isolated containment, pressurized steam generators with tubes intact and HPME failure of the containment.

LERP- 5: This incremental contributor to large early release involves incremental core damage probability followed by failure to isolate the containment.

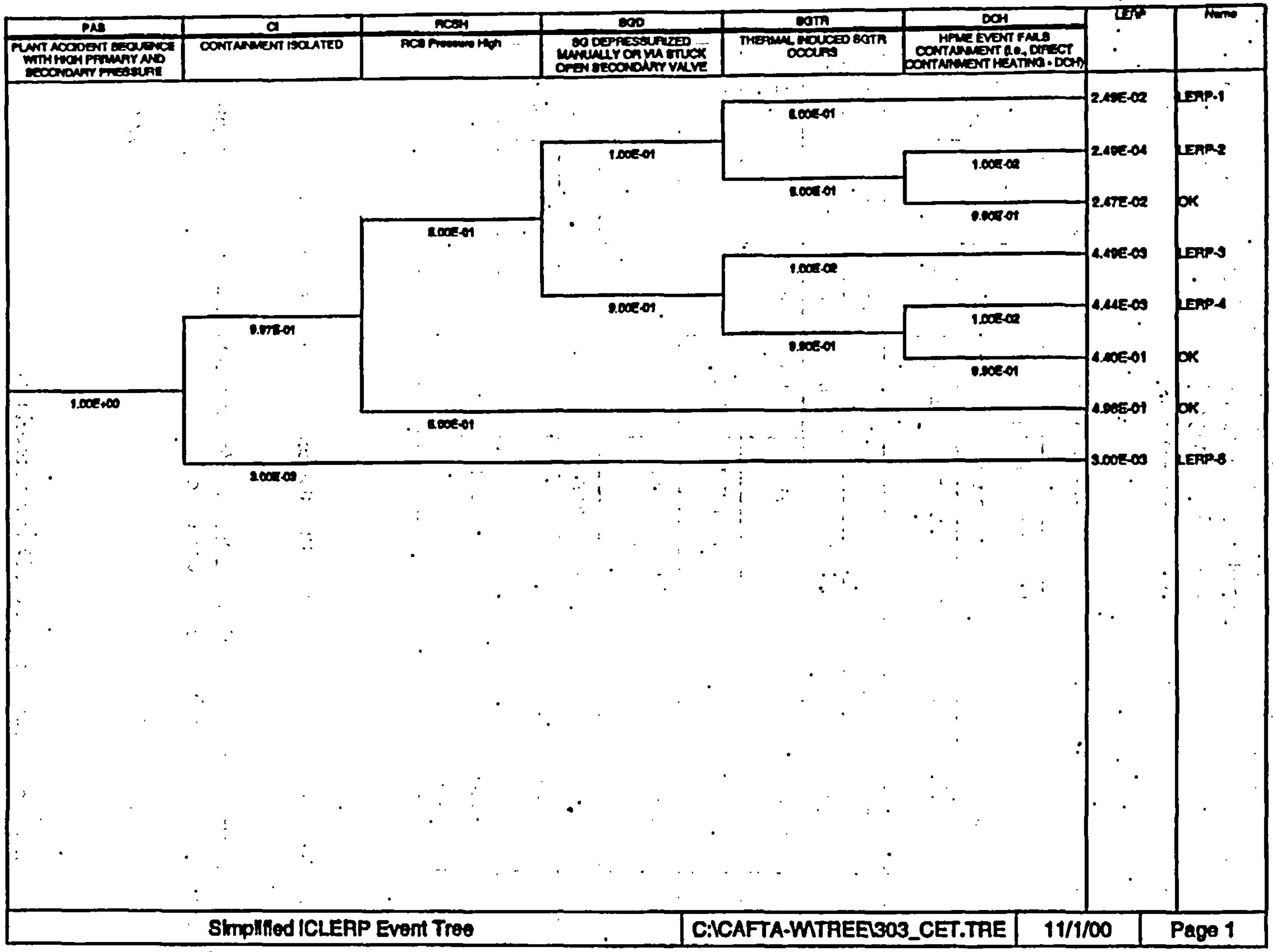

Figure 4.2-1: Simplified Incremental Large Early Release Event Tree

The simplified LER event tree (Figure 4.2-1) was quantified for each of the systems for a normalized ICCDP. Refer to Appendix B for the values used in the quantification of each system. The results of the quantification are presented in Table 4.2-1a. The conditional probability for each of the LERP scenarios is provided along with the sum of the LERP contributions for each system. The total LERP was multiplied by the CCDP taken from Table 4.1-2 for the proposed AOT to arrive at the CLERP for the proposed AOT change.

Table 4.2-1a: CLERP Estimates Due to the Unavailability of Selected Components

System / Component	Proposed AOT (hours)	CDP per Proposed AOT (from Table 4.1-2)	LERP 1 through 5 (from Figure 4.2-1) (Note 2)					Total LERP	Total CLERP per AOT
			LERP-1	LERP-2	LERP-3	LERP-4	LERP-5		
SIT	24	1.37E-8	0	0	0	0	3.0E-3	3.0E-3	4.1E-11
LPSI	24	1.23E-7	0	0	0	0	3.0E-3	3.0E-3	3.7E-10
HPSI (plants w/PORV)	4	2.33E-6	1.0E-2	1.1E-4	1.8E-3	2.0E-3	3.0E-3	1.7E-2	4.0E-8
HPSI (plants w/o PORV)	4	1.42E-6	1.0-E2	1.1E-4	1.8E-3	2.0E-3	3.0-E3	1.7E-2	2.4E-8
CS (Note 3)	12	7.53E-7	1.0E-2	1.1E-4	1.8E-3	2.0E-3	3.0E-3	1.7E-2	1.3E-8
PORV	24	9.22E-7	5.0E-2	5.5E-4	9.0E-3	9.8E-3	3.0E-3	7.2E-2	6.7E-8
Boration Systems	24	4.66E-8	5.0E-2	5.5E-4	9.0E-3	9.8E-3	3.0E-3	7.2E-2	3.4E-9
Pressurizer Heaters	24	3.00E-7 (Note 1)	2.5E-2	2.7E-4	4.5E-3	4.9E-3	3.0E-3	3.8E-2	1.1E-8

Notes for Table 4.2-1a

(1) See Section 4.1.2
(2) CLERP is defined as the conditional probability that a LER will occur following a core damage event.
(3) CARCS unavailable

The ICLERP associated with the proposed AOT for each system/component declared inoperable (and non functional) can be estimated using the following expression.

$$ICLERP = ICCDP \times LERP \qquad \text{(Eqn: 4-3)}$$

where:

ICCDP = Incremental Conditional Core Damage Probability
LERP = Large Early Release Probability

The change in LERF (i.e., ΔLERF) for each system/component can be obtained by multiplying the ICLERP value by the yearly frequency that the system/component is expected to be inoperable (and non functional). The change can be expressed as follows:

$\Delta LERF = ICLERP \times f$ (Eqn: 4-4)

where:

ΔLERF	=	Change in large early release frequency (per year)
ICLERP	=	Incremental change in large early release probability
f	=	Frequency (per year) of system/component declared inoperable

Using Equations 4-3 and 4-4, the risk measures associated with ICLERP and ΔLERF are summarized in Table 4.2-1b for each system/component. Similar to ΔCDF, the yearly frequency an inoperable system/component (and nonfunctional) is assumed to be infrequent (e.g. ranges between 1-in-3 years to 1-in-5 years).

Table 4.2-1b Large Early Release Risk Impact

System/Component Unavailable	ICCDP	LERP	ICLERP	ΔLERF (per year)	
				1-in-3 yr. Entry	1-in-5 yr. Entry
SIT	1.31E-08	3.00E-03	3.94E-11	1.31E-11	7.88E-12
LPSI	1.18E-07	3.00E-03	3.54E-10	1.18E-10	7.09E-11
HPSI: PWR w/ PORVs	1.75E-06	1.68E-02	2.94E-08	9.81E-09	5.89E-09
HPSI: PWR w/o PORVs	1.06E-06	1.68E-02	1.79E-08	5.96E-09	3.58E-09
CS (no CARC available)	6.91E-07	1.68E-02	1.16E-08	3.88E-09	2.33E-09
PORV	8.07E-07	7.22E-02	5.83E-08	1.94E-08	1.17E-08
Boration Systems	4.46E-08	7.22E-02	3.22E-09	1.07E-09	6.45E-10
Pressurizer Heaters	2.88E-07	3.76E-02	1.08E-08	3.61E-09	2.16E-09

4.2.4 Incremental Conditional LERP Sensitivity Studies

This section presents the results of four sensitivity studies. Three of the four cases involve key parameters in the assessment of the Large Early Release Probability. These parameters are: (a) the probability that a TI-SGTR will occur in advance of another RCS structural failure, (b) bounding assessment of TI-SGTR, and (c) the probability that the MSSV will fail open, depressurizing one steam generator. These parameters were selected for the sensitivity study since the TI-SGTR is a dominant LERP contributor. The fourth sensitivity case involves the risk impact associated with utilizing bounding frequencies for event initiators.

(a) Thermally-Induced SGTR occurs in Advance of Another RCS Structural Failure (TI-SGTR)

A thermally-induced SGTR depends on the steam generator design, age, operating history and the time in cycle. Each factor or combination of factors may influence the likelihood of large early releases. In this evaluation, a conservative probability of 0.5 was assumed for failure of a steam generator tube prior to the failure of another RCS structural component (e.g. hot leg or surge line). The 50% SGTR failure probability was based on a severely degraded steam generator. This value also reflects analytical uncertainties which result in inconsistent predictions of this phenomenon. To address this uncertainty, a sensitivity evaluation was performed to determine the impact of variations in TI-SGTR on the large early release probability. This sensitivity involved varying the probability of TI-SGTR from 0.4 and 0.6 and then requantifying the simplified LER event tree to estimate the normalized LERPs for each system.

Variations in the probability for TI-SGTR affect the probabilities of large early scenarios LERP-1 and LERP-2 (See Figure 4.2-1) for all CE designed NSSS plants. All of the other probabilities for the remaining large early scenarios are unaffected. The results of this sensitivity evaluation are summarized in Table 4.2-2a. This scenario results in an inadvertent plant trip which has a small probability of leading to a core damage condition. The resulting plant damage state is assumed to be high pressure 50% of the time.

Table 4.2-2a: Sensitivity Results for Incremental Conditional Large Early Release Probability: TI-SGTR Probability

INOPERABLE COMPONENT	TI-SGTR Probability	LERP-1	LERP-2	LERP-3	LERP-4	LERP-5	Total LERP
Pressurizer Heaters	0.6(1)	2.99E-02	2.19E-04	4.49E-03	4.94E-03	3.00E-03	4.26E-02
	0.5	**2.49E-2**	**2.74E-4**	**4.49E-3**	**4.94E-3**	**3.00E-3**	**3.76E-2**
	0.4(1)	1.99E-02	3.29E-04	4.49E-03	4.94E-03	3.00E-03	3.27E-02

Note for Table 4.2-2a:

1. A bounding value of 0.01 is used in the calculations for Conditional Containment Failure Probability (CCFP) due to HPME.

Using the thermally-induced SGTR probability of 0.5 as the base case, the results in Table 4.2-1a indicate that the normalized CLERP increases approximately linearly as the thermally-induced SGTR probability increases.

(b) Bounding Assessment of Thermally-Induced SGTR

A bounding case was also performed to assess the impact of LERP contributors. For this case, it was assumed that a thermally-induced SGTR occurred. It was also assumed that containment isolation was much less likely to occur, and a containment isolation failure probability of 0.01 was used. The probability of a dry (depressurized) steam generator is assumed to be as high as 0.3. This value is in the same range as the 0.27 value that was used in NUREG-1150. The results of this bounding evaluation are presented in Table 4.2-2b.

Table 4.2-2b: Bounding Estimates Given TI-SGTR

System/ Component Unavailability	LERP-1	LERP-2	LERP-3	LERP-4	LERP-5	Total LERP
SIT	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.00E-02	1.00E-02
LPSI	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.00E-02	1.00E-02
HPSI (Plants W/PORVs)	5.94E-02	0.00E+00	1.39E-03	1.52E-03	1.00E-02	7.23E-02
HPSI Plants (w/o PORVs)	5.94E-02	0.00E+00	1.39E-03	1.52E-03	1.00E-02	7.23E-02
CS	5.94E-02	0.00E+00	1.39E-03	1.52E-03	1.00E-02	7.23E-02
PORV	2.97E-01	0.00E+00	6.93E-03	7.62E-03	1.00E-02	3.22E-01
Boration System	2.97E-01	0.00E+00	6.93E-03	7.62E-03	1.00E-02	3.22E-01
Pressurizer Heaters	1.49E-01	0.00E+00	3.47E-03	3.81E-03	1.00E-02	1.66E-01

Table 4.2-2b shows that the bounding value for total LERP is 3.22E-1. This value is attributed to the PORVs being unavailable. Combining the ICCDP from Table 4.1-2 and the total LERP from Table 4.2-2b for the PORVs being unavailable produces a bounding ICLERP of 2.60E-7. If the PORVs are declared inoperable once every three years or once every five years, the corresponding ΔLERFs are 8.65E-08 per year and 5.20E-08 per year, respectively. The risk impact for bounding estimates for each system/component given a TI-SGTR is summarized in Table 4.2-2c.

Table 4.2-2c Risk Impact for Bounding Estimates Given TI-SGTR

System/Component Unavailable	ICCDP	ΔCDF (per year)		LERP	ICLERP	ΔLERF (per year)	
		1-In-3 yr. Entry	1-In-5 yr. Entry			1-In-3 yr. Entry	1-In-5 yr. Entry
SIT	1.31E-08	4.38E-09	2.63E-09	1.00E-02	1.31E-10	4.38E-11	2.63E-11
LPSI	1.18E-07	3.94E-08	2.36E-08	1.00E-02	1.18E-09	3.94E-10	2.36E-10
HPSI: PWR w/ PORVs	1.75E-06	5.82E-07	3.49E-07	7.23E-02	1.26E-07	4.21E-08	2.53E-08
HPSI: PWR w/o PORVs	1.06E-06	3.54E-07	2.12E-07	7.23E-02	7.68E-08	2.56E-08	1.54E-08
CS (no CARC available)	6.91E-07	2.30E-07	1.38E-07	7.23E-02	4.99E-08	1.66E-08	9.99E-09
PORV	8.07E-07	2.69E-07	1.60E-07	3.22E-01	2.60E-07	8.65E-08	5.19E-08
Boration Systems	4.46E-08	1.49E-08	8.93E-09	3.22E-01	1.44E-08	4.78E-09	2.87E-09
Pressurizer Heaters	2.88E-07	9.58E-08	5.75E-08	1.66E-01	4.77E-08	1.59E-08	9.53E-09

(c) <u>Steam Generator Depressurized (SGD)</u>

The potential for core damage events at high RCS pressure becoming a large early release is dependent upon the ability to maintain the steam generator tubes intact and the secondary side isolated. In this evaluation, a probability of 0.1 was conservatively assumed to bound the probability of one or more MSSVs failing to close. A sensitivity evaluation was also performed on this parameter to determine the impact on the large early release due to the changes in the probability of a MSSV to close. This study involved varying the probability of a MSSV failing open from 0.05 to 0.2 and then requantifying the simplified LER event tree for a representative event and estimating the normalized LERP. Variations of the probability for a MSSV failing open affect the probabilities of large early scenarios LERP-1 through LERP-4 (See Figure 4.2-1). The probability of large early release scenario LERP-5 (containment isolation) is not affected. The results of this sensitivity evaluation are summarized in Table 4.2-3.

Table 4.2-3: Sensitivity Results for a MSSV Failing Open: Core Damage Event Resulting from a Plant Trip Following the Unavailability of Pressurizer Heaters

MSSV Failure Probability (SGD)	LERP-1	LERP-2	LERP-3	LERP-4	LERP-5	Total LERP
0.050	1.25E-02	1.37E-04	4.74E-03	5.21E-03	3.00E-03	2.55E-02
0.075	1.87E-02	2.06E-04	4.61E-03	5.07E-03	3.00E-03	3.16E-02
0.100	**2.49E-02**	**2.74E-04**	**4.49E-03**	**4.94E-03**	**3.00E-03**	**3.76E-02**
0.125	3.12E-02	3.43E-04	4.36E-03	4.80E-03	3.00E-03	4.37E-02
0.150	3.74E-02	4.11E-04	4.24E-03	4.66E-03	3.00E-03	4.97E-02
0.175	4.36E-02	4.80E-04	4.11E-03	4.52E-03	3.00E-03	5.57E-02
0.200	4.99E-02	5.48E-04	3.99E-03	4.39E-03	3.00E-03	6.18E-02

Notes for Table 4.2-3
1. A bounding value of 0.01 is used in the calculations for CCFP due to HPME.

Using the MSSV failure probability of 0.1 as the base case, the results in Table 4.2-3 indicate that the normalized LERP increases as the MSSV failure probability increases. While ICLERP is sensitive to variations in SGD, the nominal value selected for the assessment provides a conservative basis for the assignment of risks associated with these TS changes and the impact is relatively linear.

(d) Risk Impact Associated with Bounding (95^{th} %) Initiating Event Frequencies

The initiating event frequencies contributing to the overall challenge frequency or CDF for each system/component were statistically combined. Each initiating event frequency was assumed to be log-normally distributed. The 95^{th} % upper bound challenge frequency obtained for each system/component is provided in Table 4.2-4. The corresponding ICCDP values for the proposed AOT are also provided in this table. The risk impact as measured in terms of CDF and LERF is summarized in Table 4.2-5. The yearly frequency of an inoperable system/component (and non functional) is assumed to range between once-in-3 years to once-in-5 years per plant. For purposes of assessment, this frequency range was applied to all systems evaluated in the report.

Table 4.2-4: System/Component 95^{th} % Upper Bound ICCDPs

System/Component Unavailable	Proposed AOT (hours)	95^{th} Upper Bound Frequency (per year)	ICCDP
SIT	24	1.00E-05	2.63E-08
LPSI	24	1.10E-04	2.89E-07
HPSI: PWR w/ PORVs	4	1.40E-02	4.79E-06
HPSI: PWR w/o PORVs	4	9.60E-03	3.29E-06
CS (no CARC available)	12	1.10E-03	1.38E-06
PORV	8	2.50E-03	2.00E-06
Boration Systems	24	2.50E-05	6.56E-08

Table 4.2-5: Summary of the Risk Impact for 95th % Bounding Frequencies

System/Component Unavailable	ICCDP	ΔCDF (per year)		LERP	ICLERP	ΔLERF (per year)	
		1-In-3 yr. Entry	1-In-5 yr. Entry			1-In-3 yr. Entry	1-In-5 yr. Entry
SIT	2.63E-08	8.75E-09	5.25E-09	3.00E-03	7.88E-11	2.63E-11	1.58E-11
LPSI	2.89E-07	9.63E-08	5.78E-08	3.00E-03	8.66E-10	2.89E-10	1.73E-10
HPSI: PWR w/ PORVs	4.79E-06	1.60E-06	9.59E-07	1.68E-02	8.08E-08	2.69E-08	1.62E-08
HPSI: PWR w/o PORVs	3.29E-06	1.10E-06	6.58E-07	1.68E-02	5.54E-08	1.85E-08	1.11E-08
CS (no CARC available)	1.38E-06	4.60E-07	2.76E-07	1.68E-02	2.33E-08	7.76E-09	4.65E-09
PORV	2.00E-06	6.66E-07	4.00E-07	7.22E-02	1.44E-07	4.81E-08	2.89E-08
Boration Systems	6.56E-08	2.19E-08	1.31E-08	7.22E-02	4.74E-09	1.58E-09	9.48E-10

Final Comments

It should be noted that the ICLERP values presented in Table 4.2-1b are bounded by the ICCDP associated with each event. Using an ICLERP goal of 1.0E-7 (Reference 1), the ICLERP goal is satisfied for the proposed AOT extensions. The unavailability of HPSI will impact primarily low pressure states and result in an impact in LERP that is dominated by Intersystem Loss of Coolant Accidents and low pressure vessel failures and early hydrogen deflagration (not considered). The impact of these events is considered small and would result in a combined LERP of < 0.01.

4.3 ASSESSMENT OF INCREMENTAL LARGE EARLY RELEASE PROBABILITY FOR CONDITIONS WHERE A LARGE EARLY RELEASE MITIGATING SYSTEM IS UNAVAILABLE

This section evaluates the LERP for instances where the primary impact of component unavailability is to downgrade the ability of the plant to prevent a core damage event from proceeding to a large early release. An example component in this category is the containment. Since large early releases are not impacted by incremental changes in containment leakage, the primary risks to ensuring the containment integrity, from a LERP perspective, result from a gross opening in the containment (such as a stuck open purge valve(s)) or structural anomalies which would significantly decrease the capability of containment to withstand a severe challenge.

The LERP impact of the inoperability of this component/system is established by assuming that when a system such as this is non-functional, all core damage events will proceed to a large early release. Based on RG 1.174 (Reference 1), the goal for incremental changes in LERP is that the change should result in a risk increase less than 1.0E-7. Since the core damage frequency (internal plus external events) is less than 1.0E-4 per year for typical PWRs (See Reference 8), the minimum time required to accumulate the risk goal target of 1.0E-7 may be calculated as:

$$\text{ICLERP} = (\text{CDF}) \cdot \frac{\Delta T}{8760}$$

A risk-informed AOT for containment inoperability may be established by solving for ΔT as follows:

$$\Delta T = [ICLERP_{goal} / (CDF)] * 8760$$
$$= 1.0E\text{-}7/1.0E\text{-}4 \times 8760$$
$$\approx 9 \text{ hrs}$$

This risk-informed assessment supports an AOT for containment inoperability of 8 hours.

4.4 ASSESSMENT OF OTHER DESIGN BASIS SYSTEMS

This section considers the impact of the AOT extensions on the plant when the system inoperability impacts neither core damage nor large early release probabilities. These systems can have a variety of functions. Availability of such equipment is typically required to meet design basis dose assessments, or support the equipment qualification envelope that provides protection to the containment for negative pressure events. The systems captured in this category include:

- Iodine Cleanup System (ICS)
- HVAC and Filtration Envelope
- Shield Building Emergency Air Cleanup System (EACS)
- Control Room Emergency Air Cleanup System (CREACS)
- Control Room Emergency Air Temperature Control System (CREATCS)
- Penetration Room Emergency Air Cleanup System (PREACS)
- ECCS Penetration Room Emergency Air Cleanup System (ECCS PREACS)
- Containment Spray System (CSS)

An assessment of the impact of the unavailability of these systems is presented below. A summary of the risk-informed AOTs is presented in Table 4.4-1.

4.4.1 HVAC and Filtration Envelop and ICS

The determination of these AOTs is based on the concept that equipment/function inoperability is acceptable provided that the potential for challenging the equipment in this category during the proposed AOT is acceptably low (incremental system challenge of less than 1.0E-6). That is,

$$\text{Incremental System Challenge} = (CDF)\frac{\Delta T}{8760}$$

where the CDF is assumed to be equivalent to the significant containment radiation release frequency.

Using this method, the risk-informed AOTs for the ICS and components of the HVAC and filtration envelope (with the exception of the ECCS PREACS) can be established by assuming that they will be challenged during all core damage events (approximately 1.0E-4 per year). The resulting AOT for these systems and components is 87 hours (See Table 4.4-1).

The ECCS PREACS is assumed to be challenged for all large and medium LOCAs (4.5E-5 per year). The challenge was limited to these events since recirculation cooling is generally not needed for the higher frequency smaller LOCA breaks. Using the nominal LOCA frequency, the resulting AOT for the ECCS PREACS is 195 hours (See Table 4.4-1).

Table 4.4-1: Summary of Recommended AOTs based on Limiting Challenge Probability to less than 1.0E-6

System	Proposed AOT for Inoperable System (hrs)*	Challenge Frequency (per year)	System Challenge Probability for Extended Entry into Proposed AOT	Time Required to Reach 10^{-6} Challenge Probability (hrs)
Iodine Cleanup System	24	1.0E-4 +	2.7E-7	87
Shield Bldg. EACS	24	1.0E-4 +	2.7E-7	87
CR EACS/EATCS	24	1.0E-4 +	2.7E-7	87
PREACS	24	1.0E-4 +	2.7E-7	87
ECCS – PREACS	24	4.5E-5 +	2.7E-7	195
CS#	72	1.0E-4 +	8.1E-7	87

+ Representative Bounding Estimate of Total Core Damage Frequency.
* Both trains inoperable
# With CARC operable

4.5 TRANSITION RISK CONSIDERATIONS

There is an "at power" increase in risk associated with any given AOT extension. This increase may be negligible or significant. A complete approach to assessing the change in risk accounts for the effects of avoided shutdown, or "transition risk." Transition risk represents the risk associated with changing the operating mode of a PWR from its nominal full power operating state to a lower shutdown mode following equipment inoperability. Transition risk is of interest in understanding the tradeoff between shutting down the plant and restoring the affected systems/components to operable status while the plant remained at power. When establishing a risk decision making process consistent with Regulatory Guides 1.174 and 1.177, the risk of transitioning from "at power" to a shutdown mode can be balanced against the risk of continued operation and performing corrective maintenance.

Plant transitions expose the plant to additional operational risk. This risk is typically accumulated in a short time frame. The increased risk from plant transition arises from the impact of the plant transition on increased plant trip and loss of power event frequencies, and by errors occurring during valve and system realignments required by some transitions. Common plant transitions are from full power to the shutdown modes. The risk of transitioning a plant from full power to Mode 4 with Auxiliary Feedwater (AFW) in service have been estimated to be on the order of 1.0E-6 for an uncomplicated shutdown (See for example, Reference 8).

In addition to the transition risk from power to a shutdown mode, transitions between shutdown modes and between operating configurations are also important. Based on a review of shutdown procedures, the transition risk from Mode 3 to Mode 4 as it affects AFW is relatively transparent and is judged to be low. However, entering SDC creates additional risks which are associated with the reconfiguration of the RCS. The additional risk is dominated by inventory loss events associated with misalignment of valves during entry into SDC or a Low Temperature Overpressure Protection (LTOP) relief valve lifting. These events are generally of short duration, and are important during the initial alignment of SDC. Due to the lower decay heat at shutdown, the ICCDP associated with these events is on the order of 1.00E-6.

As long as the incremental "at power" risk is low (i.e. having a ICCDP ≈ 1.0E-6 or less), avoidance of a plant transition will likely offset any accumulated "at power" risk. In any event, use of the Regulatory Guidance (RGs 1.174 and 1.177) and acknowledging the low potential for TS entry ensures that the accumulated risks due to these proposed TS modifications is negligible.

4.6 END STATES AND SHUTDOWN RISKS

The current effort is directed towards establishing an Action for conditions where a system function is lost. In most of these instances the current TS either requires a Mode 5 end state or entry into LCO 3.0.3 which also results in a Mode 5 end state.

Reference 5 discusses the risk associated with the various shutdown modes for CE designed NSSSs. The assessment concluded that for shutdowns of short duration, Mode 4 (hot shutdown) is the lowest risk shutdown mode when the Auxiliary Feedwater (AFW) system is operational. This lower risk is a combined result of the increased redundancy and diversity of equipment available for core heat removal. That is, while in Mode 4, decay heat removal may be established via turbine or motor driven AFW pumps[3] or via the Shutdown Cooling system (SDC). It is therefore recommended that when a Mode 4 end state does not presently exist, the Mode 4 end state replace the current (Mode 5, cold shutdown) end state for most of the Technical Specifications considered in this report. In addition, the Mode 4 shutdown end state on AFW minimized plant configuration changes and associated transitional risks.

In a few instances the recommended end state is not changed (retained as Mode 5) or changed to Mode 3. The specific bases for the end state recommendations are presented in TS specific discussions of Section 5. A discussion of the basis for not requiring a Mode 5 end state is discussed in response to question 11 of the request for additional information (Reference 21).

The times recommended for Mode 3 or Mode 4 transitions are consistent with those contained in NUREG-1432 (Reference 3). That is, Mode 1 to Mode 3 transitions should be completed in 6 hours, and Mode 1 to Mode 4 transitions should be completed in 12 hours.

4.7 MAINTENANCE RULE

The risk associated with implementation of these proposed TS changes will be managed in accordance with the provisions set forth in 10 CFR 50.65(a)(4) and Regulatory Guide 1.182. This will assure proper plant configuration control during entry into these LCOs.

[3] Ft. Calhoun Station also has a diesel driven AFW pump.

5.0 SYSTEM EVALUATION

This section provides a summary of the basis for the change of each of the risk-informed TS end state changes proposed. The format of each of the subsequent subsections will be as follows:

i) Description
ii) Plant Applicability
iii) Limiting Condition for Operation
iv) Licensing Basis for LCO
v) Condition Requiring Entry into a Shutdown Action Statement
vi) Proposed Modification to Actions
vii) Basis for Proposed Change
viii) Defense-in-Depth Considerations
ix) Tier 2 Restrictions

Since the TS changes being proposed generally are associated with the inoperability of an entire system (or unavailability of a given function) defense-in-depth is not maintained in the sense of assuming equipment redundancy. Instead, public safety is maintained by ensuring public risk is acceptably low and by providing an opportunity to repair the equipment during power operations thereby potentially avoiding the additional risk of plant transitions.

This section provides an integrated discussion of the risk and deterministic issues, focusing on specific technical specifications. Risk assessments presented in the following sections are quantified in Section 4.

In establishing the modified TS Actions (AOTs/CTs) it was tacitly assumed that:

- The purpose of the Action is to complete a short duration repair of the inoperable system/component.
- When a Mode 4 end state is recommended, the AFW system is not impaired.
- Mode 5 end states are supported by a fully functional shutdown cooling system.
- Times for end state entry are as follows:
  - Transitions from Mode 1 to Mode 3 are required to be less than 6 hours.
  - Transitions from Mode 1 to Mode 4 are required to be less than 12 hours.

The proposed AOTs provide the operating staff additional time to restore system/component operability while the plant remains at power. Expeditious restoration of operability "at power" reduces the overall risk of plant operation. Specifically, the extended AOTs allow additional time for the plant staff to restore system/components to operability and take appropriate corrective actions while the plant remains at power. This could avoid risks associated with unnecessary plant transitions.

The requirement for an immediate (1 hour) shutdown is based on the philosophy that inoperability of the containment is a violation of the plant design basis and a shutdown is warranted. The selection of 1 hour was chosen as a surrogate for immediately and that shutdown plans can be effected is that time frame. The goal was to place the plant in a condition where the health and safety of the public could be better assured. However, no specific risk assessments were performed to determine the 1 hour AOT. The AOT extensions proposed in this report have the same goal, but are "risk-informed" in that in establishing the AOT, the risk of continued plant operation, as well as the risks introduced by a plant shutdown are considered. When considering plant risk, it is often risk beneficial to allow restoration of system/components inoperability "at power" rather than to initiate a 1 hour shutdown. That is, the extended AOTs, as proposed, meet the intent of the initial one hour shutdown by providing AOTs based on risk insights. Furthermore, should a shutdown be required, Mode 4 is an acceptably safe end state (See Reference 4).

5.1 STANDBY SAFETY SYSTEMS

5.1.1 LCO 3.1.9 – Boration Systems - Operating

The boration systems are required to ensure that adequate shutdown reactivity margin exists to bring the plant to cold shutdown with the most reactive Control Element Assembly (CEA) not fully inserted and the decay of all xenon poison. The systems are also intended to mitigate possible return to power scenarios following a Main Steam Line Break (MSLB) and to mitigate ATWS events. Boration systems are not included in NUREG-1432, since it does not satisfy any of the 10 CFR 50.36 criteria. TS 3.1.9 and non-ISTS plants TS require that boration systems are operable during the modes of applicability. Two boration paths that are to operable are: (1) the Refueling Water Storage Tank (RWST) and its flow path to the charging pumps, and (2) one or both Boric Acid Makeup (BAMU) tanks with their respective flow paths to the charging pumps.

Plant Applicability

ANO-2, Millstone 2, SONGS 2 & 3, St Lucie 1 & 2, Waterford 3

Limiting Condition for Operation (LCO)

Default entry into LCO 3.0.3 when both boration paths are unavailable in Modes 1, 2, 3 & 4.

Licensing Basis for LCO

The boration systems are required to ensure that adequate Shutdown Margin (SDM) exists to bring the plant to Mode 5 (cold shutdown) with the most reactive CEA stuck out and the decay of all xenon poison. The systems are also intended to mitigate possible return to power scenarios following an MSLB or Reactor Coolant Pump restart. Boration systems are also necessary to ensure power reduction during a ATWS events.

Condition Requiring Entry into a Shutdown Action Statement

Both boration paths inoperable, as follows: 1) the RWST and its flowpath to the charging pumps, and 2) both BAMU tanks with their respective flowpaths to the charging pumps.

Proposed Modification to Actions

Increase the time available to take action to restore one boration flow path to operable status to 24 hours for the cases in which both boration paths are inoperable, and allow Mode 3 as the final end state for conditions where the boric acid source tank volume, temperature or concentration are out of limits.

Basis for Proposed Change

The boration system provides the normal means to establish Shutdown Margin (SDM) and RCS boration as RCS temperature is reduced. However, from a core damage perspective, the risk importance of the boration system is low. For example in the SONGs Probabilistic Risk Assessments (PRA), Chemical and Volume Control System injection function is modeled only for small-small LOCA, SGTR and ATWS. The impact of charging flow on LOCAs and SGTRs is small, since both types of initiating events may be effectively mitigated via HPSI. However, HPSI is not an effective backup for ATWS events since ATWS events will rapidly repressurize above the HPSI shutoff head.

If it is assumed that the plant can shutdown with both boration pathways unavailable, then the risk increase associated with providing an AOT of 24 hours is computed based upon the risk increase resulting from the inability of the plant to mitigate ATWS events during the time interval the boration systems are unavailable. This risk assessment approach is consistent with results of the SONGS PSA which indicate that the risk increase is dominated by a turbine trip–induced ATWS. For a Mode 1 system inoperability, the increase in core damage probability is about 4.5E-8, which is an acceptably small increase (See Section in 4.1). In shutdown modes, ATWS events are precluded and the associated risk is negligible.

The risk impact of boration system unavailability during this interval is low. HPSI subsystem availability will minimize the impact of an inoperable boration system for non-ATWS events.

ICLERP results associated with this extended AOT are established in Section 4.2. Conservatively, assuming that all incremental core damage events proceed to high pressure core damage states, the ICLERP is 3.2E-9. Even then, the resulting ICLERP is well below the RG 1.177 incremental risk (ICLERP) goal 5.0E-8 for a TS change.

A Mode 3 end state is recommended for conditions where the tank contents are out of limits, as entry into Mode 3 will further reduce (or eliminate) the risk impact of boron system unavailability and further mode changes are complicated by lack of boration capability during plant cooldown. Maintaining the plant in this mode also eliminates concurrent transient risk associated with plant mode changes.

Defense-in-Depth Consideration

In the event that a loss of redundancy of charging pumps occurs, the impact on plant risk will be very small since boration (and injection) may be provided by other injection equipment (e.g. HPSI pumps) for many events. Therefore, the availability of HPSI during this interval ensures the plant Defense in Depth is maintained. During operational periods when the Moderator Temperature Coefficient (MTC) < 0, the Mode 3 end state is also the end state with the least boration demand. It should further be noted that from a shutdown margin perspective, that when MTC is negative, increased boration is required at lower temperatures. For plant conditions with a negative MTC, at similar boron concentration levels, Mode 3 should have greater SDM than Mode 4. Either mode would have greater shutdown margin than Mode 5.

Tier 2 Restrictions

None.

5.1.2 (ISTS) LCO 3.4.9 – Pressurizer Heaters

The pressurizer provides a point in the RCS where the liquid and vapor water phases are maintained in equilibrium under saturated conditions for pressure control purposes to prevent bulk boiling in the remainder of the RCS. The pressure control components addressed by this LCO include the pressurizer, the required groups of heaters and their controls and the Class 1E power supplies. The liquid to vapor interface exists to permit RCS pressure control, using the sprays and heaters during normal operation and in response to anticipated design basis transients.

The unavailability of the Class 1E pressurizer heaters covered by this TS may complicate steady state plant pressure control and may increase the potential of an unplanned reactor trip.

Class 1E powered pressurizer heaters are used post accident to maintain plant subcooling during a Natural Circulation (NC) cooldown. The unavailability of the pressurizer heaters during an NC cooldown will extend the time to reach Shutdown Cooling System entry conditions. However, core/RCS heat removal will be adequately established via the use of SG cooling.

Plant Applicability

All except St Lucie-2

Limiting Condition for Operation (LCO)

Two groups of pressurizer heaters each with a capacity ≥ (150 KW) [capable of being powered from an emergency power supply,] operable in Modes 1, 2 and 3.

Licensing Basis for LCO

All analyses performed from a critical reactor condition assume the existence of a steam bubble and saturated conditions in the pressurizer. Safety analyses presented in the Final Safety Analysis Report do not take credit for pressurizer heater operation; however, an implicit initial condition assumption of the safety analyses is that the RCS is operating within its normal operating pressure band and pressurizer level is in the programmed band. The TS requires both the existence of an adequately sized pressurizer steam bubble and two groups of pressurizer heaters [capable of being powered by emergency AC power] to maintain pressure control. The emergency powered heaters are used, in particular, to help maintain subcooling in the RCS loops during natural circulation cooldown conditions that would exist during a LOOP event. While a LOOP is a coincident occurrence assumed in the accident analyses, maintaining hot, high pressure conditions over an extended time period is not evaluated in the accident analyses.

Condition Requiring Entry into a Shutdown Action Statement

Default entry into LCO 3.0.3 is required when two safety-related pressurizer heater groups are inoperable.

Proposed Modification to Actions

Include new action statement for two groups of [required] pressurizer heaters inoperable. Allow an outage time of 24 hours to restore one group of safety-related pressurizer heaters before requiring entry into the existing Condition C (Mode 3 in 6 hours, Mode 4 in 12 hours).

Basis for Proposed Change

Pressurizer heaters enable RCS pressure to be readily controlled within its normal operating pressure band. Plants have more than the two groups of heaters required by this specification. Failure of the TS heater group is not expected to result in an inability to control RCS pressure. If loss of these heaters results in loss of plant pressure control, emergency procedures would dictate appropriate action. For the purpose of this evaluation, loss of these heaters is conservatively assumed to reduce the ability of the operator to control the plant within its normal operating band and consequently increase the potential of plant trip. Therefore, the risk impact was assessed as the typical risk of an uncomplicated plant trip.

It should be noted that inoperability of the safety-related heaters during the 24 hour period requested would not have any significant impact on the plant transient response. Therefore no quantifiable change in CDF or LERF would be expected. It should be noted that the existence of a pressurizer steam bubble is implicitly assumed in the PSA and pressurizer heaters are normally not modeled.

Pressurizer heaters are beneficial in assisting the recovery from a SGTR and for post-accident transitioning to long-term cooling. However, since a number of non-safety related heater banks are also available, the only scenarios that would be impacted would be those that involved an extended LOOP following a plant transient or accident. Also, while the unavailability of pressurizer heaters may complicate post-trip cooldowns, a successful cooldown is expected with a minimal impact on plant risk due to the availability of the RV head and pressurizer vents.

The risk impact of pressurizer heater system inoperability is conservatively assessed assuming that the unavailability of the pressurizer heaters increases the probability of plant trip from 0.004 per day (about 1.5 per year) to 0.05. This implies that during the proposed 24 hour AOT, the plant has a 5% chance of a plant trip during the time interval that the Class 1E pressurizer heaters are inoperable. A review of the CE designed NSSSs indicates that the conditional core damage probability associated with an uncomplicated plant trip is 6.0E-6. This results in an incremental CDP of 2.9E-7 (See Section 4.1.2). The resulting LERP increment is 1.1E-8 (See Section 4.2). Both results are below the RG 1.174 incremental risk guidelines and derivative RG 1.177 guidance as discussed in Section 4.

Note, when the inoperability of the pressurizer heaters does not affect plant operation (such as a loss of emergency power supply), the core damage incremental risk will be negligible.

Defense-in-Depth Consideration

Both safety-related and non-safety related heaters are normally available, providing considerable system redundancy for many transient events (except following a loss of offsite power event).

Without the pressurizer heaters, a natural circulation cooldown may be required (as 20 °F subcooling may not be assumed). Such cooldowns may be conducted via use of the pressurizer and RV gas vent lines, and SG venting via the Atmospheric Dump Valves (ADVs).

Tier 2 Restrictions

None.

5.1.3 LCO 3.4.11 PRESSURIZER PORVS & ASSOCIATED BLOCK VALVES

PORVs are automatically opened at a specific set pressure when the pressurizer pressure increases and automatically close on decreasing pressure. The PORVs may be manually operated using controls installed in the control room.

An electric, motor-operated, normally open, block valve is installed between the pressurizer and the PORV. The function of the block valve is to ensure RCS integrity by isolating a leak or stuck open PORV. Block valve closure is accomplished manually using controls in the control room and may be used to isolate a leaking PORV to permit continued power operation. Most importantly, the block valve is used to isolate a stuck open PORV in order to restore the RCS pressure boundary integrity. Block valve closure terminates the RCS depressurization and coolant inventory loss.

The PORV and its block valve controls are powered from normal power supplies. Their controls are also capable of being powered from emergency supplies. Power supplies for the PORV are separate from those for the block valve.

The PORV TS varies among CE NSSS plants. Several CE NSSSs are designed without PORVs. St. Lucie 2 and Palisades operate with one or more PORVs block valve closed (See Table 5.1.3-1).

Table 5.1.3-1: Summary of PORV/Block Value TS

Plant	Action Statement AOT/CT	Required Action End State when AOT/CT Not Met
Calvert Cliffs 1 & 2	Restore 1 PORV in 72 hours.	Mode 3 in 12 hours
Palisades	Close associated Block Valve (BV) in 1 hour and restore at least 1 PORV in 2 hours	Mode 3 in 6 hours
Fort Calhoun Station	Restore 1 PORV in 1 hour or close both BVs	Mode 4 in 42 hours (PORVs) Mode 4 in 42 hours (BVs)
Millstone 2	PORVs: restore 1 in 1 hour Block valves: Restore in 2 hours	Mode 4 in 12 hours
St. Lucie 1 & 2	None on PORVs, TS on Block Valve only.	Mode 5 in 36 hours (BVs)

Plant Applicability

Calvert Cliffs 1 & 2, St Lucie 1 & 2 (Block Valves), Millstone 2, Palisades, Fort Calhoun Station

Limiting Condition for Operation (LCO)

Each PORV and associated block valve shall be operable in Modes 1, 2 & 3.

Licensing Basis for LCO

The primary purpose of this LCO is to ensure that the PORVs and the block valves are operable so the potential for a small break LOCA through the PORV pathway is minimized.

The PORV functions as an automatic overpressure protection device and limits challenges to the primary safety valves. Overpressure protection for the RCS is provided by the primary safety valves (PSVs), and the safety analyses do not take credit for the PORV opening for accident mitigation.

The PORV setpoint is above the high pressurizer pressure reactor trip setpoint and below the opening setpoint for the PSVs. The purpose of the relationship of these setpoints is to limit the number of transient pressure increase challenges that might open the PSV, which, if opened, could fail in the open position. The PORV setpoint thus limits the frequency of PSV challenges from transients, and the PORV block valve limits the possibility of a small break LOCA from a failed open PORV. Unlike the PORVs, the PSVs cannot be isolated if they fail to re-close after opening.

The PORVs may be manually operated to depressurize the RCS as deemed necessary by the operator in response to abnormal transients or accidents. The PORV may be used for RCS depressurization when the pressurizer spray is not available, a condition that may be encountered during a loss of offsite power. Operators can manually open the PORVs to reduce RCS pressure in the event of a Steam Generator Tube Rupture (SGTR) with offsite power unavailable.

The PORVs may also be used for feed and bleed (once through core cooling) in the case of multiple equipment failure events that are not within the design basis, such as a total loss of feedwater.

For some PWRs, PORVs also provides Low Temperature Overpressure Protection (LTOP) during heatup and cooldown. LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System," addresses this function.

<u>Condition Requiring Entry into a Shutdown Action Statement</u>

Various LCO entry requirements exist for both PORVs inoperable or both block valves inoperable. ISTS 3.4.11 requires the plant to restore 1 PORV to operable status or prepare to shutdown in 1 hour and enter Mode 4 in 12 hours. When both block valves are inoperable, ISTS 3.4.11 requires restoring at least 1 block valve in 2 hours or entering Mode 4 in 12 hours. Palisades TS requires the plant to be in Mode 3 in 8 hours if both PORVs are inoperable. Calvert Cliffs allows 72 hours to restore one PORV to operable. Status of the PORV the plant is required to be in Mode 3 in 6 hours.

St Lucie 1 & 2 has no PORV TS, but allows 1 hour to restore or close an inoperable block valve or be in Mode 5 in 36 hours. For convenience, highlights of the PORV TS for CE designed NSSSs are summarized in Table 5.1.3-1. Plant specific TSs should be consulted for additional details.

<u>Proposed Modification to Required Actions</u>

Revise ISTS 3.4.11 Condition E (or equivalent) CT to be consistent with other CE designed NSSSs (with PORVs) to allow 8 hours to restore one PORV to operable status for conditions where a PORV is unable to re-close once challenged, but may be isolated. However, this extension does not apply to PORVs that are leaking, and that can not be isolated by block valves, or to PORVs that are not expected to be isolable following a demand.

Revise ISTS 3.4.11 Condition F.2 CT to allow 8 hours to restore one block valve to operability status for conditions where the associated PORV is unable to re-close.

Basis for Proposed Change

The PORV functions as an automatic overpressure protection device and limits challenges to the Primary Safety Valves. However, overpressure protection is provided by the Primary Safety Valves, and the analyses do not take credit for the PORV opening for accident mitigation. Section 4.1 indicates that the increased CDP associated with extending the CT/AOT to 8 hours for inoperable PORVs (unable to open) is small, 8.1E-07.

Defense-in-Depth Consideration

The PORVs limit the number of pressure transients that may challenge the PSVs. Experience indicates that challenges to PORVs or PSVs are rare and that the PSVs are highly reliable. As a result, 3410 Mwt and 3800 Mwt CE NSSS designs do not include PORVs. A core heat removal application of PORVs was identified post-TMI. PORVs may also be used to control offsite releases following a limited class of severe accidents. PSVs provide overpressure protection for the RCS.

Tier 2 Restrictions

None.

5.1.4 (ISTS) LCO 3.5.1 – Safety Injection Tanks

The Safety Injection Tanks (SITs) are pressurized passive injection devices used to effect rapid refill of the RCS following the onset of Large Break LOCAs. The SITs are partially filled with borated water and pressurized with nitrogen gas. These devices are passive components, since no operator or control action is required for them to perform their function. The internal tank pressure is sufficient to discharge the contents to the RCS, when the RCS pressure decreases below the SIT pressure.

Each SIT is piped into one RCS cold leg via the injection lines utilized by the High Pressure Safety Injection (HPSI) and Low Pressure Safety Injection (LPSI) systems. Each SIT is isolated from the RCS by two check valves in series. The motor operated isolation valve in the SIT flow path is normally open, with power removed from the valve motor to prevent inadvertent closure prior to or during an accident.

Additionally, the isolation valves are interlocked with the pressurizer pressure instrumentation channels to ensure that the valves will automatically open as RCS pressure increases above the SIT pressure and to prevent inadvertent closure prior to an accident. The valves also receive a Safety Injection Actuation Signal (SIAS) to open. This ensures that the SITs will be available for injection without reliance on operator action.

Plant Applicability

All

Limiting Condition for Operation (LCO)

Explicit LCO 3.0.3 entry for 2 or more SITs inoperable during Modes 1, 2 and 3 with pressurizer pressure > [700] psia.

Licensing Basis for LCO

When more than one SIT is inoperable, the unit is in a condition outside of its design basis accident analyses. Therefore, LCO 3.0.3 must be entered immediately. The LCO establishes the minimum conditions required to ensure that the SITs are available to accomplish their core cooling safety function following a LOCA. CENP licensing analyses consider four SITs to be operable. The Operability of four SITs ensures that the contents of three SITs will be injected into the RCS following a large LOCA. The water from the SITs serves to rapidly refill the RV and shortens the adiabatic heatup, thus helping to limit the peak clad temperature to ≤ 2200 °F.

For a SIT to be considered OPERABLE, the isolation valve must be fully open, power removed above [2000] psig, and the limits established in the Surveillance Requirements for contained volume, boron concentration and nitrogen cover pressure must be met.

Although cooling requirements decrease as core power decreases, the SITs are still required to provide core cooling as long as elevated RCS pressures and temperatures exist. Therefore, the SITs are also required to be operable in Modes 2 and 3 with pressurizer pressure ≥ [700] psia.

Condition Requiring Entry into a Shutdown Action Statement

LCO condition [D] requires immediate entry into LCO 3.0.3 if two or more SITs are inoperable.

Proposed Modification for Actions

Many CE designed NSSSs have-been granted an extended AOT for the inoperability of one SIT.

Revise Condition D to allow 24 hours for two or more inoperable SITs.

Basis for Proposed Change

SIT availability may alter the progression of smaller break LOCAs of, and potentially alter the extent of core damage. However, the impact on the event core damage potential will be negligible. The SITs are needed primarily to mitigate the Large Break LOCA event. Therefore, even if one assumes all Large Break LOCAs are not successfully mitigated (that is, proceed to a core damage condition), the risk impact of a short duration unavailability is negligible. Based on the calculations of Section 4.1 and 4.2, the ICCDP associated with a 24 hour CT/AOT is 1.3E-8. Similarly for LERP, the conservative bounding calculation results in an ICLERP of 3.9E-11. These results confirm that the risk impact of the CT/AOT extension is negligible.

Defense-in-Depth Consideration

The unavailability of the SITs will compromise the ability of the plant to respond to Large Break LOCA events. In this same instance, the unavailability of 2 or more SIT(s) will result in an extended fuel heatup and effect the extent of fuel damage that may occur for a limited range of small LOCA break sizes. Depending on the severity of the transient and degree of inoperability of the SITs, a core damage condition may arise. Long term core cooling will be assured via availability of the plant's LPSI and HPSI subsystems. It is proposed that the current requirement for an "immediate" response be extended to include the risk-informed interval of 24 hours. As a result of the low anticipated frequency of occurrence of a Large Break LOCA, a 24 hour CT/AOT to restore SIT operability is appropriate. At the end of this period, the operator will be instructed to exit the LCO via resolution of the problem, or take actions to bring the plant to hot shutdown.

The proposed CT/AOT is consistent with the requirements of 10 CFR 50.46 which require that the license propose immediate steps to "bring plant design or operation" into compliance by ensuring the defined outage time is commensurate with the risk significance of the system. Availability of both LPSIs and all HPSIs will limit the impact of SIT unavailability. Maintenance rule assessment per 10 CFR 50.65(a)(4) will ensure the integrated risk of this inoperability is small.

Tier 2 Restrictions

None.

5.1.5 (ISTS) LCO 3.5.2 ECCS - Operating (High Pressure Safety Injection System)

Two redundant, 100% capacity ECCS trains are required to be OPERABLE in MODES 1, 2 and 3, (with pressurizer pressure ≥ [1700] psia). Each train consists of a High Pressure Safety Injection (HPSI) and a Low Pressure Safety Injection (LPSI) subsystem.

A suction header supplies water from the RWST or the containment emergency sump to the HPSI pumps. Separate piping supplies each HPSI train. The discharge headers from each HPSI pump divide into four supply lines. Both HPSI trains feed into each of the four injection lines. Control valves or orifices are set to balance the flow to the RCS. This flow balance directs sufficient flow to the core to meet the analysis assumptions following a LOCA in one of the RCS cold legs.

There are two phases of HPSI operation; injection and recirculation. In the injection phase, borated water stored in the RWST is added to the Reactor Coolant System (RCS). Initially injection is added via the cold legs. After the RWST has been depleted, the HPSI recirculation phase is entered and the HPSI suction is automatically transferred to the containment emergency sump. Several hours following a large LOCA, recirculation flow is delivered to the RCS via the hot and cold legs.

<u>Plant Applicability</u>

All

<u>Limiting Conditions for Operation (LCO)</u>

In MODES 1, 2 and 3, with pressurizer pressure ≥ [1700] psia, both trains of HPSI must be operable. In general, when 2 HPSI trains are inoperable, an explicit entry into LCO 3.0.3 is required (See for example Reference 3).

<u>Licensing Basis for LCO</u>

The function of the HPSI subsystem is to provide RCS inventory control, core cooling and negative reactivity to ensure that the reactor core is protected after any of the following accidents:

a. Loss of Coolant Accident (LOCA);
b. Control Element Assembly (CEA) ejection accident;
c. Loss of secondary coolant accident, including uncontrolled steam release or loss of feedwater; and
d. Steam Generator Tube Rupture (SGTR).

HPSI subsystems are assumed to be operable in the design basis large and small design basis LOCA analyses. The SGTR and MSLB analyses also credit HPSI for event mitigation.

This LCO ensures that the HPSI pump will deliver sufficient water during a small break LOCA and provide sufficient boron to maintain the core subcritical following an MSLB. The addition of negative reactivity is designed primarily for the MSLB where a primary cooldown could add enough positive reactivity to achieve criticality and return to significant power with rod that fails to insert.

Condition Requiring Entry into a Shutdown Action Statement

The inoperability of two HPSI subsystems will result in an explicit entry into LCO 3.0.3 in accordance with ISTS Condition D.

Proposed Modification to Required Actions

It is proposed that a condition be added to the LCO addressing actions to be taken following the inoperability of both HPSI pumps (or HPSI subsystem). The action would allow 4 hours to-restore one train of HPSI subsystem before commencement of a plant shutdown.

The next section discusses changes to the Low Pressure Safety Injection subsystem requirements. When taken with these proposed changes, the existing condition of "Less than 100% of the ECCS flow equivalent to a single OPERABLE train available" will no longer be needed as that condition will be addressed by the Conditions for two HPSI subsystems inoperable or two LPSI subsystems inoperable.

Basis for Proposed Change

The availability of the HPSI subsystem is extremely important in ensuring that the plant is capable of responding to a wide range of plant upsets. The following results are based on the calculations of Section 4.1. Table 4.1-2 indicates that for a short duration (4 hrs) inoperability of both HPSI subsystems would result in an ICCDP between 1.1E-6 and 1.6E-6, depending on whether or not the plant is equipped with PORVs. The range of the corresponding ICLERP is between 1.8E-8 and 2.9E-8. The risk associated with system inoperability of both subsystems in this time frame is partially offset by plant risks associated with mode transition and shutdown. These assessments are considered bounding and generic in that they do not include consideration of partial subsystem inoperabilities, due to valve inoperabilities, or credit the availability of alternate injection equipment and backup accident management strategies that may be available to the plant operator during many of these scenarios.

Defense-in-Depth Consideration

The LCO requires the operability of a number of independent subsystems. In many instances, due to the redundancy of trains and the diversity of subsystems, the inoperability of one component in a train does not necessarily render the HPSI subsystem incapable of performing its function. Neither does the inoperability of two different components, each in a different train, necessarily result in a loss of function for the ECCS. Examples of typical inoperabilities would include the unavailability of a single header injection valve or degradation of HPSI delivery curves below minimum design basis levels. This risk-informed extension to the current one hour AOT/CT allows for the potential resolution of minor HPSI subsystem inoperabilities and provides time to prepare for a controlled plant shutdown while increasing very small incremental plant risks.

Additional defense in depth considerations includes preparation for the use of non-TS equipment, such as charging pumps. These components may be capable of mitigating a spectrum of small LOCA events. While evaluation did not assume availability of charging pumps, the overall plant risk can be further reduced as the charging pumps may be used to support accident responses to smaller sized pipe failure events and for events with one stuck open PORVs, PSVs, or SGTRs. Also, maintenance practices that minimize the simultaneous unavailability of similar equipment (e.g. SITs, LPSIs and swing HPSIs if available) will also help control risk.

The proposed 4 hour CT/AOT is consistent with the risk significance of the HPSI subsystem and the intent of 10 CFR 50.46 which requires the design basis of the ECCS be maintained. Incurred risks at this level are consistent with the maintenance rule and will require operations staff awareness and implementation of compensatory measures and work controls (e.g. limitation of concurrent maintenance, etc.).

Tier 2 Restrictions

None.

5.1.6 (ISTS) LCO 3.5.2 ECCS – Operating (Low Pressure Safety Injection System)

Two redundant, 100% capacity ECCS trains are required for plant operation in MODES 1, 2 and 3, (with pressurizer pressure ≥ [1700] psia). Each train consists of a High Pressure Safety Injection (HPSI) and a Low Pressure Safety Injection (LPSI) subsystem.

A suction header supplies water from the RWST or the containment emergency sump to the LPSI pumps. Separate piping supplies each LPSI train. The discharge from the LPSI pumps divides into four lines, each feeding the injection line to four RCS cold legs. Control valves or orifices are set to balance the flow to the RCS. This flow balance directs sufficient flow to the core to meet the analysis assumptions following a LOCA in one of the RCS cold legs.

There are two phases of ECCS operation: injection and recirculation. The LPSI subsystem increases the inventory in the RPV following events with a severe loss of inventory. The LPSI subsystem operates during ECCS injection phase only. In the injection phase, borated water from the RWST is added to the Reactor Coolant System (RCS) by the LPSI subsystem. Initially injection is via the cold legs. After the RWST has been depleted, the LPSI subsystem is normally shutdown and the ECCS recirculation phase is entered. During ECCS recirculation, the ECCS suction is automatically realigned to the containment sump for continued operation with the HPSI subsystem.

The LPSI pumps also support the shutdown cooling system. However, this function is not considered within the scope of this technical specification. The shutdown cooling functions of the LPSI pumps are addressed by the RCS Loop specifications and requirements for RCS and SDC loop operability, which encompasses feedwater, cooling water, instrumentation and control, etc.

Plant Applicability

All

Limiting Conditions For Operation (LCO)

In MODES 1, 2 and 3, (with pressurizer pressure ≥ [1700] psia), both trains of LPSI must be operable.

Licensing Basis for LCO

The LPSI subsystem is designed to enhance the reflooding of the core following a Large Break LOCA. These events are characterized by a rapid loss of RCS inventory accompanied by a significant decrease in RCS pressure. The high volumetric flow capability of the LPSI pumps allows for a timely RCS refill. The LPSI subsystems are not required to mitigate other design basis accidents.

The large break LOCA event with a loss of offsite power and a single failure (disabling one ECCS train) establishes the OPERABILITY requirements for the ECCS. During the blowdown stage of a LOCA, the RCS depressurizes as primary coolant is ejected through the break into the containment. The nuclear reaction is terminated either by moderator voiding during large breaks or Control Element Assembly (CEA) insertion during small breaks. Following depressurization, borated water is injected into the cold legs, flows into the downcomer, fills the lower plenum, and refloods the core.

Condition Requiring Entry into a Shutdown Statement

In the event that both LPSI trains are inoperable, the design basis assumptions for the large break LOCA analyses are not met and an explicit default entry into LCO 3.0.3 is required.

Proposed Modification Actions

The previous section discusses changes to the High Pressure Safety Injection subsystem requirements. When taken with these proposed changes, the existing condition of "Less than 100% of the ECCS flow equivalent to a single OPERABLE train available" will no longer be needed as that condition will be addressed by the Conditions for two HPSI subsystems inoperable or two LPSI subsystems inoperable. Taken together, the Conditions and Required Actions for ISTS 3.5.2 will be:

Condition	Required Action	Completion Time
1 LPSI subsystem inoperable	Restore subsystem to OPERABLE status	7 days
2 LPSI subsystems inoperable	Restore at least one subsystem to OPERABLE status	24 hours
1 HPSI subsystem inoperable	Restore subsystem to OPERABLE status	72 hours
2 HPSI subsystems inoperable	Restore at least one subsystem to OPERABLE status	4 hours
Required Action and associated Completion Time not met	Be in MODE 3 Reduce pressurizer pressure to < [1700] psia	6 hours 12 hours

Basis for Proposed Change

The design basis analysis requires that one subsystem of LPSI be available to suppress the peak fuel temperature heatup during a large LOCA event. The unavailability of the LPSI subsystems for this limited time interval will result in a small increase in CCDF of 4.5E-5 per year for the plant risk associated with large LOCA events. There is no significant impact of the unavailability of LPSI following SGTR events as for many systems the LPSI would be required to be aligned to the SDC to effect entry into Mode 5. The risk impact of a plant shutdown with availability of the SDCS will offset any operational increase. A short term unavailability of the LPSI subsystems will result in a negligible incremental increase in the plant risk associated with large LOCA events.

A risk assessment of the ICCDP and ICLERP associated with LPSI unavailability is presented in Tables 4.1-2 and 4.2-1b, respectively. These analyses indicate that the ICCDP is 1.2E-7 and the ICLERP is 3.5E-10 for the proposed 24 hour AOT duration. These results are offset by the risk of transitioning the plant to Mode 4 (> 1.0E-6) (See References 4 and 8).

Defense-in-Depth Consideration

The primary impact of the unavailability of the LPSI subsystems will be the reduction in the capability of the plant to provide RCS inventory makeup to accommodate a large LOCA. A twenty-four hour AOT/CT is proposed for this condition based on the low incremental plant risk associated with continued plant operation.

In accordance with maintaining defense in depth, SIT availability should be assured to offset the large LOCA risks associated with LPSI subsystem inoperability.

Maintenance Rule risk assessments will consider the risk impact of the unavailability of the LPSI pump to support decay heat removal in establishing work controls and in performing maintenance.

Tier 2 Restrictions

None.

5.2 CONTAINMENT SYSTEMS

The series of Containment Systems Technical Specifications (TSs) is primarily focused on ensuring containment integrity and limiting offsite exposures due to events leading to core damage. The TS impacted by the proposed risk-informed change is 3.6.1 (Containment).

5.2.1 (ISTS) LCO 3.6.1 Containment

Containment Systems TSs are primarily focused on ensuring containment integrity and limiting offsite exposures due to events leading to core damage.

The requirements stated in the LCO define the performance of the containment as a fission product barrier. Specifically, LCO 3.6.1 requires that the containment maximum leakage rate, L_a be limited in accordance with 10 CFR 50 Appendix J. Other LCOs place additional restrictions on containment air locks and containment isolation valves. The integrated effect of these TSs is to ensure that the containment leakage is well controlled within limits that assure that the post accident whole body and thyroid dose limits of 10 CFR 100 are satisfied following a Maximum Hypothetical Accident (MHA) initiated from full power. The inability to meet this leakage limit renders the containment inoperable.

As a fission product barrier, the containment has an important role in ensuring plant safety. While containment integrity issues will not impact the core damage probability, there is a direct relationship of containment integrity to LERP and the public health and safety. The ICLERP relationship has been used to establish a risk-informed AOT for conditions when the containment integrity is not assured.

Plant Applicability

All

Limiting Condition for Operation (LCO)

In Modes 1, 2, 3, and 4 containment shall be operable.

Licensing Basis for LCO

In Modes 1, 2, 3 and 4, a design basis accident (DBA) could cause a release of radioactive material into containment. DBAs of specific concern are LOCAs, MSLBs and CEA ejection accidents.

The containment performs as a fission product barrier in the event a radiological release occurs within the containment. Specifically, this LCO requires that the containment allowable leakage rate, L_a is limited in accordance with 10 CFR 50 Appendix J. In addition, other TS place restrictions on containment air locks and containment isolation valves. The integrated effect of these TSs is to ensure that the containment leakage is within limits that assure that the post accident whole body and thyroid dose limits of 10 CFR 100 are satisfied following a Maximum Hypothetical Accident (MHA) initiated from full power. Failure to meet this leakage limit renders the containment inoperable. Containment operability is defined as maintaining the total leakage within specified limits.

Condition Requiring Entry into End State

Containment is declared inoperable due to excessive leakage (including leakage from airlocks and isolation valves) for a time period greater than one hour. If the containment is not restored to operable status within one hour, a plant shutdown is required.

Proposed Modification to Required Actions

Revise the action to allow 8 hours to restore containment operability. Revise the end state to Mode 4 if the action is not met and the containment leakage is excessive due to reasons other than the inoperability of two or more Containment Isolation Valves (CIVs) in the same flow path.

Basis for Proposed Change

The recommended change applies to containment conditions where containment integrity is essentially maintained and adequate ECCS Net Positive Suction Head (NPSH) is expected following an event. This applies to conditions when containment leakage is far in excess of L_a. As long as the containment has not experienced gross failure, the proposed change is appropriate. Containment "leakage" at or near design basis levels is not explicitly modeled in the PSA. The PSA implicitly requires that containment "gross" integrity must be available to ensure adequate NPSH for ECCS pumps. In the Level 2 model, containment "leakage" is not considered to contribute to a large early release.

Defense-in-Depth Consideration

The requirement for an immediate (1 hour) shutdown is based on the philosophy that inoperability of the containment is a violation of the plant design and a shutdown is warranted. The selection of 1 hour was chosen as a surrogate for immediately and that shutdown plans can be effected is that time frame. The goal was to place the plant is a condition where the health and safety of the public could be better assured. No specific risk assessments were performed. In fact, it is more appropriate from a health objective viewpoint to consider the risk of continued plant operation as well as that introduced by the shutdown. In consideration of total plant risk, it is a beneficial short term risk to allow a small potential "at power" risk (to resolve a TS inoperability) than to undertake a 1 hour shutdown. That is, 8 hours, as proposed, meets the intent of the current one hour shutdown requirement.

Tier 2 Restrictions

None.

5.3 HVAC AND RADIOLOGICAL CLEANUP SYSTEMS

HVAC and radiological cleanup systems provide the plant with the capability to protect the control room personnel and control radiological exposure to site personnel and the public. These devices are typically not credited for core damage mitigation/prevention and do not impact the probability of a large early release. There are ancillary impacts of these systems on some of these functions particularly those that protect Control Room (CR) staff. Furthermore, the control of long-term releases is an important design basis function. The risk-informed AOTs for these systems were therefore determined based on the concept of expected challenge (See Section 4.4). That is, a risk-informed AOT should limit the probability of expected challenge to these systems to about 1.0E-6 per year.

5.3.1 (ISTS) LCO 3.6.10 Iodine Cleanup System (ICS)

The purpose of the ICS is to remove elemental iodine from the post-accident containment atmosphere. The system was initially installed based on the understanding that radiological iodine releases would be predominantly in elemental form. Decades of research have indicated that most iodine will be released in the form of Cesium Iodine particulates. Consequently, the impact of the system on public doses is negligible.

The ICS consists of two 100% capacity trains. Each train consists of a heater, cooling coils, prefilter, moisture separator, High Efficiency Particulate Air (HEPA) filter, charcoal adsorber, another HEPA filter and a fan. No credit is taken for the second HEPA filter that is primarily there to collect carbon fines from the charcoal adsorber. The heater maintains the air below 70% humidity before entering the charcoal adsorbers for iodine removal efficiency. The moisture separator functions to reduce the moisture content of the airstream.

Plant Applicability

Calvert Cliffs 1 & 2, St Lucie 1 & 2

Limiting Conditions for Operation (LCO)

Default entry into LCO 3.0.3.

Licensing Basis for LCO

For several PWRs, the ICS contributes to meeting 10 CFR 100 (Reference 9) siting requirement dose limits and supports General Design Criteria (GDC)-19 of 10 CFR 50 Appendix A (Reference 10) for Control Room (CR) doses. These design basis calculations assume a high concentration of elemental iodine in the fission product release (See References 11 and 12). Two ICS trains are provided to meet the requirement for separation, independence and redundancy. The moisture separators function to reduce the moisture content of the airstream.

Condition Requiring Entry into Shutdown Action

Both ICS trains inoperable.

Proposed Modification Actions

Add a condition, which allows 24 hours to restore one ICS train, when both ICS trains are inoperable before requiring a shutdown. Allow Mode 4 as final end state.

Basis for Proposed Change

The ICS functions together with the containment spray and containment air recirculation cooling systems following a DBA that causes the failure of the fuel cladding, and a release of radioactive material (principally iodine) to the containment. The ICS is specifically designed to respond to a MHA with a large assumed contribution due to elemental iodine.

The DBAs that result in a release of radioactive iodine within containment are a Loss of Coolant Accident (LOCA), a Main Steam Line Break (MSLB) or a Control Element Assembly (CEA) ejection accident. In the analysis for each of these accidents, it is assumed that adequate containment leak tightness is present at event initiation to limit potential leakage to the environment. Additionally, that the amount of radioactive iodine release will be reduced by the containment sprays.

There is no significant risk impact of extending the potential system inoperability to 24 hours (see Table 4.4-1). The system does not provide a preventive function with respect to core damage events. Furthermore, unavailability of the ICS will have no significant impact on anticipated radiological releases to the public or CR. This is due to: (1) iodine releases are predominantly particulate (see Reference 13), so that removal via sprays and will be effective, (2) availability of elemental iodine is low so that the ICS has a limited benefit and (3) containment leak tightness significantly limits potential releases. Significant release events that contribute to LERPs (such as containment bypass events and SGTR with a loss of secondary isolation) will bypass these filters regardless of their availability.

Modification of the TS to support a Mode 4 end state if the action is not met avoids the risks associated with an unnecessary mode transition and the increased redundancy and diversity of RCS heat removal equipment in Mode 4.

Defense-in-Depth Consideration

See above discussion.

Tier 2 Restrictions

None.

5.3.2 (ISTS) LCO 3.6.13 Shield Building Exhaust Air Cleanup System (SBEACS)

The SBEACS provides radionuclide removal capability for fission products leaked into the shield building. The SBEACS consists of two separate and redundant trains. Each train includes a heater, cooling coils, a prefilter, a moisture separator, a High Efficiency Particulate Air (HEPA) filter, an activated charcoal adsorber section for removal of radioiodines and a fan. Ductwork, valves and/or dampers and instrumentation also form part of the system.

Plant Applicability

St Lucie 1 & 2, WSES and Millstone 2.

Limiting Conditions For Operation (LCO)

Default entry to LCO 3.0.3.

Licensing Basis for LCO

The SBEACS is required to ensure that radioactive material leaking from the primary containment of a dual containment into the Shield Building (SB) (secondary containment) following a DBA is filtered and adsorbed prior to being exhausted to the environment. The loss of the SBEACS could cause site boundary doses, in the event of a DBA, to exceed the values given in the licensing basis. Only the upstream HEPA filter and the charcoal adsorber section are credited in the analysis. The system initiates and maintains a negative air pressure in the shield building by means of filtered exhaust ventilation of the shield building following receipt of a Safety Injection Actuation Signal (SIAS).

Condition Requiring Entry into Shutdown Action

Both trains inoperable.

Proposed Modification for End State Required Actions

Allow 24 hours to take action if both SBEACS trains are inoperable and allow Mode 4 as the final end state if the action is not met.

Basis for Proposed Change

Following a LOCA, the SBEACS establishes a negative pressure in the annulus between the shield building and the steel containment vessel. Filters in the system control the release of radioactive materials to the environment.

A risk-informed AOT is established based on the methodology described in Section 4.4. The unavailability of the SBEACS has no direct impact on ICCDP or ICLERP. This system does impact the magnitude of long term radionuclide releases. The resulting risk-informed AOT is proposed to be 24 hours.

Containment "leakage" at or near design basis limiting is not explicitly modeled in the PSA. The PSA implicitly requires that containment "gross" integrity must be available to ensure adequate NPSH for the ECCS pumps. In the Level 2 model, containment "leakage" is not considered to contribute to a large early release. If accidents were to occur in Mode 4, the resulting containment pressures would be significantly less than the DBA conditions. Hence, leakage would be further reduced. While in Mode 4, the probability of a LOCA or a MSLB is reduced compared to Mode 1.

The implied licensing basis assumption that Mode 5 is inherently a lower operational risk than in Mode 4 is not supported by risk evaluations. Mode 5 risks are either about equal to or likely greater than equivalent risks in Mode 4 and therefore produce radiation releases to containment on par with those of Mode 4. Furthermore, plant shutdown actions that require entry into SDC introduce potential containment bypass risks including LOCAs. Thus, based on these PSA insights, it appears that remaining in Mode 4 (vs. Mode 5) is an appropriate action while the SBEACS inoperability is corrected. This end state would maintain more mitigation systems available to respond to any event that could lead to a loss of RCS inventory or decay heat removal. Furthermore, in Mode 4 the SIAS and CIAS will be available to aid the operator in responding to events that threaten the reactor and/or containment integrity.

Defense-in-Depth Consideration

See above discussion.

Tier 2 Restrictions

None.

5.3.3 ISTS LCO 3.7.11 Control Room Emergency Air Cleanup System (CREACS)

The Control Room Emergency Air Cleanup System (CREACS) provides a protected environment from which operators can control the plant following an uncontrolled release of radioactivity, chemicals or toxic gas. Alternate designations of this system include the acronyms CREACS, CREVAS, CREVS, or CREAFS. The current TS requires CREACUS to be operable in Modes 1 through 4 to support the operator response to a DBA. Operability in Mode 5 and 6 may also be required at some plants for chemical and toxic gas concerns. Long-term plant operation in the presence of degraded CREACUS should be based on placing the plant in a state which poses the lowest plant risk. In general, plant operation in Mode 4 poses a lower operation risk of core damage than in Mode 5.

<u>Plant Applicability</u>

All

<u>Limiting Condition for Operation (LCO)</u>

In Modes 1, 2, 3 and 4, the CREACUS must be operable to limit operator exposure during and following a DBA. In Mode 5, the CREACUS is required to cope with the release from a rupture of an outside waste gas tank or external toxic gas challenges. During movement of irradiated fuel assemblies [and CORE ALTERATIONS], the CREACUS must be OPERABLE to cope with the release from a fuel handling accident.

<u>Licensing Basis for LCO</u>

The CREACUS provides a protected environment from which operators can control the unit following an uncontrolled release of radioactivity [chemicals, or toxic gas].

The CREACUS consists of two independent, redundant trains that recirculate and filter the control room air. Each train consists of a prefilter and demister, a High Efficiency Particulate Air (HEPA) filter, an activated charcoal adsorber section for the removal of gaseous activity (principally iodine), and a fan. Ductwork, valves or dampers, and instrumentation and controls also form part of the system, as do demisters that remove water droplets from the air stream. A second bank of HEPA filters follows the adsorber section to collect carbon fines, and to back up the main HEPA filter bank if it fails.

The CREACUS is an emergency system, part of which may also operate during normal unit operation. Upon receipt of the actuating signal(s), the normal air supply to the control room is isolated, and the stream of ventilation air is recirculated through the filter trains of the system. The prefilters and demisters remove any large particles in the air, and any entrained water droplets present to prevent excessive loading of the HEPA filters and charcoal adsorbers.

Actuation of the CREACUS places the system into either of two separate states of the emergency mode of operation, depending on the initiation signal. Actuation of the system to the emergency radiation state of the emergency mode of operation closes the unfiltered outside air intake and unfiltered exhaust dampers. The system is also aligned for recirculation of control room air through the redundant trains of HEPA and charcoal filters. The emergency radiation state initiates pressurization and filtered ventilation of the air supply to the control room. The toxic gas isolation state is the same as the emergency radiation state, except that the signal switches the control room ventilation to an isolation mode.

Condition Requiring Entry into a Shutdown Action

Both CREACUS trains inoperable in Modes 1, 2, 3 or 4.

Proposed Modification for Actions

Increase the time available for taking action to 24 hours (or the time to reach 5 REM, which may be less than 24 hours, from the radiation field associated with main steam safety valves lifting concurrent with a SGTR) for the cases in which both CREACUS trains are unavailable. The modification applies to the radiation protection function only. Modify allowable end state to be Mode 4. Site specific validation is necessary to support extension to toxic gas and chemical protection functions.

Basis for Proposed Change

Operation of the CREACUS has no direct impact on CDF and LERF as analyzed in the plant's PRA. Operator radiation protection equipment is available as a partial radiation protection or chemical protection backup suits can help to control post-accident exposure.

Regardless of the CREACUS status, the plant risk during Mode 4 operation is lower than (or equivalent to) the similar Mode 5 operating state. This is based on the availability of more mitigating systems in Mode 4 to respond to an event and the additional risks associated with the transition to Mode 5 from Mode 4.

Defense-in-Depth Consideration

The CREACUS provides a protected environment from which operators can control the plant following an uncontrolled release of radioactivity, chemicals or toxic gas. The current TS requires two trains of CREACUS to be operable in Modes 1 through 4 to support the operator response to a DBA. The CREACUS is designed to ensure that the dose to the operators following a DBA is < 5 REM. To limit risk of exposure to the operator, the extended AOT should not be implemented when steam generator leakage is [> 30 gpd], or when additional steam generator tube leakage monitoring is required.

Operability in Mode 5 may also be required at some plants for chemical and toxic gas concerns. The CREACUS is needed to protect the control room operators in a wide variety of circumstances. Long-term plant operation with a degraded CREACUS should be based on placing the plant in a state that poses the lowest plant risk. The operation of CREACUS has no direct impact on CDF and LERF as analyzed in the plant's PRA. In general, plant operation in Mode 4 poses a lower operation risk of core damage than in Mode 5. Hence, sufficient Defense-in-Depth is retained when the end state is modified from Mode 5 to Mode 4.

Implementation of paragraph (a) (4) of 10 CFR 50.65 (Maintenance Rule) will assure proper plant configuration control. Other technical specifications (e.g. 3.38 {analogue), 3.39 (digital)) require the availability of equipment to identify the onset of a radiological challenge to the control room (or if applied to non-radiation atmospheric cleanup, a toxic gas release). Also, TS 3.3.12 requires the availability of alternate shutdown panels and local shutdown stations should remote actions become necessary.

Tier 2 Restrictions

None.

5.3.4 (ISTS) LCO 3.7.12 Control Room Emergency Air Temperature Control System (CREATCS)

The CREATCS provides temperature control for the control room following isolation of the control room. The CREATCS consists of two independent, redundant trains that provide cooling and heating of recirculated control room air. Each train consists of heating coils, cooling coils, instrumentation and controls to provide for control room temperature control.

Plant Applicability

Calvert Cliffs 1 & 2, Fort Calhoun, Palisades, PVNGS 1, 2 & 3, Waterford 3 and ANO 2.

(Note: Cooling for St. Lucie Units 1 & 2 is included in the air cleanup system discussed in TS 3.7.11, however, the cooling system discussions contained in this section apply to St. Lucie Units 1 & 2.)

Limiting Condition for Operation (LCO)

Two CREATCS trains shall be OPERABLE in Modes 1, 2, 3 and 4, and during movement of irradiated fuel assemblies.

Licensing Basis for LCO

CREATCS is required to ensure continued control room habitability and ensure that the control room temperature will not exceed equipment operability requirements following the isolation of the CR for a period of at least 30 days.

Condition Requiring Entry into a Shutdown Action Statement

Both CREATCS trains inoperable in Modes 1, 2, 3 or 4 for reasons other than on an inoperable boundary.

Proposed Modification Actions

Modify Condition E to allow 24 hours to restore one CREATCS train to operable status before requiring a plant shutdown. Modify the end state to Mode 4 if the action is not met.

Basis for Proposed Change

A 24-hour AOT is based on limiting the containment challenge probability to 1.0E-6 (see Section 4.4). The operation of CREATCS has no direct impact on ICCDP and ICLERP. Regardless of the system status, the risk of Mode 4 is lower than, or equivalent to, the similar Mode 5 operating state (see Reference 4), since more mitigating systems are available in Mode 4 to respond to an event and there are additional risks associated with the transition to Mode 5 from Mode 4.

Defense-in-Depth Consideration

The CREATCS provides a protected environment from which operators can control the plant following an uncontrolled release of radioactivity, chemicals or toxic gas. The CREATCS is needed to protect the CR in a wide variety of circumstances. The current TS requires operability of two trains of CREATCS from Mode 1 through 4 to support operator response to a DBA. An extension of the AOT is based on the low risk of system inoperability, compared to the associated risks of plant shutdown. In addition, several short term actions associated with cooling the control room may be implemented to further mitigate the risk consequences.

The CREATCS is needed to protect the CR in a wide variety of circumstances. If the CREATCS cannot be restored to operable status should be transitioned to a low risk mode. Mode 4 provides the greatest redundancy and diversity in core heat removal equipment and therefore provides an acceptable end state for this condition. Hence, sufficient Defense-in-Depth is retained when the end state is modified from Mode 5 to Mode 4.

As part of plant maintenance activities, administrative actions should be take to ensure plant staff is aware of the system inoperability and that respiratory units and CR pressurization systems are available and operational and that leakage pathways are properly controlled. Compensatory measures may include temporary cooling which may also be established via use of portable fans, propping open the doors, or similar actions. Also, the availability of alternate shutdown panels and local shutdown stations should be ensured.

Tier 2 Restrictions

None.

5.3.5 (ISTS) LCO 3.7.13 ECCS Pump Room Exhaust Air Clean Up System (PREACS)

The ECCS PREACS is an emergency system that filters air from the area of the active Engineered Safety Feature (ESF) components during the recirculation phase of a LOCA. The ECCS PREACS consists of two independent, redundant trains of equipment that provide filtering of air in the ECCS pump rooms during post LOCA recirculation cooling.

Plant Applicability

Calvert Cliffs 1 & 2, St Lucie 1 & 2, Waterford 3 [At Waterford 3 the functions of the ECCS PREACS and Penetration Room Exhaust Air Cleanup System is combined within the Controlled Ventilation Area (CVAS) Technical Specification.]

Limiting Condition for Operation (LCO)

Two ECCS PREACS trains shall be operable (default entry into LCO 3.0.3, if two trains are inoperable for reasons other than an inoperable boundary).

Licensing Basis for LCO

ECCS PREACS is typically credited in evaluating the ability of the plant to meet 10 CFR 100 and Appendix A GDC-19 radiation dose limits.

Condition Requiring Entry into a Shutdown Action

Both ECCS PREACS trains inoperable

Proposed Modification of End State Required Actions

Allow 24 hours to restore one train of ECCS PREACS to operable status before requiring a plant shutdown. Allow Mode 4 as final end state if the action is not met.

Basis for Proposed Change

A 24 hour AOT is based on the likelihood of repair and limiting the system challenge to < 1.0E-6 per year (See Section 4.4.1). While the ECCS pump room EACS affects the magnitude of post accident radionuclide releases, operation of ECCS pump room EACS has no direct impact on ICCDP and ICLERP as analyzed in the PSA. Regardless of the system status, the risk of Mode 4 is lower than (or equivalent to) the Mode 5 operating state since more mitigating systems are available in Mode 4 to respond to an event and there are additional risks associated with the transition to Mode 5 from Mode 4.

Since the risk of a transition to SDC and subsequent Mode 5 operation is greater than that incurred by continued operation in Mode 4, and the likelihood of a LOCA initiated from Mode 4 is low, repairing the system in Mode 4 is preferred.

Defense-in-Depth Consideration

The ECCS PREACS only impacts radiation releases to the public when ECCS recirculation is in progress. This system typically only operates in response to LOCA transients. Radiological releases are typically low since functional recirculation typically implies successful event mitigation. Extension of the AOT/CT to 24 hours provides time to restore component operability during power operation. This may potentially avert a plant shutdown and the associated transition risks.

Tier 2 Restrictions

None.

5.3.6 (ISTS) LCO 3.7.15 Penetration Room Exhaust Air Cleanup System (PREACS)

The PREACS filters air from the penetration area between the containment and the auxiliary building.

The PREACS consists of two independent, redundant trains. Each train consists of a heater, demister or prefilter, HEPA filter, activated charcoal absorber and a fan.

Plant Applicability

Calvert Cliffs 1 & 2, Waterford 3 [at Waterford 3 the functions of the ECCS PREACS and Penetration Room Exhaust Air Cleanup System (PREACS) are combined within the Controlled Ventilation Area (CVAS) Technical Specification.]

Limiting Condition for Operation (LCO)

Two PREACS trains shall be operable. (Default entry into LCO 3.0.3, if two trains are inoperable for reasons other than an inoperable boundary.)

Licensing Basis for LCO

The PREACS must be operable to ensure that the penetration room filtering capability is within the 10 CFR 100 design basis assumptions. The PREACS filters air from the penetration area between the containment and the auxiliary building.

Condition Requiring Entry into Shutdown Action

Both PREACS trains inoperable for reasons other than an inoperable boundary.

Proposed Modification to Actions

Allow 24 hours to restore one train of PREACS to operable status before requiring shutdown. Allow Mode 4 as the end state if the action is not met.

Basis for Proposed Change

A 24 hour risk-informed AOT is based on limiting the system challenge to < 1.0E-4 per year (see Section 4.4-1). While the PREACS affects the magnitude of the post accident radionuclide releases, operation of the PREACS has no direct impact on ICCDP and ICLERP as analyzed in the PRA. Regardless of the system status, the risk of Mode 4 is lower than (or equivalent to) the similar Mode 5 end state, since more mitigating systems are available in Mode 4 to respond to an event and there is additional risk associated with the transition to Mode 5 from Mode 4.

Since the risk of a transition to SDC and subsequent Mode 5 operation is greater than that incurred by continued operation in Mode 4, repairing the system while in Mode 4 is preferred.

Defense-in-Depth Consideration

The PREACS protects the public from radiological exposure resulting from containment leakage through penetrations. The role of the PREACS on control of large early releases is negligible. The current TS requires operability of PREACS from Modes 1 through 4. The need for the PREACS is of particular importance following a severe accident with high levels of airborne radionuclides. These events are of low probability (for example, for Mode 1, the plant core damage frequency is on the order of 2.0E-5 to 1.0E-4 per year).

Tier 2 Restrictions

None.

5.3.7 (ISTS) LCO 3.6.6 Containment Spray System & LCO 3.6.6.1 Containment Sprays/ Coolers

Containment Cooling Systems provide containment heat removal following accidents that release high energy steam to the containment. For most CE designed NSSSs, containment sprays represent a portion of a diverse and redundant heat removal system. In addition to containment heat removal, containment sprays enhance post accident fission product removal.

Plant Applicability

All.

Limiting Conditions for Operation (LCO)

See Table 5.2.3-1.

Licensing Basis for LCO

The Standard Technical Specifications (STS) requirements of NUREG-1432 distinguish between containment spray systems that are credited in containment iodine removal and containment spray systems that are not credited in containment iodine removal (ISTS 3.6.6A and 3.6.6B). The required actions for restoring inoperable containment spray systems that are not credited for iodine removal are less stringent than the requirements for containment spray systems that are credited for iodine removal.

Both spray and coolers are credited for containment pressure/temperature (P/T) control following a large LOCA or MSLB, assuming a Loss of Offsite Power (LOOP) and worst single failure. (MSLB is often the limiting accident for containment P/T control.) Depending on the plant design, the unavailability of the containment spray system will compromise the ability of the containment to respond to a containment pressure challenge and to maintain sump subcooling. The inability to maintain subcooling will prevent ECCS recirculation cooling. For plants with diverse and redundant containment heat removal capability, consisting of both Containment Air Coolers (CACs) and Containment Spray (CS), the availability of the CACs* will compensate for the unavailability of the CS system. Containment Spray also can have the additional function of removing fission products from the post-LOCA atmosphere, in which case the loss of both trains would result in a loss of fission product scrubbing capability.

Some plants include dedicated Iodine Cleanup Systems (ICS) consisting of recirculation filter units. These units are separately discussed in Section 5.3.1.

Condition Requiring Entry into a Shutdown Action Statement

Inoperability of both Containment Spray trains.

Proposed Modification for to Actions

Increase the time available to initiate shutdown to 72 hours when the Containment Spray system is inoperable and at least one train of CACs is operable.

* Also known as Containment Air Recirculation Coolers (CARCs)

Increase the time available to initiate shutdown to 12 hours when the CS system is inoperable and both trains of CAC are inoperable for containment heat removal. See Tables 5.2.3-2 and 5.2.3-3 for details.

Basis for Proposed Change

The design basis of the CS and CAC systems varies among the CE designed NSSSs. The plant design bases for many CE designed NSSSs require CS and CAC systems for containment pressure and temperature control and one of the two systems for radioactive removal. Best estimate analyses performed by a CE designed PWR indicate that one train of CAC is sufficient to effect containment pressure control. The Palo Verde units are designed with only the CS system (containing full capacity redundant CS pumps) which it credits for both functions.

For CE designed NSSSs with diverse containment heat removal capability (employing both CACs and CSs), the unavailability of the CS system poses a negligible plant risk.

Containment Spray and CAC are used to support long-term containment heat removal. This heat removal is needed to ensure that the ECCS recirculation mode can continue to effectively remove decay heat. Containment analyses performed for San Onofre indicates that successful containment heat removal occurs when at least one CS train or one CAC operates. Consequently, a minimum containment heat removal capability is required to ensure both long term containment integrity and core damage prevention. Containment Spray and CAC are also considered in the PSA Level 2 model.

The design of each of the Palo Verde Units relies entirely on the CS system for both containment heat removal and post accident iodine removal. Therefore, the unavailability of the CS system will compromise both post-accident containment integrity and ECCS recirculation cooling. Since ECCS recirculation cooling will be compromised thus leading to the inoperability of the HPSI pumps, it is proposed that a condition be added to the Palo Verde Unit TS. For the Palo Verde Units, CCDP increments will be acceptable when the AOT is limited to less than 12 hours. This limitation is also applicable to other CE designed NSSSs under the condition that all containment heat removal systems are inoperable.

Risk-Informed Assessment

A generic risk-informed AOT assessment was performed qualitatively by assuming that a loss of CS (in the presence of a fully operational CAC system) will have a negligible impact on any core damage prevention on mitigation function and would not impact post-accident containment pressure control. These conclusions were demonstrated by SONGS Units 2 & 3 specific analyses.

For the loss of two CS trains, the complete PSA model was re-solved assuming that both containment spray trains were unavailable. The results show an annual CDF of 7.09E-5 (vs. 6.68E-5 for the normal case). Over a 24-hour period, this results in an increase in core damage probability of 1.1E-8, which is acceptably low. With the CS trains out of service, LERF shows an annual frequency of 5.58E-7 (vs. the normal result of 4.96E-7). Over a 24-hour period the increased large early release probability is 1.7E-10. Again, this is an acceptably small increase.

For loss of three CS/CAC trains, the complete PSA model was re-solved, assuming both CS trains and one CAC train was unavailable. The annual CDF for this case was 1.77E-4, which results in a 24-hour increase in core damage probability of 3.0E-7. For LERF, the calculated frequency was 6.85E-7. This results in an increase in the LERP over the 24-hour period of 5.2E-10. Both of these risk increases are acceptably small.

Based on representative plant analyses performed in support of PSA containment success criteria, containment integrity may be established via use of a single fan cooler as documented in the SONGS 2 & 3 Individual Plant Examination. Qualitatively, similar conclusions could be drawn for one train of CS. Consequently, in Mode 4 one train of CAC or one train of CS assures adequate heat removal capability. Furthermore, for plants that credit CS for iodine removal by containment spray, accidents initiated in Mode 4 may be adequately supported via one OPERABLE spray pump.

For the case of CACs and CCSs unavailable, Table 4.1-2 indicates a CDP impact of 7.5E-7 for a 12 hour unavailability. ICLERP impacts will also be acceptable since these systems have a limited impact on prevention of early containment failures. A 12 hour AOT provides a sufficiently low risk impact from the perspective of late containment failure as well.

Defense-in-Depth Consideration

The inoperability of the CS or CACs will degrade the capability of the plant to respond to a containment challenge. However, provided the other system is available, the plant remains capable of controlling containment pressure. Loss of sprays will expose some plant equipment to beyond environmental qualification temperature limits should a main steam line break occur (~ 2.0E-5 per week). However, the ability of the plant to cope with the event is not compromised.

Tier 2 Restrictions

None.

Table 5.2.3-1: Summary of Conditions Leading to 3.0.3 Entry for a Representative PWR (Containment Cooling)

Plant	Inoperability	Action [1]
San Onofre 2&3	2 CS trains or 3 or more CS/CC trains.	Explicit 3.0.3 entry
Arkansas 2	2 CS trains or 3 or more CS/CC trains.	Default 3.0.3 entry
Calvert Cliffs 1 & 2	3 or more CS/CC trains unavailable	Explicit 3.0.3 entry
Fort Calhoun Station	All 3 CS pumps inoperable All 3 containment fan coolers inoperable	Implicit 2.0.1 entry (3.0.3 equivalent)
Palisades	<100% of required post accident containment cooling capability	Explicit 3.03 entry
Waterford 3	2 CS trains inoperable	Default 3.0.3 entry
Millstone 2	2 CS trains inoperable	Explicit 3.0.3 entry

Note 1 Default and implicit actions result in 3.0.3 or equivalent entry.

Table 5.2.3-2: Proposed Modifications to the Actions of ISTS 3.6.6A

Condition	Required Actions	Completion Time
1 containment spray train inoperable	Restore train to operable status	72 hours* (existing)
2 containment spray trains inoperable	Restore at least one train to operable status	72 hours (proposed)
1 containment cooling train inoperable	Restore train to operable status	7 days (existing)
2 containment cooling trains inoperable	Restore at least one train to operable status	72 hours (existing)
2 containment spray trains inoperable and 2 containment coolers inoperable	Restore at least one train of containment spray to operable status OR Restore at least one train of containment cooler to operable status	12 hours (proposed)
Required Action and associated Completion Time not met.	Be in MODE 3 Be in MODE 5	6 hours 36 hours

* This Completion Time should be extended to 7 days based on similarity to 3.6.6B and the risk-based Completion Time for two trains inoperable. This extension will be addressed in the generic change to the ISTS, but is not justified in this document.

6.0 SUMMARY

This report justifies modifications to various Technical Specification (TS) Action Statements for the conditions that results in a loss of function related to a system or component included within the scope of the plant technical specifications. It is recommended that the current required action be changed from either a default or explicit 3.0.3 entry (or equivalent action) to a risk-informed action based on the system's risk significance. In most instances, this AOT/CT is recommended to be 24 hours. In specific instances, recommendations for longer and shorter AOTs/CTs are made, as appropriate.

The proposed TS changes covered in this report are summarized in Table 2-1. These changes are risk-informed and are in conformance with RG 1.174, resulting in very small changes in CDF and LERF. Furthermore, the bounding assessments of several of the recommended AOT extensions meet the risk guideline value for RG 1.177. In some instances small potential risk increments are recommended where extension of the AOT could potentially allow minor repairs or support a more thorough condition evaluation and avert risks associated with a plant shutdown. It should be noted that risk assessments performed to support these modifications are based on bounding analyses and are applicable to the entire fleet of CE NSSS designs operated in the United States. Risks associated with the implementation of these TS changes will be managed in accordance with paragraph (a)(4) of 10 CFR 50.65 (Maintenance Rule).

The benefit from these changes is that the proposed AOT extensions provide needed flexibility in the performance of corrective maintenance of these components during power operation. These actions will avert the costs and risks associated with plant shutdowns and ensure that the public health and safety is preserved.

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7.0 REFERENCES

1. RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant Specific Changes to the Licensing Basis," USNRC, July 1998.
2. RG 1.177, "An Approach for Plant Specific Risk-Informed Decision Making: Technical Specifications," USNRC, August 1998.
3. NUREG-1432, "Standard Technical Specifications, Combustion Engineering Plants," Revision 2, April 2001, USNRC.
4. CE NPSD-1186, "Technical Justification for the Risk-Informed Modification to Selected Required Action End States for CEOG PWRs," CE Owner's Group, April 2000.
5. NUREG-1570, "Risk Assessment of Severe Accident Induced Steam Generator Tube Rupture," USNRC, March 1998.
6. NUREG/CR-6338, "Resolution of Direct Containment Heating Issue for all Westinghouse Plants with Large Dry Containments or Subatmospheric Containments," Pilch, M.M., et. al., Sandia National Laboratories, January 1996.
7. NUREG/CR-5750, "Rates of Initiating Events at U.S. Nuclear Power Plants: 1987 - 1985," USNRC, February 1999.
8. CE NPSD-1045-A, "Joint Application Report: Modifications to Containment Spray System, and the Low Pressure Safety Injection System Technical Specifications," CE Owner's Group, March 1998.
9. 10 CFR 100, "Reactor Site Criteria," USNRC, 1991.
10. 10 CFR 50, "Appendix A: General Design Criteria 79," USNRC, 1991.
11. TID 14844, "Calculation of Distance Factors for Power Reactor Sites," USAEC, 1962.
12. Regulatory Guide 1.4, Revision 2, "Assumptions used for Evaluating the Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors," USNRC, June 1974.
13. NUREG-1465, "Accident Source Terms for Light Water Reactors," February 1995.
14. CEN-259, "An Evaluation of the Natural Circulation Cooldown Test Performed at the San Onofre Nuclear Generating Station: Compliance with the Testing Requirements of Branch Technical Position RSB 5-1," Combustion Engineering, January 1984.
15. SONGS Units 2 Technical Specifications, Amendment No. 127, 9/9/1999 (page B.3.6-28).
16. NUREG-0212, Revision 3, "Standard Technical Specifications for Combustion Engineering Pressurized Water Reactors," July 9, 1982.
17. Not used.
18. CE NPSD-1168-A, "Joint Application Report: Modifications to CIV Technical Specification," Westinghouse Electric Company, September 2000.
19. NUREG/CR-5500, "Reliability Study: Auxiliary/Emergency Feedwater System, 1987-1995," USNRC, August 1998.

20. Arkansas Nuclear One – Unit 2 Letter, 2CAN030003, "Proposed License Change For Cycle 14 Risk-Informed Operation," from Craig Anderson (ANO-2) to U.S. Nuclear Regulatory Commission, dated March 9, 2000.
21. Westinghouse Letter, LTR-ESI-02-39, "Response to NRC RAIs Concerning Topical Report CE NPSD-1208, Justification of Risk-Informed Modifications to Selected Technical Specifications for Conditions Leading to Exigent Plant Shutdown" (TSTF-360), dated February 20, 2002.

APPENDIX A

Technical Specification Cross-Reference

This information is a condensed version of the plant TS information and is provided for convenience only. For the current plant-specific TS wording, the reader should consult the actual plant TS.

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Table A-1 [1]
Results of Selected Technical Specification Review: Summary of 3.0.3 End States

ISTS		SONGS TS #	Title	End State										
Analog	Digital			ISTS	SONGS	ANO	Calvert Cliffs	Palo Verde	SL-1	SL-2	WSES	FCS[3]	PAL	MP2
3.1 Reactivity Control System														
None	None	3.1.9 (Mode 1-4)	Boration Systems - Operating	NA	**Default 3.0.3**	[5] **Implicit 3.0.3** (RWST)	NA	NA	**Default 3.0.3 (3 of 3 inop.)** ---------- [6]	**Default 3.0.3 (3 of 3 inop.)** ---------- [7]	**Default 3.0.3** (2 of 2) -------- Mode 3 in 78 hrs, then Mode 5 in 8.25 days (1 of 2)	Mode 3 in 6, (2 of 2 inop.) .... Mode 3 in 78 hrs, then Mode 5 in 8.25 days (1 of 2 inop.)	NA	**Restore 1 in 48 hrs or Mode 3 & borated in 2 hrs, then 7 days to restore 1 or Mode 5 in 36 hrs**
3.4 Reactor Coolant System														
3.4.9 (Mode 1-3)	3.4.9 (Mode 1-3)	3.4.9	Pressurizer - Heaters	Default 3.0.3	Default 3.0.3	3.4.4	Default 3.0.3	Default 3.0.3	(Mode 1-2) **Mode 4 in 6 hrs**	NA	Restore in 72 hrs or in Mode 4 in 12 hrs	Restore in 72 hrs or Mode 3 in 12 hrs	Default 3.0.3	Mode 4 in 12 hrs

Table A-1 [1]
Results of Selected Technical Specification Review: Summary of 3.0.3 End States

ISTS		SONGS TS #	Title	End State										
Analog	**Digital**			**ISTS**	**SONGS**	**ANO**	**Calvert Cliffs**	**Palo Verde**	**SL-1**	**SL-2**	**WSES**	**FCS[3]**	**PAL**	**MP2**
3.4.11 E (Mode 1-3)	3.4.11 E (Mode 1-3)	NA	Pressurizer PORVs & Block valves	**Mode 4 in 13 hrs**	NA (no PORVs)	NA (no PORVs)	**Restore 1 in 72 hrs or Mode 3 & ≤ 365F-U1 301F-U2 in 12 hrs**	NA (no PORVs, but 4 PSVs)	NA (for PORVs) **Restore or close 1 in 1 hr, or Mode 5 in 36 hrs** (both BVs inop.)	NA (for PORVs) **Close 1 in 1 hr, or Mode 5 in 36 hrs** (both BVs open)	NA (no PORVs)	**Restore 1 in 1 hr or close both block valves & Mode 4 in 42 hrs (PORVs) Restore 1 in 2 hrs & both in 74 hrs or Mode 4 in 12 hrs (BVs)**	**Restore 1 in 2 hrs or Mode 3 in 6 hrs**	**Restore 1 in 1 hr or Mode 4 in 12 hrs (PORVs)** **Restore 1 in 2 hrs and both in 74 hrs or Mode 4 in 12 hrs (BVs)**
3.5 Emergency Core Cooling System														
3.5.1 D (Mode 1-3)	3.5.1 D (Mode 1-3)	3.5.1 D	SITs (2 or more of 4)	Explicit 3.0.3	Explicit 3.0.3	Default 3.0.3	Explicit 3.0.3	Explicit 3.0.3	Default 3.0.3	Default 3.0.3	(Mode 1-4 Default 3.0.3)	Default 2.0.1	Explicit 3.0.3	Explicit 3.0.3
3.5.2 A (Mode 1-3)	3.5.2 A (Mode 1-3)	3.5.2	HPSI (2 of 2)	Explicit 3.0.3	Default 3.0.3	Implicit 3.0.3	Restore 1 in 72 hrs or Mode 3 & Pzr < 1750 psi in 12 hrs	Implicit 3.0.3	Implicit 3.0.3	Implicit 3.0.3	Implicit 3.0.3	Default 2.0.1	Explicit 3.0.3	Implicit 3.0.3

Table A-1 [1]
Results of Selected Technical Specification Review: Summary of 3.0.3 End States

ISTS		SONGS TS #	Title	End State										
Analog	**Digital**			**ISTS**	**SONGS**	**ANO**	**Calvert Cliffs**	**Palo Verde**	**SL-1**	**SL-2**	**WSES**	**FCS**[3]	**PAL**	**MP2**
3.5.2 A (Mode 1-3)	3.5.2 A (Mode 1-3)	3.5.2 A	LPSI (2 of 2)	Restore 1 in 72 hrs or Mode 4 & Pzr <1700 in 12 hrs	Default 3.0.3	Default 3.0.3	Restore 1 in 72 hrs or Mode 3 & Pzr < 1750 psi in 12 hrs	Restore 1 in 72 hrs or Mode 3 & Pzr < 1837 psi & < 485F in 12 hrs	Default 3.0.3	Default 3.0.3	Default 3.0.3	Default 2.0.1	Explicit 3.0.3	Default 3.0.3
3.6 Containment Systems														
3.6.1 B (Mode 1-4)	3.6.1 B (Mode 1-4)	3.6.1 B	Containment	Implicit 3.0.3 — 24 hours (Tendons)	Implicit 3.0.3	Implicit 3.0.3	Implicit 3.0.3	Implicit 3.0.3	Implicit 3.0.3 — (Leak Testing)	Implicit 3.0.3 — (Leak Testing)	Implicit 3.0.3	Implicit 2.0.1	Implicit 3.0.3	Implicit 3.0.3 — (Tendons)
3.6.12 (Mode 1-4)	3.6.12 (Mode 1-4)	NA	Containment - Vacuum Relief valves (2 of 2)	Default 3.0.3	NA	NA	NA	NA	Default 3.0.3 (inop. on delta pressure)	Default 3.0.3 (inop. on absolute pressure)	Default 3.0.3 (inop. on absolute pressure)	NA	NA	NA
3.6.13 (Mode 1-4)	3.6.13 (Mode 1-4)	NA	Shield Building EACS	Default 3.0.3	NA	NA	NA	NA	Default 3.0.3 (SBVS)	Explicit 3.0.3 (SBVS)	Default 3.0.3 (SBVS)	NA	NA	Default 3.0.3

Table A-1 [1] Results of Selected Technical Specification Review: Summary of 3.0.3 End States														
ISTS		**SONGS TS #**	**Title**	**End State**										
Analog	**Digital**			**ISTS**	**SONGS**	**ANO**	**Calvert Cliffs**	**Palo Verde**	**SL-1**	**SL-2**	**WSES**	**FCS[3]**	**PAL**	**MP2**
3.6.6A (Mode 1-3 &4)	3.6.6A (Mode 1-3 &4)	3.6.6.1 D&E (Mode 1-3)	CTMT Spray and Cooling Systems (Credit Taken for Iodine Removal)	(Mode 1-4 **Explicit 3.0.3** (for 2 CS or 3 or more CS/CC) **Restore 1 in 72 hrs or Mode 5 in 36 hrs** (for both CC)	(Mode 1-3 **Explicit 3.0.3** (for 2 CS or 3 or more CS/CC) **Restore 1 in 72 hrs or Mode 4 in 36 hrs** (for both CC)	(Mode 1-3) **Default 3.0.3** (for both CS &3 or more CS/CC (Mode 1-4) **Restore 1 in 72 hrs & both in 7 days or Mode 5 in 36 hrs** (for both CC)	NA NA	NA NA	(Mode 1-3 **Explicit 3.0.3** (for both CS &3 or more CS/CC (Mode 1-3 **Mode 4 in 84 hrs** (for both CC)	(Mode 1-3 **Explicit 3.0.3** (for both CS &3 or more CS/CC (Mode 1-3 **Mode 4 in 84 hrs** (for both CC)	(Mode 1-4 **Default 3.0.3** (for both CS) **NA** (CC)	**Default 2.0.1** (for all 3 CS) **Default 2.0.1** (for all 3 CC)	**NA** **NA**	Mode 1-3 **Explicit 3.0.3** (for both CS) Mode 1-3 **Restore 1 in 48 hrs or Mode 4 in 12 hrs** (for both CC)
3.6.6B (Mode 1-3 &4)	3.6.6B (Mode 1-3 &4)	3.6.6.2 B (Mode 4 only)	CTMT [Spray and] Cooling Systems (Credit not taken for Iodine Removal)	(Mode 1-4) **Explicit 3.0.3** (for 3 or more CS/CC) **Restore 1 in 72 hrs or Mode 5 in 36 hrs** (for both CS or both CC)	(Mode 4 only) **NA** (CS) **Restore 1 in 72 hrs or Mode 5 in 36 hrs** (for both CC)	**NA** (CS) **NA** (CC)	(Mode 1-3) **Explicit 3.0.3** (for 3 or more CS/CC) **Restore 1 in 72 hrs or Mode 4 in 12 hrs** (for both CS or both CC)	(Mode 1-4) **Explicit 3.0.3** (CS) **NA** (CC)	**NA** (CS) (Mode 3 <1750 psi) **Explicit 3.0.3** (for both CC)	**NA** (CS) (Mode 3 <1750 psi) **Explicit 3.0.3** (for both CC)	**NA** (CS) Mode 1-4) **Default 3.0.3** (for both CC)	**NA** (CS) **NA** (CC)	**NA** (CS) (Mode 1-3) **Explicit 3.0.3** (for both CC)	**NA** (CS) **NA** (CC)
3.6.10 (Mode 1-4)	3.6.10 (Mode 1-4)	NA	Iodine Cleanup System	Default 3.0.3	NA	NA	Implicit 3.0.3 (IRS)	NA	(Mode 1-3 Restore in 72[8] hrs	(Mode 1-3 **Restore in 72[9] hrs**	NA	Restore in 24[10] hrs	NA	NA

Table A-1 (1) **Results of Selected Technical Specification Review: Summary of 3.0.3 End States**														
ISTS		**SONGS TS #**	**Title**	**End State**										
Analog	**Digital**			**ISTS**	**SONGS**	**ANO**	**Calvert Cliffs**	**Palo Verde**	**SL-1**	**SL-2**	**WSES**	**FCS**(3)	**PAL**	**MP2**
3.7 Plant Systems														
3.7.11 E (Mode 1-6)	3.7.11 E (Mode 1-6)	3.7.11 F	CREACUS	Explicit 3.0.3	Explicit 3.0.3	3.7.6.1 Default 3.0.3 (CREAS)	Explicit 3.0.3 (CREVS)	Explicit 3.0.3 (CREFS)	Complex Actions (CREVS)	Restore 1 in 24	Implicit 3.0.3 (CREAFS)	Explicit 2.0.1	Explicit 3.0.3 (CRV)	Implicit 3.0.3 (CREVS)
3.7.12 E (Mode 1-4)	3.7.12 E (Mode 1-4)	NA	CREATCS	Explicit 3.0.3	NA	3.7.6.1 (CREACS)	Explicit 3.0.3 (CRETS)	Explicit 3.0.3 (CREATC)	NA	NA	Implicit 3.0.3 (CRATS)	Explicit 2.0.1	Explicit 3.0.3 (CRC)	NA
3.7.13 (Mode 1-4)	3.7.13 (Mode 1-4)	NA	ECCS Pump Room EACS	Default 3.0.3	NA	NA	Default 3.0.3	Default 3.0.3 (ESF Pump Room EACS)	Default 3.0.3	Default 3.0.3	Default 3.0.3 (CVAS)	NA	NA	NA
3.7.15 (Mode 1-4)	3.7.15 (Mode 1-4)	NA	Penetration Room EACS	Default 3.0.3	NA	NA	(Mode 1-3) - Mode 4 in 12 hrs	NA	NA	NA	Default 3.0.3 (CVAS)	NA	NA	NA

Footnotes to Table A-1

(1) Default and implicit actions result in 3.0.3 entry.
(2) Not applicable to all PWR designs.
(3) Fort Calhoun end states are different:

Mode 1 = Operating (Reactor Power $\geq$ 2%)
Mode 2 = Hot Standby (Reactor Power < 2% & T_{AV} > 515 °F)
Mode 3 = Hot Shutdown (T_{AV} > 515 °F & reactor subcritical)
Mode 4 = Cold Shutdown (T_{cold} < 210 °F & RCS at shutdown boron concentration)
Mode 5 = Refueling Shutdown (T_{cold} < 210 °F & RCS at refueling boron concentration)

(4) Not used.
(5) Restore in 72 or Mode 4 in 6, then 7 days or Mode 5 in 36 hrs (Flowpaths and BAMT).
(6) Restore to 2 paths in 72 or Mode 3 in 2, then restore in 7 days or Mode 5 in 30. (2 of 3 inop.)
(7) Restore to 2 paths in 72 or Mode 3 in 6, then restore in 7 days or Mode 5 in 30 (2 of 3 inop.)
(8) Mode 4 in 6, then restore in 48 or Mode 5 in 30.
(9) Mode 4 in 6, then restore in 48 or Mode 5 in 30 (SAS).
(10) Mode 3 in 12, then restore in 48 or Mode 4 in 24 (IRS).
NA Not Applicable

Table A-2
Technical Specification Numbering Cross-Reference

ISTS		SONGS TS #	Title	Current End State								
Analog	**Digital**			**ANO**	**CC**	**Palo Verde**	**SL-1**	**SL-2**	**WSES**	**FCS**	**PAL**	**MP2**
3.1 Reactivity Control System												
None	None	3.1.9	Boration Systems - Operating	3.1.2.2 -flow path 3.1.2.8 -BAT	NA	NA	3.1.2.2 3.1.2.8	3.1.2.2 3.1.2.8	3.1.2.2	2.2.2(2)	NA	3.1.2.2 3.1.2.8b
3.4 Reactor Coolant System												
3.4.9	3.4.9	3.4.9	Pressurizer - Heaters	NA	3.4.9	3.4.9	3.4.4	3.4.3	3.4.3b	2.1.7a	3.4.9	3.4.4b
3.4.11	3.4.11	NA	Pressurizer PORVs & Block valves & RCS & Pzr Vent Valves	NA (PORV) 3.4.11B (RCS & Pzr Vent Valves)	3.4.11D (PORV) 3.4.11E (BV)	NA (PORV) 3.4.12B (RCS & Pzr Vent Valves)	NA (PORV) 3.4.12 (BV) 3.4.15 (RCS & Pzr Vent Valves)	NA (PORV) 3.4.4b (BV)	NA (PORV) 3.4.10b (RCS & Pzr Vent Valves)	2.1.6(5)	3.4.11 C, D, & E	3.4.3 C&D (PORV & BV) 3.4.11A (RCS & Pzr Vent Valves)
3.5 Emergency Core Cooling System												
3.5.1	3.5.1	3.5.1.E	SITs	3.5.1	3.5.1 D	3.5.1 D	3.5.1	3.5.1	3.5.1		3.5.1 D	3.5.1 E
3.5.2 A	3.5.2 A	3.5.2	HPSI	3.5.2	3.5.2 A	3.5.3 B	3.5.2	3.5.2	3.5.2		3.5.2.D	3.5.2
3.5.2 A	3,5,2 A	3.5.2	LPSI	3.5.2	3.5.2 A	3.5.3 B	3.5.2	3.5.2	3.5.2		3.5.2.D	3.5.2
3.6 Containment Systems												
3.6.1 B	3.6.1 B	3.6.1 B	Containment	3.6.1.1 3.6.1.5 (Tendons)	3.6.1.B	3.6.1.B	3.6.1.1 3.6.1.6 (Leak Rate)	3.6.1.1 3.6.1.6 (Leak Rate)	3.6.1.1	2.6(1)	3.6.1.B	3.6.1.1 3.6.1.6 (Tendons)
3.6.12	3.6.12	NA	Containment - Vacuum Relief Valves	NA	NA	NA	3.6.5.1	3.6.5	3.6.5	NA	NA	NA
3.6.13	3.6.13	NA	Shield Building EACS	NA	NA	NA	3.6.6.1 (SBVS)	3.6.6.1 (SBVS)	3.6.6.1 (SBVS)	NA	NA	3.6.5.1
3.6.6A	3.6.6A	3.6.6.1 D&E	CTMT Spray and Cooling Systems (Credit for Iodine Removal)	(Mode 1-3) 3.6.2.1 (CS) (Mode 1-4) 3.6.2.3b (CC)	3.6.8.C (Iodine Removal System)	NA	3.6.2.1.1 E (CS) 3.6.2.1.1 D (CC)	3.6.2.1.1 E (CS) 3.6.2.1.1 D (CC)	3.6.2.1 (CS) NA (CC)	2.4	NA	3.6.2.1.E (CS) 3.6.2.1 D (CC)

Table A-2
Technical Specification Numbering Cross-Reference

ISTS		SONGS TS #	Title	Current End State								
Analog	**Digital**			**ANO**	**CC**	**Palo Verde**	**SL-1**	**SL-2**	**WSES**	**FCS**	**PAL**	**MP2**
3.6.6B	3.6.6B	3.6.6.2 B	CTMT [Spray and] Cooling Systems [Mode 4] (Credit not taken for Iodine Removal)	NA	3.6.6.F	3.6.6.C	NA (CS) (Mode 3 <1750 psi) 3.6.2.1.2b (CC only)	NA (CS) (Mode 3 <1750 psi) 3.6.2.1.2b (CC only)	NA (CS) 3.6.2.2 (CC)	NA	3.6.6.C (CTMT Cooling)	NA
3.6.10	3.6.10	NA	Iodine Cleanup System	NA	3.6.8	NA	3.6.2.2 (SAS)	3.6.2.2 (IRS)	NA	2.4(2) (IRS)	NA	NA
3.7 Plant Systems												
3.7.11 E	3.7.11 E	3.7.11 D	CREACUS	3.7.6.1 (CREVAS)	3.7.8 G (CREVS)	3.7.11 F (CREFS)	3.7.7.1 (CREVS)	3.7.7 B (CREACS)	3.7.6.1b (CREAFS) 3.7.6.5 (CRIP)	2.12.1(3)	3.7.10 F (CRV)	3.7.6.1b (CREVS)
3.7.12 E	3.7.12 E	NA	CREATCS	NA	3.7.9 C (CRETS)	3.7.12 F	NA	NA	3.7.6.3b (CRATS)	2.12.2(3)	3.7.11 E (CRC)	NA
3.7.13	3.7.13	NA	ECCS Pump Room EACS	NA	3.7.10	3.7.13 (ESF Pump REACS)	3.7.8.1	3.7.8	3.7.7 (CVAS)	NA	NA	NA
3.7.15	3.7.15	NA	Penetration Room EACS	NA	3.7.12	NA	NA	NA	3.7.7 (CVAS)	NA	NA	NA

APPENDIX B

System Specific LER Event Trees

This appendix contains the simplified Large Early Release event trees for the systems evaluated. The values used to estimate the probability for the event tree scenarios for a normalized ICCDP are also shown.

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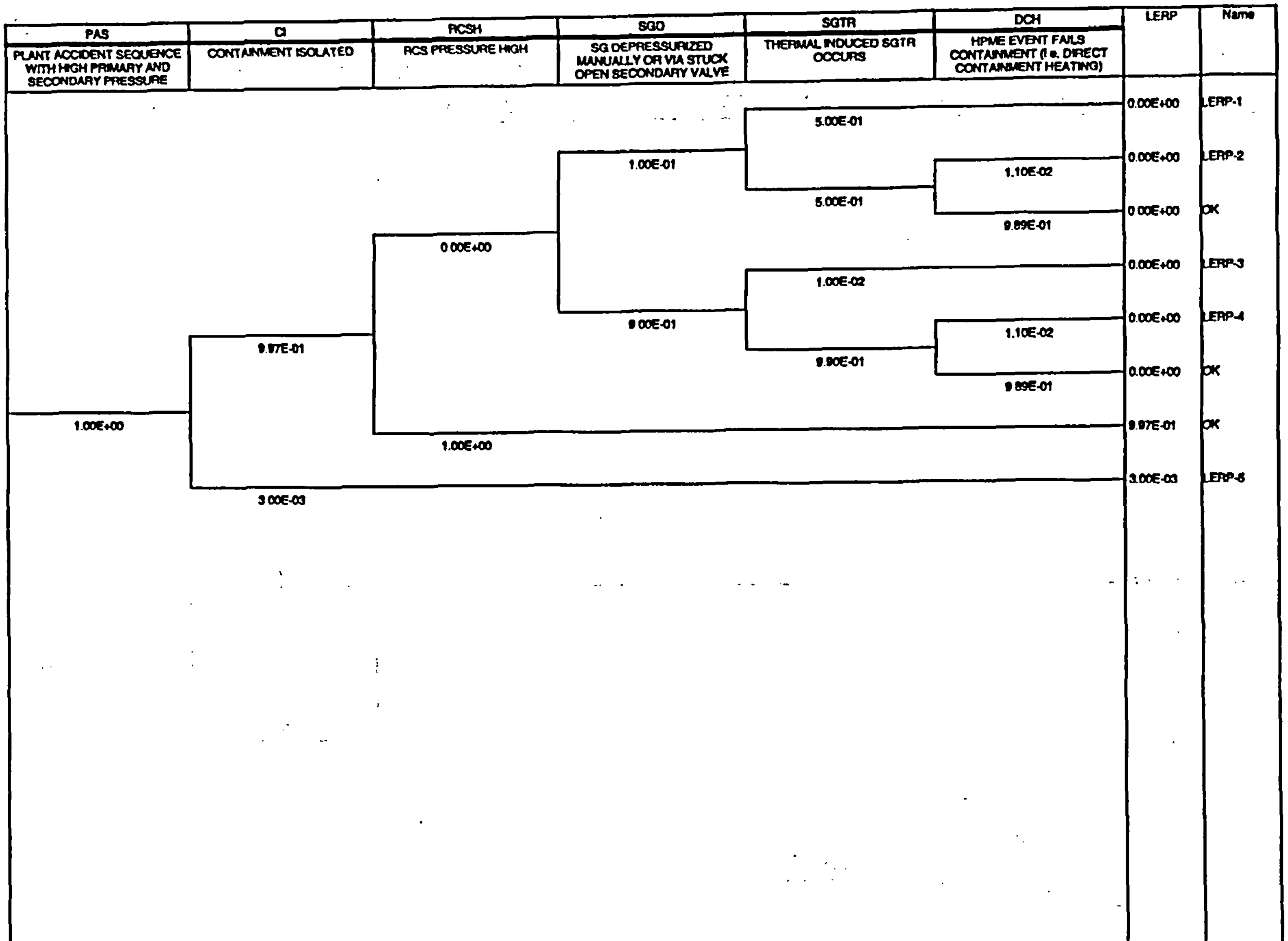
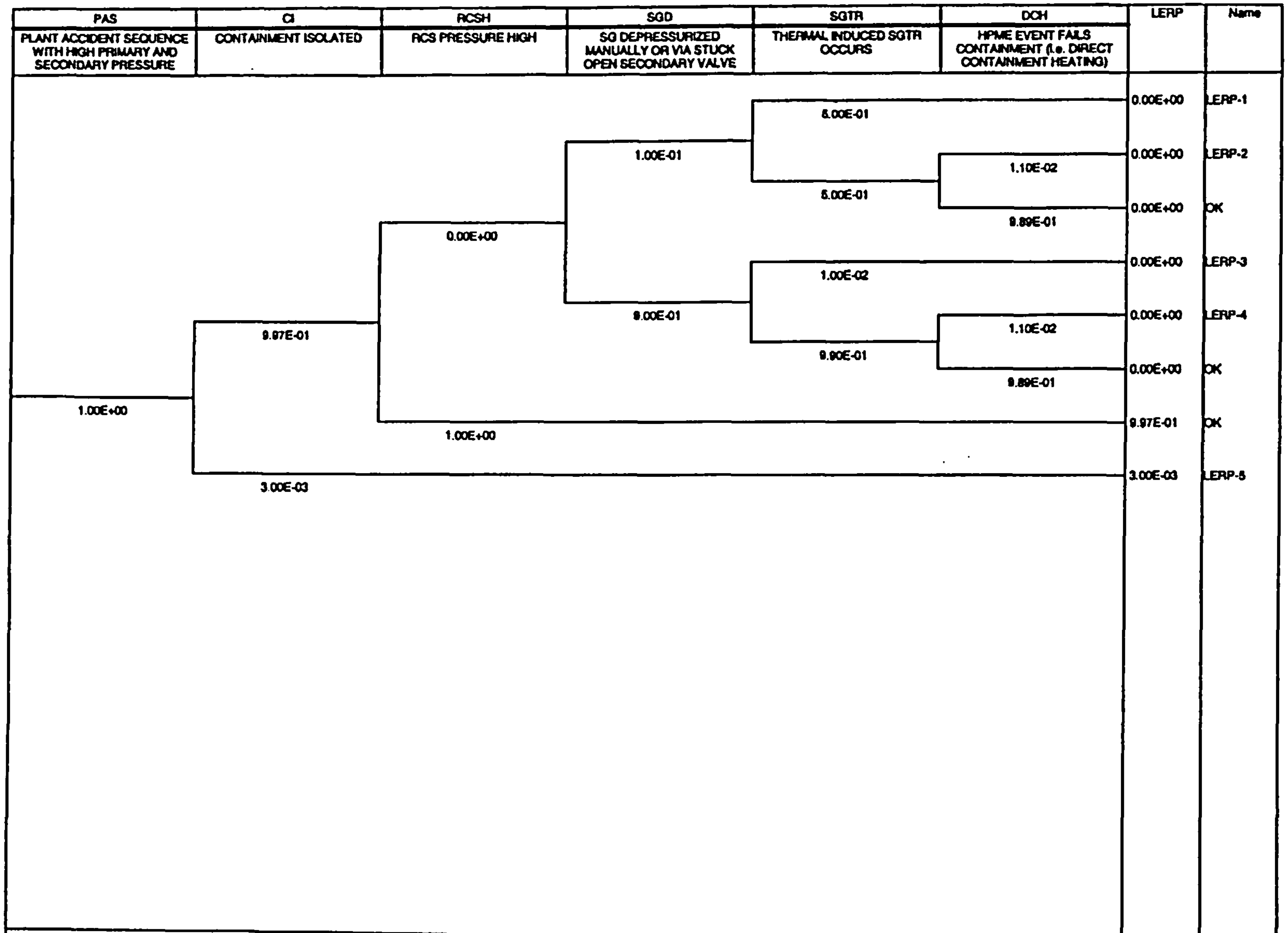
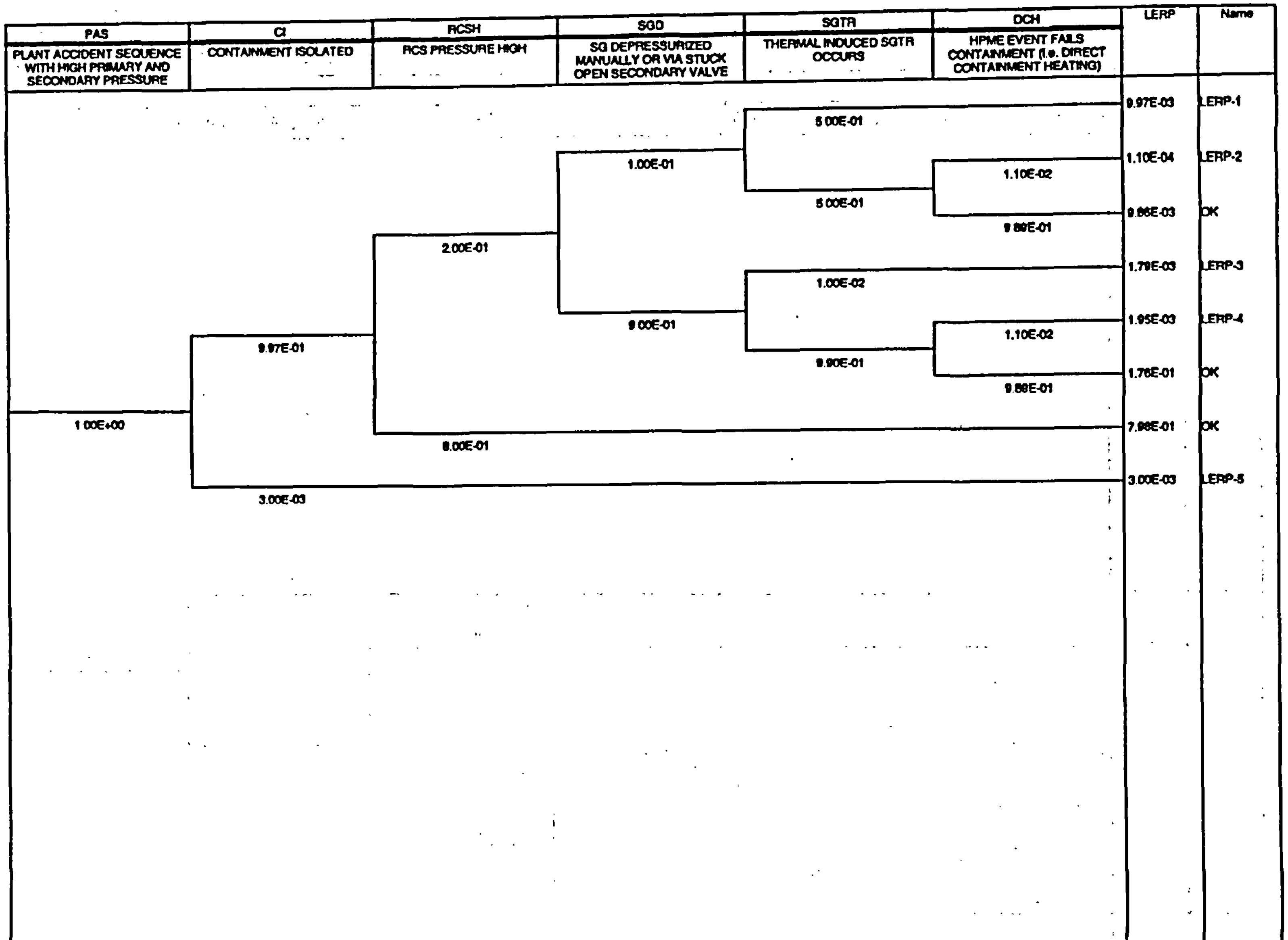


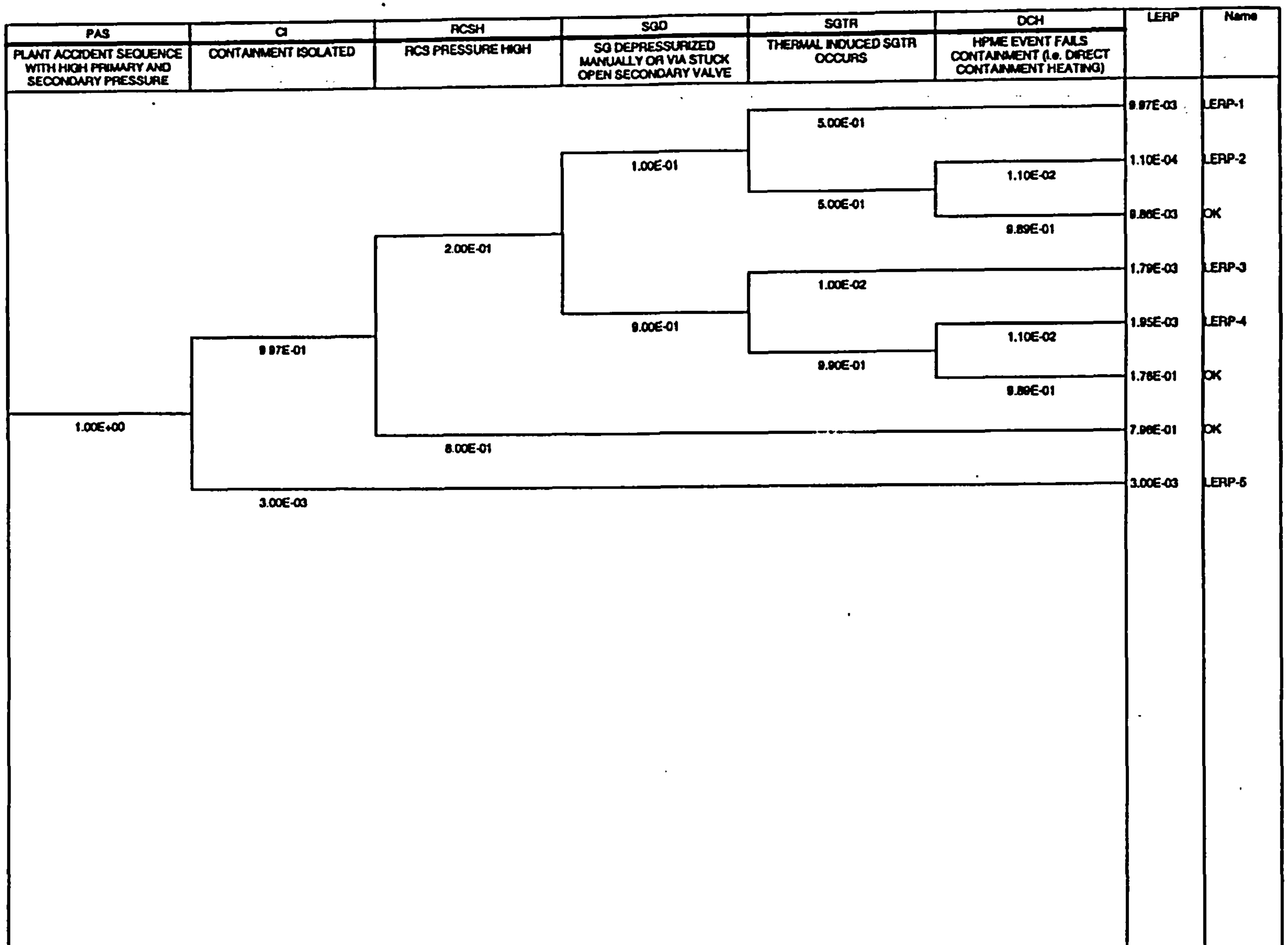
B-1: Simplified CLERP Event Tree for SIT	D:\RUPERT\303_SIT.eta	8/4/03	Page 1



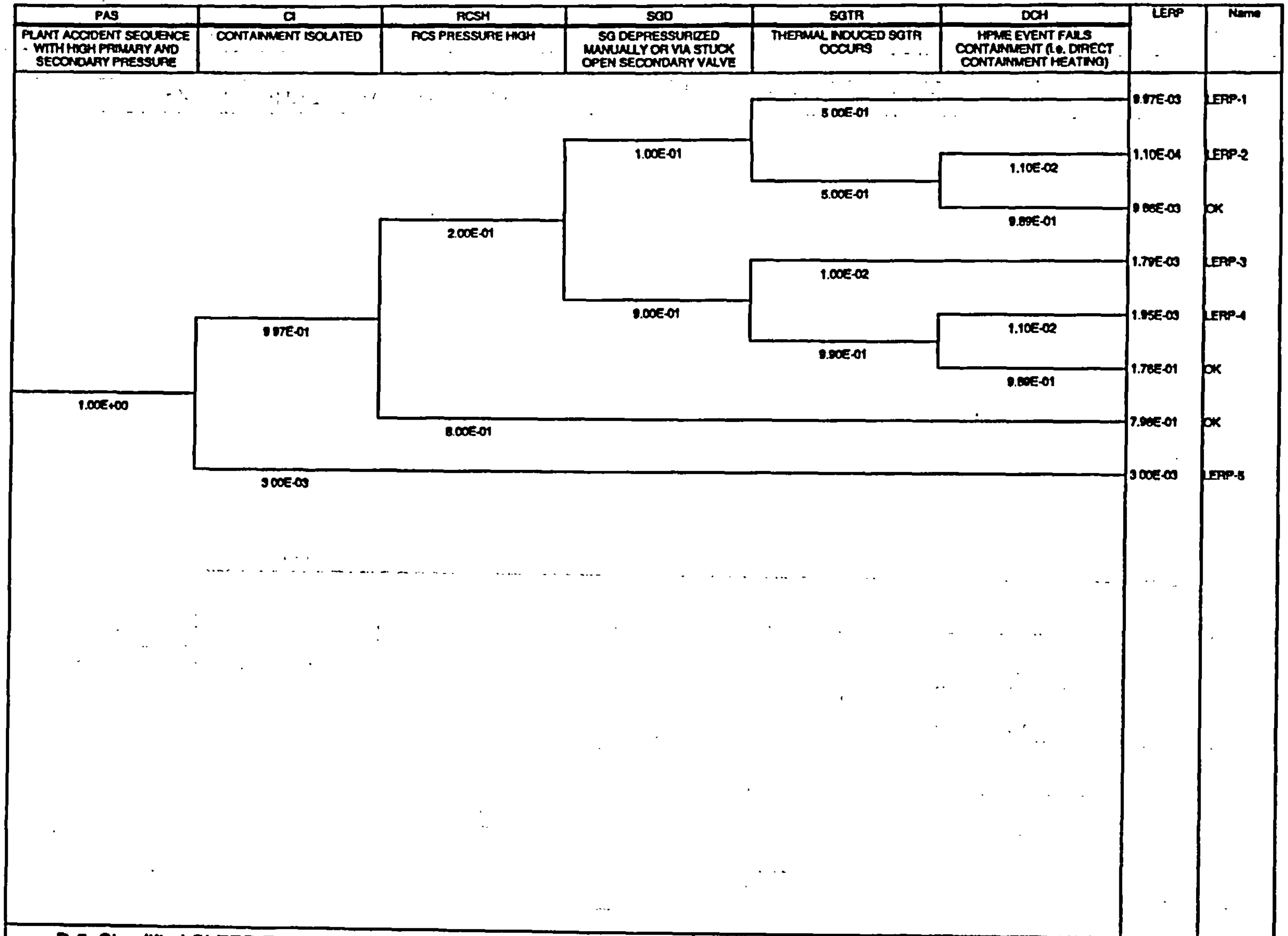
B-2: Simplified CLERP Event Tree for LPSI	D:\RUPERT\303_LPSI.eta	8/4/03	Page 1



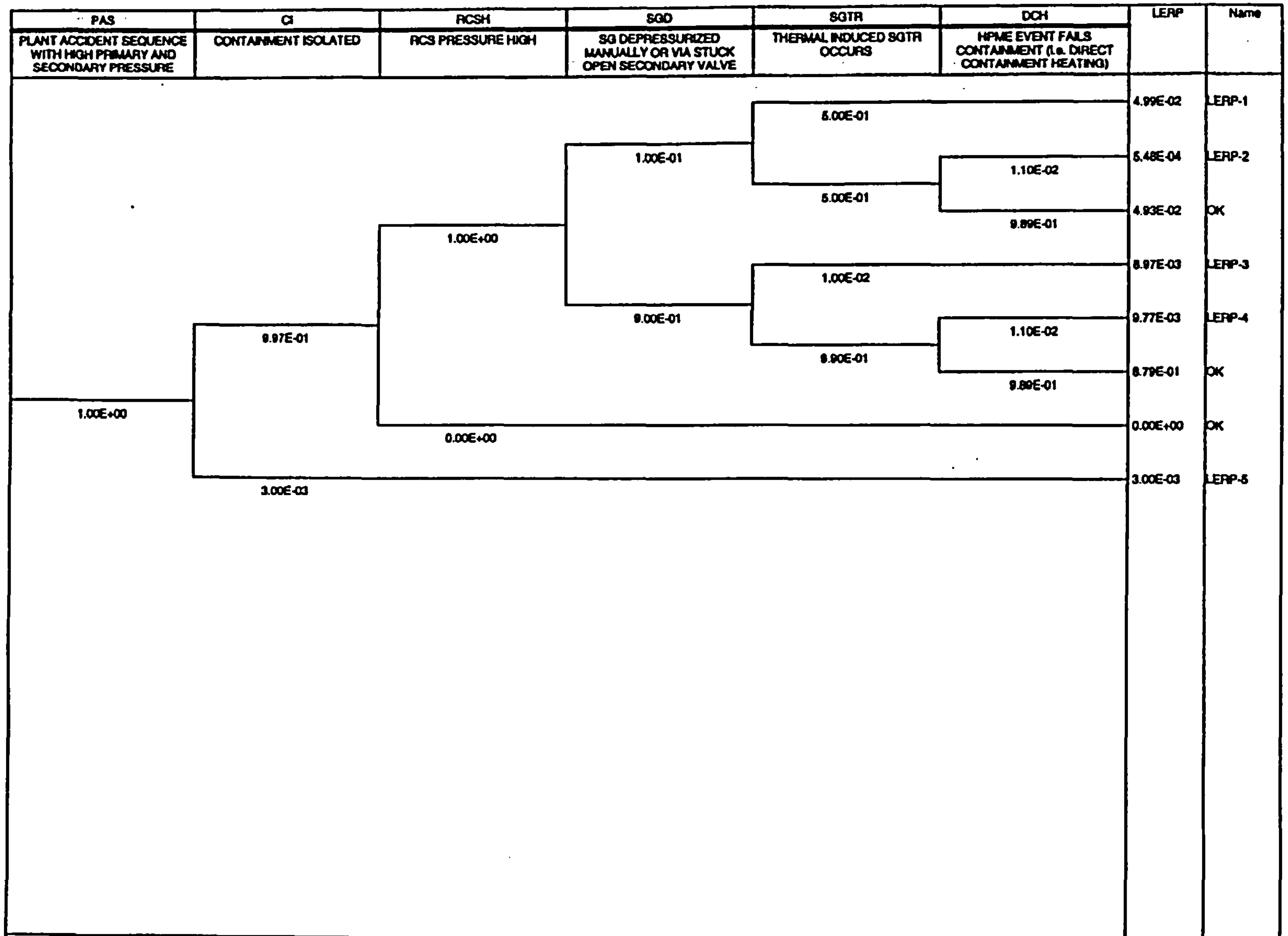
B-3: Simplified CLERP Event Tree for HPSI with PORV | D:\RUPERT\303_HPSP.eta | 8/4/03 | Page 1



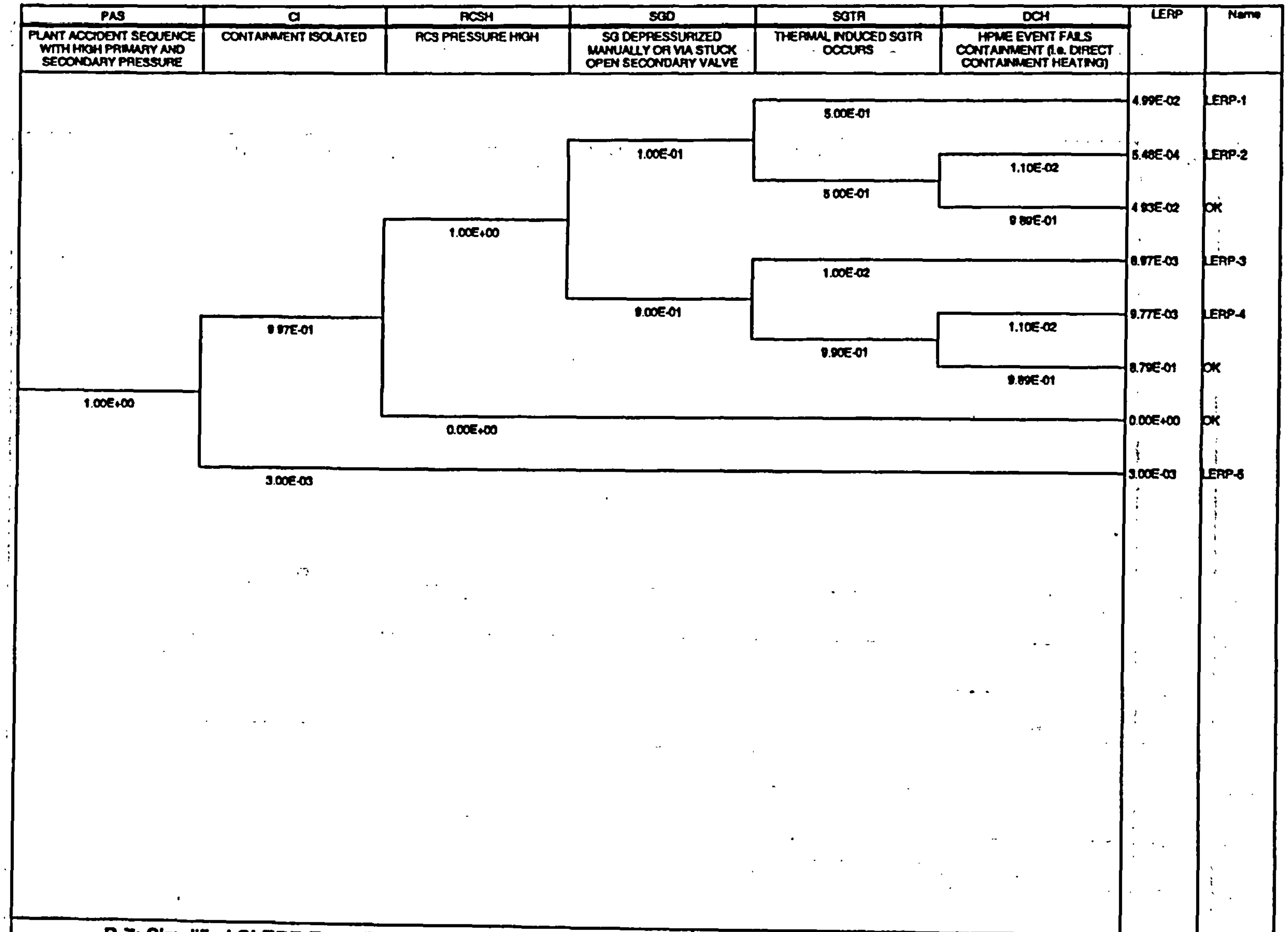
B-4: Simplified CLERP Event Tree for HPSI w/o PORV	D:\RUPERT\303_HPSI.eta	8/4/03	Page 1



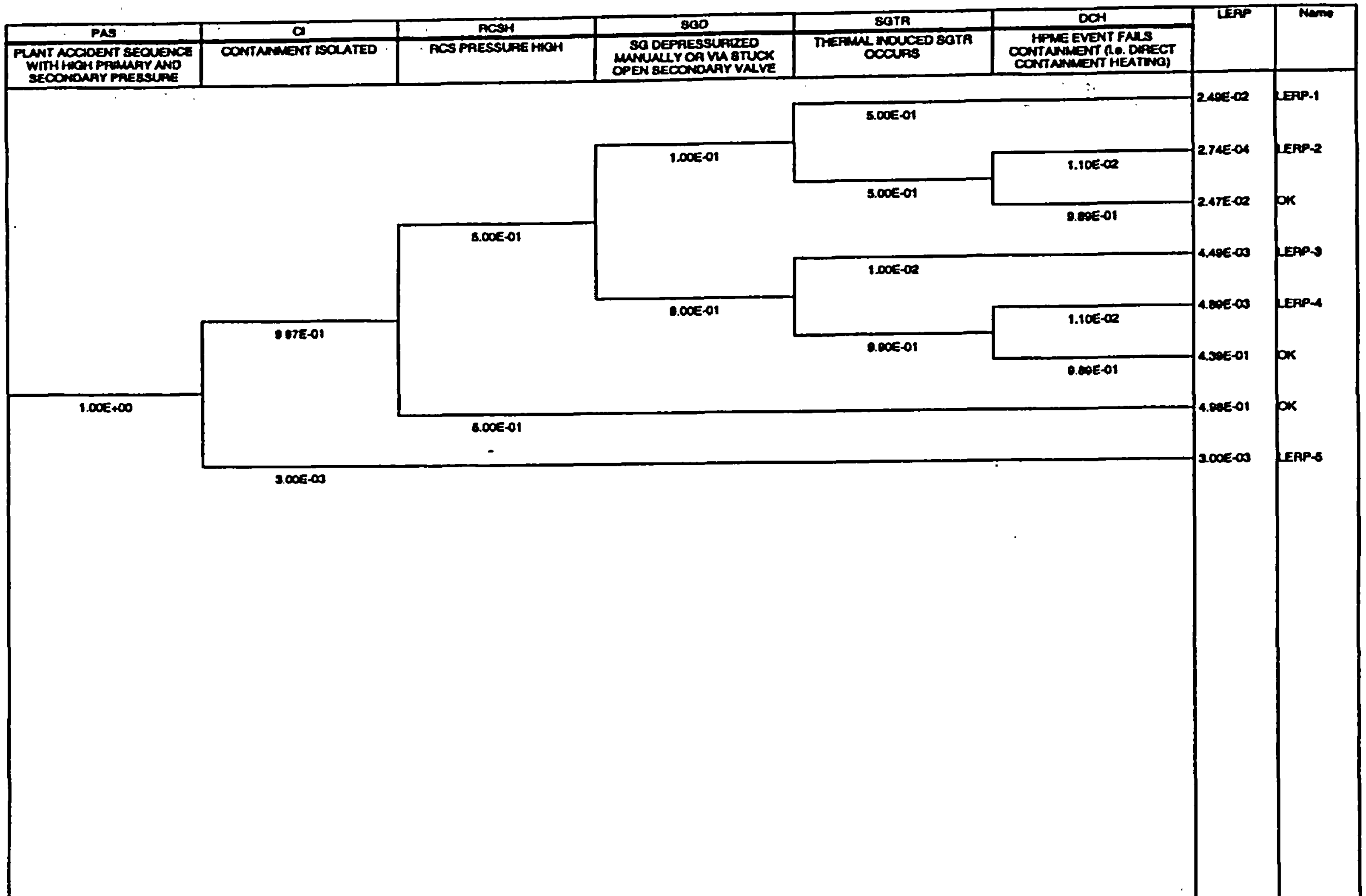
B-5: Simplified CLERP Event Tree for Containment Spray System	D:\RUPERT\303_CSS.eta	8/4/03	Page 1



B-6: Simplified CLERP Event Tree for PORV	D:\RUPERT\303_PORV.eta	8/4/03	Page 1



B-7: Simplified CLERP Event Tree for Boration Systems | D:\RUPERT\303_BOR.eta | 8/4/03 | Page 1



B-8: Simplified CLERP Event Tree for Pressurizer Heaters	D:\RUPERT\303_PZR.eta	8/4/03	Page 1

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Westinghouse Electric Company LLC
2000 Day Hill Road
Windsor, Connecticut 06095-0500