COMBUSTION ENGINEERING OWNERS GROUP

CE NPSD-1208, Rev 0

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

CEOG Task 1115

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CE Nuclear Power LLC
P.O. Box 500
Windsor, Connecticut 06095-0500

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Final Report

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Author: [signature]
Raymond E. Schneider
Probabilistic Safety Analysis

Reviewer: [signature]
Rupert A. Weston
Probabilistic Safety Analysis

Approved: RCWhipple
Richard C. Whipple
Probabilistic Safety Analysis

EXECUTIVE SUMMARY

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

Specifically, this report justifies modifications to various Technical Specification (TS) Required Action Statements for the conditions that imply a loss of function related to a system or component included within the scope of the plant TSs. 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, an extended operating period of 24 hours is recommended. In specific instances, recommendations for shorter or longer action times are made, as appropriate.

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 the entire fleet of Combustion Engineering (CE) designed Pressurized Water Reactors (PWRs) operated in the United States. Furthermore, risks associated with the implementation of these TS changes will be managed in accordance with paragraph a(4) of 10CFR50.65 (Maintenance Rule).

The benefit derived from these changes is that the proposed Allowed Outage Time (AOT) 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	-	Containment Core Damage Probability
CD	-	Core Damage
CDF	-	Core Damage Frequency
CDFP	-	Core Damage Frequency Probability
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
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
$C_{S}I$	-	Cesium Iodine
CT	-	Completion Time
CTMT	-	Containment
CVCS	-	Chemical and Volume Control System
DBA	-	Design Basis Accident
DCH	-	Direct Containment Heating
EACS	-	Emergency Air Cleanup System
EATCS	-	Emergency Air Temperature Control System
ECCS	-	Emergency Core Cooling System
ECW	-	Emergency Chilled Water
EDGs	-	Emergency Diesel Generators
ESF	-	Engineered Safety Feature
FCS	-	Fort Calhoun Station
FSAR	-	Final Safety Analysis Report

ACRONYMS CONTINUED

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
ICLERP	-	Incremental Conditional Large Early Release Probability
ICS	-	Iodine Cleanup System
IEF	-	Initiating Event Frequency
IPE	-	Individual Plant Examination
ISLOCA	-	Intersystem Loss of Coolant Accident
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
MP	-	Millstone Plant
MR	-	Maintenance Rule
MSLB	-	Main Steam Line Break
MSSV	-	Main Steam Safety Valves
MTC	-	Moderator Temperature Coefficient
NC	-	Natural Circulation
NOED	-	Notice of Enforcement Discretion
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
RCP	-	Reactor Coolant Pump

ACRONYMS CONTINUED

RCS	-	Reactor Coolant System
RGs	-	Regulatory Guides
RPS	-	Reactor Protection System
RPV	-	Reactor Pressure Vessel
RWST	-	Refueling Water Storage Tank
SAMG	-	Severe Accident Management Guidelines
SB	-	Shield Building
SBLOCA	-	Small Break Loss of Coolant Accident
SBEACS	-	Shield Building Exhaust Air Cleanup System
SBFAS	-	Shield Building Filtration Actuation Signal
SCS	-	Shutdown Cooling System
SDC	-	Shutdown Cooling
SDM	-	Shutdown Margin
SG	-	Steam Generator
SGD	-	Steam Generator Depressurized
SGHR	-	Steam Generator Heat Removal
SGTR	-	Steam Generator Tube Rupture
SIAS	-	Safety Injection Actuation Signal
SIT	-	Safety Injection Tank
SONGS	-	San Onofre Nuclear Generating Station
SGTR	-	Steam Generator Tube Rupture
SL	-	St. Lucie
SLB	-	Steam Line Break
S.O. PORV	-	Stuck Open Power Operated Relief Valve
S.O. PSV	-	Stuck Open Pressurizer Safety Valve
SR	-	Surveillance Requirement
TS	-	Technical Specification
STS	-	Standard Technical Specification
UHS	-	Ultimate Heat Sink
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 for two trains of a redundant system declared inoperable resulting in the loss of a safety function, and there is either an unspecified action for the condition (requiring a default LCO 3.0.3 entry) or conditions exist where the defined 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 both the lowest risk state and most economical option. 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 Combustion Engineering (CE) designed Pressurized Water Reactors (PWRs) (as appropriate). Where plant uniqueness results in a variation from the risk assessment, plant specific assessments are provided.

2.0 SCOPE OF PROPOSED CHANGES TO TECHNICAL SPECIFICATION

This report justifies modifications to various Technical Specification (TS) Required Action Statements for the conditions that imply a loss of function related to a system or component included within the scope of the plant TSs. 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, an extended operating period is recommended to be 24 hours. In specific instances, recommendations for shorter or longer action times are made, 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 and their associated risk impact. The Improved Standard Technical Specification (ISTS) numbering system (See Reference 3) is used for convenience. However, the technical evaluation supports these changes for all CE designed PWRs with equivalent TS numbers. Cross-comparisons of the associated TS LCOs used throughout the fleet of CE designed PWRs are presented in Appendix A.

The benefit from these changes is that the proposed AOT 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 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.

It should be noted that many of the proposed TS changes affect the existing plant shutdown requirements for plant conditions where the plant operation is not in explicit compliance with the plant design basis. 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.

Table 2-1: Summary of Risk Impacts Resulting from Proposed Modifications to Technical Specifications

ISTS #	SYSTEM	INOPERABILITY	CURRENT ACTION / AOT	PROPOSED TIME TO RESTORE ONE TRAIN (OR OPERABILITY)	PROPOSED END STATE for CONT'D INOPERABILITY (See Note 6)	INCREMENTAL CDP (See Notes 1 & 2)	INCREMENTAL LERP (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	3.3E-9
3.4.9	Pressurizer Heaters	Both Groups of Class 1E provided Heaters Inoperable	No Conditions defined. Default 3.0.3 entry	24 hrs provided plant pressure control may be successfully maintained.	Mode 4 in 12 hrs (See Note 9)	3.0E-7 (See Note 10)	1.1E-8
3.4.11	PORVs	Inability of both PORVs to Open, or Inability of both PORVs to close and block valves to be closed	Separate Condition Entry Allowed for each PORV Mode 4 In 13 hrs	24 hrs for conditions in which PORV is 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	4.9E-7	3.5E-8
3.5.1	SITs	Two or More SITs Inoperable	Explicit 3.0.3 entry	24 hrs	Mode 4 in 12 hrs (See Note 9)	1.4E-8	4.1E-11
3.5.2	LPSI (See Note 3)	Two Trains Inoperable	Defined 1 hr shutdown	24 hrs	Mode 4 in 12 hrs	1.2E-7	3.7E-10
3.5.2	HPSI	Two Trains Inoperable	Defined 1 hr shutdown	4 hrs	Unchanged	< 2.0E-6	< 3.3E-8
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)	Mode 4 in 12 hrs	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	NA	1.0E-7
3.6.1	CTMT	Inoperable	Defined 1 hr Shutdown. Mode 5 Entry in 36 hrs.	8 hrs	Mode 4 in 12 hrs	NA	1.0E-7

Table 2-1: Summary of Risk Impacts Resulting from Proposed Modifications to Technical Specifications

ISTS #	SYSTEM	INOPERABILITY	CURRENT ACTION / AOT	PROPOSED TIME TO RESTORE ONE TRAIN (OR OPERABILITY)	PROPOSED END STATE for CONT'D INOPERABILITY (See Note 6)	INCREMENTAL CDP (See Note 1)	INCREMENTAL LERP
3.6.13	SBEACS	Two Trains Inoperable	No defined action. 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	24 hrs Nuclear Hazard Only [plant specific] hrs otherwise	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	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 defined action. Default 3.0.3 entry.	24 hrs	Mode 4 in 12 hrs	NA (See Note 7)	NA (See Note 7)
3.7.15	PREACS	Two Trains Inoperable	No defined action. Default 3.0.3 entry.	24 hrs	Mode 4 in 12 hrs	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 crediting the current one hour, See Table 4.1.2).
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 4.2-1).
7 AOT based on controlling system challenge probability to < 10^{-6} (See Section 4.4).
8 End state consistent with Reference 4.
9 3.0.3 entry implies Mode 5 end state.
10 Assumes probability of plant pressure control is high. If plant trip is considered likely a controlled shutdown should be initiated.

3.0 BACKGROUND

In response to the (Nuclear Regulatory Commission (NRC's) initiative to improve plant safety by developing risk-informed TSs, the CEOG 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 modification of various TSs to define and/or modify Required Action statements to accommodate extension of the time required to initiate plant shutdown from 1 hour (e.g. TS 3.0.3) to a defined risk-informed interval varying from 4 hours to 72 hours, dependent upon the TS system/component and plant design features. In addition, a proposal is included to modify many of the action statements to allow for a Mode 4 end state when the time requirements of the action statement cannot be met.

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

(a) Avoiding unnecessary unscheduled 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 for plant operation and whose removal may increase the plant risk via creating 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 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 for this assessment is the containment.

The last category includes those components that impact the plant design basis and may affect offsite exposure following normal 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 control room, penetration room 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 RG 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 CEOG members consider the supporting analytical material contained within the document to be applicable to their respective member utilities regardless of the category 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 RG 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 failure 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 (NOED).

The criteria for the risk-informed assessment of the AOTs were selected based on RG 1.174. Regulatory Guide 1.174 indicates that plant changes which would result in an increase in Core Damage Frequency (CDF) of < 1.0E-6 per year and an increase in Large Early Release Frequency (LERF) of < 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 as TS entry is envisioned as being involuntary) an effective surrogate single entry metric is appropriate. Conservatively 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. In this evaluation more restrictive CDP/LERP guidelines were employed. They 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 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 as the requested AOT extension is consistent with either guideline.

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 AOT inoperabilities of 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 most often not 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 position was not always possible. In these cases defense-in-depth is considered via controlling the outage time for related equipment, restrict activities which may challenge these systems, small intervals, and where possible, using contingency actions to limit concurrent unavailabilities appropriately and evaluating repair activities and alternatives.

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 a risk-informed equivalent. 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

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

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 given the unavailability of Standby Mitigation Equipment

The present methodology provides a bounding generic approach for evaluating the incremental Contentment Core Damage Probability (CCDP) where possible. This approach can be implemented for evaluating the risks associated with 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 Injections (LPSIs), High Pressure Safety Injections (HPSIs) 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. In general, it is assumed that unavailability of the affected system will lead to all associated events progressing towards core damage. Detailed table notes provide additional information pertaining to the Initiating Event Frequency (IEF) assessment.

The general expression used for estimating the duration 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 = annual initiating event frequency 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.

4.1.1.1 Assessment of AOTs for Unavailability of Mitigating Systems and Components

Using the above equation, 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-6	(b)	(b)	(b)	(b)	(b)	(b)	(b)	5.0E-6
LPSI	5.0E-6	4.0E-5	(d)	(d)	(b)	(b)	(b)	NA	4.5E-5
HPSI	5.0E-6	4.0E-5	5.0E-4	7.0E-5 (e)	1.0E-3	2.5E-3 (l)	1.7E-4 (c)	NA	(h)
CS (No CARCS available)	5.0E-6	4.0E-5	5.0E-4	(j)	(j)	(j)	(j)	(j)	5.5E-4
PORV	(b)	(b)	(b)	(b)	NA	(b)	1.7E-4 (c)	8.4E-6 (f)	1.8E-4 (f)
Pressurizer Heaters	NA	NA	NA	NA	NA	NA	NA	NA	NA
Boration System	NA	NA	NA	NA	NA	NA	NA	1.7E-5 (k)	1.7E-5 (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) This is taken as the product of two events: Frequency of Loss of Main Feedwater (MFW) Event followed by a loss of all AFW. The initiating event frequency for loss of main feedwater is 8.5E-2 per year. The auxiliary feedwater failure probability is 2.0E-3. This probability is the bounding value for CE PWRs. [See Table D-6 of Reference 19.] The frequency of events Total Loss of Feedwater (TLOF) leading to Feed & Bleed becomes 1.7E-4 per year.
(d) Component may be used as a backup mitigating component, however it's 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) = 0.00007/yr.
(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 total of applicable initiating event frequencies.
(h) 4.3E-3 per year for plants with PORVs; 3.1E-3 per year for plants without PORVs.
(i) NA – Not applicable.
(j) Containment heat removal 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: $ATWS_f = I_T \times RPS = 1.4 \times 1.2E\text{-}5 = 1.68E\text{-}5$ per year (value rounded up to 1.7E-5 per year).
(l) Based on one event for the operating period considered in Reference 7.

Table 4.1-2: Time (hrs) (a) for an Unavailable System to Accumulate 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
SIT	5.0E-6	1752	24	1.37E-8	1.31E-8
LPSI	4.5E-5	195	24	1.23E-7	1.18E-7
HPSI: PWR w/ PORVs	4.3E-3	2	4	1.96E-6	1.47E-6
HPSI: PWR w/o PORVs	3.1E-3	3	4	1.42E-6	1.06E-6
CS (no CARC available)	5.5E-4	16	12	7.50E-7	6.91E-7
PORV	1.8E-4	49	24	4.93E-7	4.73E-7
Boration Systems	1.7E-5	516	24	4.66E-8	4.59E-7

Notes for Table 4.1-2

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

The above table suggests that the SITs, LPSI, PORVs and boration systems are clear candidates for having alternative system required action in the Technical Specification. A small change to the HPSI TS is also proposed. The proposed full AOT risk is greater than the nominal goal of 1.0E-6. However, the low expected utilization of this TS (~ once in a plant operating life) supports these extensions as providing a low yearly risk increase of $< .05 \times 10^{-7}$, well within the guidelines of RG 1.174. The above changes will allow time for the operating staff to resolve minor inoperabilities and hence avert the risk associated with plant shutdown.

The inability of a PORV to open can impact the outcome of 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 in the order of 1.7E-4 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 indefinite inoperability of the PORVs becomes 1.8E-4 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 PWRs sump cooling is accomplished via use of heat exchangers in the spray line. 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 for all LOCAs.

Unavailability of the boration system affects post trip cooldown and ATWS mitigation. Insertion of control rods will typically ensure reactor shutdown. The boration systems are used in controlling shutdown margin in the vent of a stack 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.

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

The pressurizer Technical Specification (3.4.9) includes requirements for 2 banks to have 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 events. 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 the TS required and non-TS required will increase the potential for plant trip. The risk associated with this component unavailability was evaluated by assuming that without pressurizer heaters, the plant pressure will be controlled manually by other means (i.e. changing and letdown, HPSI or RCS Heat Removal). The current methodology assumes that the incremental risk of unavailability of these systems is approximately:

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

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

In this case, unavailability of 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 maybe 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 availability of AFW and Emergency Diesel Generators (EDGs), 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. Therefore, if operational control can be confidently expected to avoid a plant trip, 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 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 availability of alternative mitigative systems are not credited. Overall, using the upper bound 95th percentile value for IEFs, as shown below, would increase the risk values presented in Table 4.1-2 by a factor of approximately 3 or less.

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	5.0E-3	1.1E-2

A review of the above table indicates that the more risk significant initiating events IEF error factors are on the order of 2 to 3. The impact of these uncertainties on the plant risks demonstrates that even at the upper bound IEF the proposed AOT does not introduce a significant increase in plant risk for all AOTs. This conclusion is further supported by the fact that system inoperability entries are infrequent events and that capabilities to resolve inoperabilities 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 large early release scenarios for the CE designed PWRs indicates that early releases arise as a result of one of the following class of scenarios:

1. Containment Bypass Events

   These events include interfacing system Loss of Coolant Accidents (LOCAs) and Steam Generator Tube Ruptures (SGTRs) with a concomitant 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 CEOG member 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 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 Model for ICLERP

Incremental Conditional Large Early Release Probability (ICLERP) is a measure of 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 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 CEOG member utilities 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 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, as 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 unavailability of the boron system assumed to result in a high pressure core damage sequence (RCSH = 1). Analogously, contributions to the LOCA CDP increment LOCAs are assumed to not 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 an SGD, 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 should be bounding provided SG tubes meet their design limits. Studies conducted by many researchers (See for example Reference 20), indicated 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 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 Result in Containment Failure (DCH)

Failure of the Reactor Pressure Vessel (RPV) lower head releases an energetic discharge of molten core materials into the containment. 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. The calculations for these plants 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, Units 2 & 3 (SONGS) 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 PWRs 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 of the CEOG designed 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 ICLERP model:

1. The probability of containment isolation failure used in the PSAs for the CEOG member utilities 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 resulting from 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 unavailability of 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 thermally-induced SGTRs are classified as a 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 of the CEOG designed 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 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 thermally-induced steam generator tube rupture.

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 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 thermally-induced 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.

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 4.2-1. 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 ICCDP

taken from Table 4.1-2 for the proposed AOT to arrive at the ICLERP for the proposed AOT change.

Table 4.2-1: ICLERP Estimates Due to Unavailability of Selected PWR Components

System / Component	Proposed AOT (hours)	Mean Incremental Conditional CDP per AOT (from Table 4.1-2)	CLERP 1 through 5 (from Figure 4.2-1) (Note 2)					Total CLERP	Total ICLERP 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	**1.96E-6**	1.0E-2	1.0E-4	1.8E-3	1.8E-3	3.0E-3	1.7E-2	**3.3E-8**
HPSI (plants w/o PORV)	4	**1.42E-6**	1.0-E2	1.0E-4	1.8E-3	1.8E-3	3.0-E3	1.7E-2	**2.4E-8**
CS (Note 3)	12	**7.50E-7**	1.0E-2	1.0E-4	1.8E-3	1.8E-3	3.0E-3	1.7E-2	**1.3E-8**
PORV	24	**4.93E-7**	5.0E-2	5.0E-4	9.0E-3	8.9E-3	3.0E-3	7.1E-2	**3.5E-8**
Boration Systems	24	**4.66E-8**	5.0E-2	5.0E-4	9.0E-3	8.9E-3	3.0E-3	7.1E-2	**3.3E-9**
Pressurizer Heaters	24	**3.00E-7** (Note 1)	2.5E-2	2.5E-4	4.5E-3	4.4E-3	3.0E-3	3.7E-2	**1.1E-8**

Notes for Table 4.2-1

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

4.2.4 Incremental Conditional LERP Sensitivity Studies

This section presents a sensitivity study of two key parameters in the assessment of Large Early Release Probability. These parameters are: (a) the probability that a TI SGTR will occur in advance of another RCS structural failure and (b) 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.

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

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 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 and is generally conservative. This value also reflects analytical uncertainties which result in inconsistent predictions of this phenomena. 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 thermally-induced 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 thermally-induced SGTR affect the probabilities of large early scenarios LERP-1 and LERP-2 (See Figure 4.2-1) for all of the CEOG plant groups. All of the other probabilities for the remaining large early scenarios are unaffected. The results of this sensitivity evaluation are summarized in Table 4.2-2. 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 4.2-2: Sensitivity Results for Incremental Conditional Large Early Release Probability: Thermally-Induced SGTR Probability

INOPERABLE COMPONENT	TI SGTR Probability	LERP-1	LERP-2	LERP-3	LERP-4	LERP-5	Total CLERP
Pressurizer Heaters	0.6	2.99E-2	1.99E-4	4.49E-3	4.44E-3	3.00E-3	4.20E-2
	0.5	**2.49E-2**	**2.49E-4**	**4.49E-3**	**4.44E-3**	**3.00E-3**	**3.71E-2**
	0.4	1.99E-2	2.99E-4	4.49E-3	4.44E-3	3.00E-3	3.21E-2

Notes for Table 4.2-2
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 4.2-1 indicate that the normalized LERP increases approximately linearly as the thermally-induced SGTR probability increases.

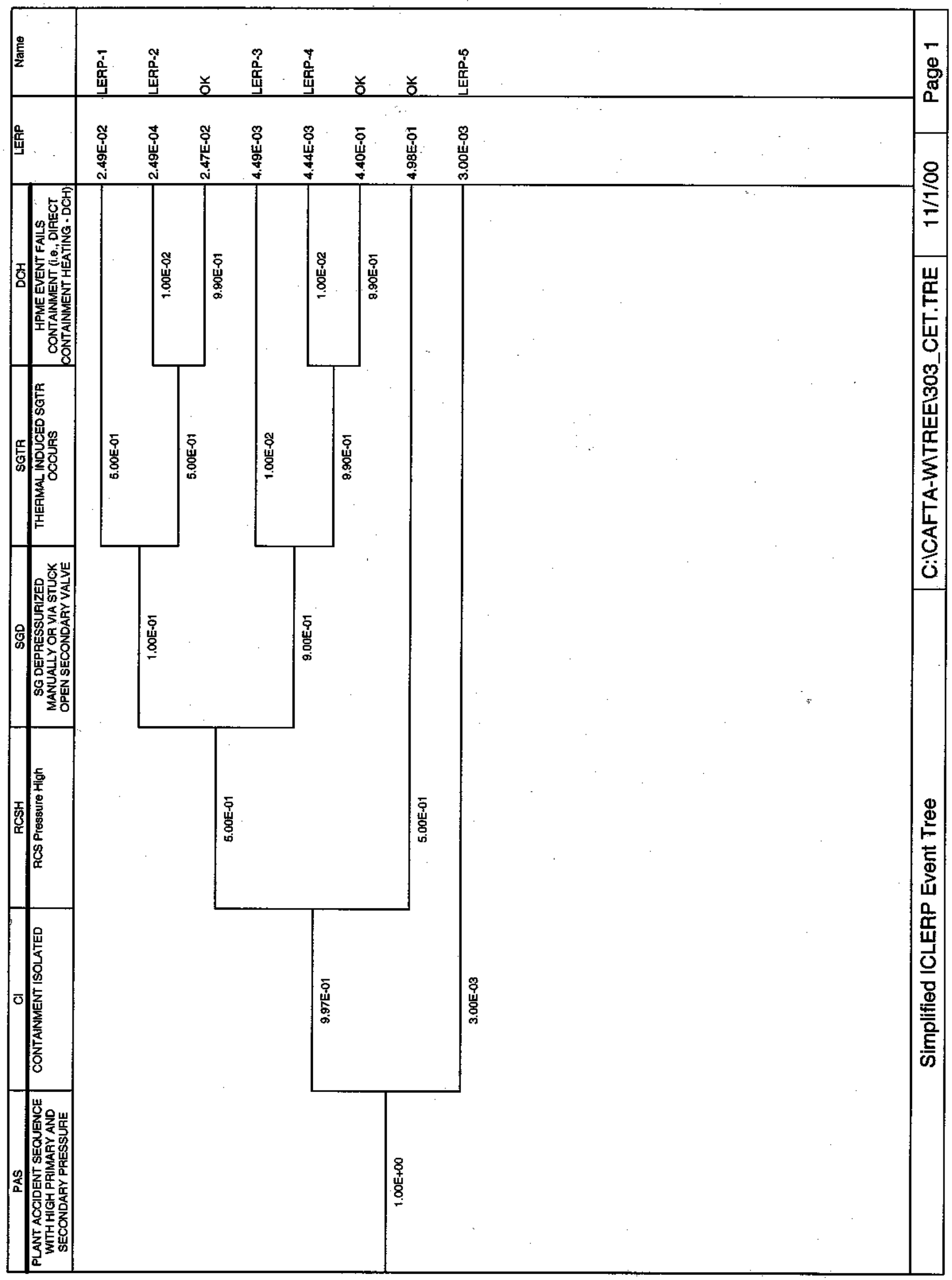


Figure 4.2-1: Simplified Incremental Large Early Release Event Tree

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

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 1 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 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 Unavailability of Pressurizer Heaters

MSSV Probability	LERP-1	LERP-2	LERP-3	LERP-4	LERP-5	Total LERP
0.050	1.25E-2	1.25E-4	4.47E-3	4.69E-3	3.00E-3	2.51E-2
0.075	1.87E-2	1.87E-2	4.61E-3	4.57E-3	3.00E-3	3.11E-2
0.100	**2.49E-2**	**2.49E-4**	**4.49E-3**	**44.4E-3**	**3.00E-3**	**3.71E-2**
0.125	3.12E-2	3.12E-4	4.36E-3	4.32E-3	3.00E-3	4.32E-2
0.150	3.74E-2	3.74E-4	4.24E-3	4.19E-3	3.00E-3	4.92E-2
0.175	4.36E-2	4.36E-4	4.11E-3	4.07E-3	3.00E-3	5.52E-2
0.200	4.98E-2	4.98E-4	3.99E-3	3.95E-3	3.00E-3	6.12E-2

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 that the impact is relatively linear.

(c) <u>Final Comments</u>

It should be noted that ICLERP values presented in Table 4.2-1 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 extension. Unavailability of HPSI will impact primarily low pressure states and result in an impact in LERP that is dominated by Intersystem Loss of Coolant Accident (ISLOCAs) 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 CLERP 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 containment capability to withstand a severe challenge.

The LERP impact of the non-functionality 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:

$$\mathrm{ICLERP} = (\mathrm{CDF}) * \frac{\Delta T}{8760}$$

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

$$\begin{aligned}\Delta T &= [\mathrm{ICLERP}_{goal} / (\mathrm{CDF})] * 8760 \\ &= 1.0\mathrm{E}{-7}/1.0\mathrm{E}{-4} \times 8760 \\ &\approx 9 \text{ hrs}\end{aligned}$$

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 extension 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 provide 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 (EACS)

- Control Room Emergency Air Temperature Control System (EATCS)
- Penetration Room Emergency Air Cleanup System (EACS)
- ECCS Penetration Room Emergency Air Cleanup System (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 all inoperable time intervals 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 interval is acceptably low (incremental system challenge of less than 1.0E-6). That is,

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

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

Using this method, the risk-informed AOTs for of 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 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 sizes. Using the nominal LOCA frequency, the resulting AOTs 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 of less than 1.0E-6

System	Recommended AOT for Unavailability of System (hrs)*	Challenge Frequency (per year)	System Challenge Probability for Extended Entry into Proposed AOT (per year)	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.
* All TS trains inoperable
# With CARC available

4.5 Transition Risk Considerations

For any given AOT extension, there is an "at power" increase in risk associated with it. 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 an PWR from its nominal full power operating state to a lower shutdown mode following equipment failure. Transition risk is of interest in understanding the tradeoff between shutting down the plant and restoring the trains to operability while the plant continues to operation. When establishing a risk decision making process consistent with the regulatory guides the risk of transitioning from "at power" to a shutdown mode can be balanced against the risk of continued operation and performing corrective maintenance while the plant is at power.

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 on Auxiliary Feedwater (AFW) have been estimated using CEOG transition risk methodology to be on the order of 1.0E-6 for an uncomplicated shutdown (See for example, Reference 8).

In addition to 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 reconfiguration of the RCS. The additional risk is dominated by inventory loss events associated with misalignment of valves during entry into SDC or an Low Temperature Overpressure Protection (LTOP) relief valve lift. These events are generally of short duration, and are important during the initial alignment of SDC. To a lesser extent, inventory loss events are possible when heating up to return to SG cooling prior to returning to power. Due to the lower decay heat at shutdown, the ICCDP associated with these events is on the order of 1.0E-6.

So 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 and acknowledging the low potential for TS entry ensures accumulated risks due to these TS modifications is negligible.

4.6 End States and Shutdown Risks

The current effort is directed towards establishing an alternate action statement for conditions where a system function is typically lost. In most of these instances the current TS either requires a Mode 5 end state or directs the operator to enter the LCO action statement for TS 3.0.3 which also leads to a Mode 5 end state.

Reference 5 discusses the risk associated with the various shutdown modes for CE designed PWRs. 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 consequence of the increased redundancy and diversity of equipment 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 AFW end state minimized plant configuration changes and associated transitional risks.

In a few instances the recommended end state is not changed (kept as Mode 5) or changed to Mode 3. Specific bases for end state recommendations is presented in TS specific discussions of Section 5.

The time recommended for Mode 3 or Mode 4 entries will be consistent with the ISTS generic philosophy. 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.

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

4.7 Maintenance Rule

Risk associated with implementation of the TS changes will be managed in accordance with provisions set forth in 10CFR50.65 paragraph a(4) and Regulatory Guide 1.182. This will assure proper plant configuration control during entry into these LCOs.

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 Shutdown Action Statement
vi) Proposed Modification to Required Actions
vii) Basis for Proposed Change
viii) Defense-in-Depth Considerations
ix) Tier 2 Restrictions

In performing the Defense-in-Depth assessment, it is assumed that the purpose of the TS Required Action to enter shutdown is to complete a short duration repair of the component under consideration. Since the TS changes being discussed 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 equipment on-line thereby potentially avoiding 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 action statements (allowed outage times/completion times and end states) it was tacitly assumed that:

- The purpose of the Required Action is to complete a short duration repair of the component under consideration.
- 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.
- Timing for end state entry is as follows:
  - Transitions from Mode 1 to Mode 3 is required to be < 6 hours.
  - Transitions from Mode 1 to Mode 4 is required to be < 12 hours.

The recommended AOT is intended to provide the operating staff additional time to resolve an inoperability while the plant remains at power. Expeditious resolution of the inoperability “at power” reduces the overall risk of plant operation. In many instances the proposed AOT alters the plant response to situations that place the plant outside of the design basis.

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. 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 risks introduced by a plant shutdown are considered. When considering plant risk, it is often risk beneficial to allow an inoperability to be resolved “at power” than to undertake 1 hour shutdown. That is, the extended AOTs, as proposed, meets the intent of the initial one hour shutdown. Furthermore, should a shutdown be required, Mode 4 would be an acceptably safe end state (See Reference 5).

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) stuck out and the decay of all xenon poison. The systems are also intended to mitigate possible return to power scenarios following an Main Steam Line Break (MSLB) and to mitigate ATWS events. The ISTS is silent on boration systems. TS 3.1.9 implies that boration systems and non-ISTS TS plants require that boration systems are available during the modes of applicability, two boration paths that are to remain available are: (1) the Refueling Water Storage Tank (RWST) and its feed to the charging pumps, and (2) one or both Boric Acid Makeup (BAMU) tanks with their respective feed paths to the charging pumps.

<u>Plant Applicability (non-ISTS plants)</u>

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

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

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

<u>Licensing Basis for LCO</u>

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 (RCP) restart. Boration systems are also necessary to ensure power reduction during an ATWS events.

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

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

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

Increase the time available to take action to restore one boration flow path 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 (CVCS) 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 extending the 3.0.3 allowed time to 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 4.7E-8, which is an acceptably small increase (See Section in 4.1). In shutdown modes, ATWS events are precluded and associated risk is negligible.

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 3.3E-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 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, availability of HPSI during this interval ensures the plant Defense in Depth is maintained. During operational periods when 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. Risk impact of boration system unavailability during this interval is low. HPSI system availability will minimize impact of an inoperable boration system for non-ATWS events.

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.

Unavailability of 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. Unavailability of pressurizer heaters during an NC cooldown will extend time to reach Shutdown Cooling System (SCS) entry conditions. However, core/RCS heat removal will be adequately established via use of SG cooling.

<u>Plant Applicability</u>

All (except ANO-2 & St Lucie-2)

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

Two groups of pressurizer heaters [capable of being powered from an emergency power supply,] operable in either Modes 1, 2 or 3.

<u>Licensing Basis for LCO</u>

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 (FSAR) 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 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 Shutdown Action Statement

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

Proposed Modification for Required Actions

Include an explicit action statement for two groups of safety related pressurizer heaters inoperable. Allow an outage time of 24 hours to restore one group of safety-related pressurizer heaters before entry into LCO 3.0.3 with Mode 4 as the final end state.

Basis for Proposed Change

Pressurizer heaters enable plant pressure to be readily controlled within its normal operating pressure band. Unavailability of these heaters would reduce the plant's ability to control the normal operating parameters and consequently increase the potential of plant trip. Therefore the risk impact may be 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 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 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 unavailability of pressurizer heaters may complicate post-trip cooldowns, successful cooldown is expected with minimal impact on plant risk due to availability of RV head and pressurizer vents.

The risk impact of pressurizer heater system inoperability is assessed assuming that 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 compromised. A review of the CE designed PWRs indicates the conditional core damage probability associated with an uncomplicated plant trip is 6.0E-6. This results in incremental CDP of 3.0E-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 pressurizer heaters a natural circulation cooldown may be required (as 20 ^{o}F subcooling may not be assumed). Such cooldowns may be conducted via use of pressurizer and RV gas vent lines, and SG venting via the Atmospheric Dump Valves (ADVs).

Tier 2 Restrictions

This extension is not applicable if equipment unavailability results in the inability to control plant pressure. When this LCO is entered, the risk will be considerably reduced if reliable plant pressure control can be maintained via backup equipment. If additional equipment failures are present that increase the likelihood of plant trip, a controlled plant shutdown should be initiated.

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 to isolate the resulting small break Loss of Coolant Accident (LOCA). 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 CEOG utilities. Several CE designed PWRs are designed without PORVs and St. Lucie 2 and Palisades operate with one or more PORVs blocked or 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	Restore 1 PORV in 72 hours.	Mode 3 in 72 hours.
Ft. Calhoun Station	Restore 1 in 1 hour or close both Block Valves (BVs)	Mode 4 in 42 hours (PORVs) Mode 4 in 72 hours (BVs)
MP2	PORVs restore 1 in 1 hour Block valves: Restore in 2 hrs	Mode 4 in 12 hours
St. Lucie 1/2	None on PORVs TS on Block Valve only	Mode 5 in 36 hours (BVs)

<u>Plant Applicability</u>

Calvert Cliffs, St Lucie 1 & 2 (Block Valves), Millstone 2, Palisades, FCS

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

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

<u>Licensing Basis for LCO</u>

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 safety valves, and analyses do not take credit for the PORV opening for accident mitigation.

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

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 depressurization when the pressurizer spray is not available, a condition that may be encountered during 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 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 Shutdown Action Statement</u>

Various LCO entry requirements exist for both PORVs inoperable or both block valves inoperable. ISTS requires the plant to restore 1 PORV to operable status or prepare to shutdown in 1 hour and maneuver to Mode 4 in 12 hours. When both block valves are inoperable, for the conditions of the PORVs inoperable, ISTS requires restoring at least 1 block valve in 2 hours or entering Mode 4 in 12 hours. Palisades requires the plant to maneuver to Mode 3 in 8 hours if both PORVs inoperable. Calvert Cliffs allows 72 hours to restore one PORV. Following inability to restore the PORV the plant is required to maneuver to 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, PORV TS for CE designed PWRs are summarized in Table 5.1.3-1. Plant specific TSs should be consulted if additional details are required.

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

Revise ISTS LCO condition E (or equivalent) allowed outage time to be consistent among CE designed PWRs (with PORVs) to allow 24 hours to restore one PORV to operability for conditions where PORV is unable to 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 LCO condition F.2 allowed outage time to allow 24 hours to restore one block valve to operable status for conditions where PORV is unable to open.

<u>Basis for Proposed Change</u>

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 analyses do not take credit for the PORV opening for accident mitigation. Section 4.1 indicates that the increased CDP associated with extending the AOT to 24 hours for inoperable PORVs (unable to open) is small (4.9E-7).

<u>Defense-in-Depth Consideration</u>

PORVs provide protection for the PSVs. Experience indicates that challenges to PORVs or PSVs are rare and that PSVs are highly reliable. This has lead to two classes of CE designed PWRs that do not include PORVs. Subsequently, a non-design core heat removal application of PORVs was identified. PORVs may also be used to control offsite releases following a limited class of severe accidents. PSVs exist and provide overpressure protection to the RCS.

<u>Tier 2 Restrictions</u>

None, except as stated in proposed modifications.

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. Internal tank pressure is sufficient to discharge the contents to the RCS, when 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 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.

<u>Plant Applicability</u>

All

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

Explicit LCO 3.0.3 entry for 2 or more SITs inoperable during Modes 1, 2 & (3 with pressurizer pressure ≥ [700] psia).

<u>Licensing Basis For Operation (LCO)</u>

When more than one SIT is inoperable, the unit is in a condition outside 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. Operability of four SITs ensures that the contents of three of the 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 below 2200 ^{o}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 Requirement (SR) 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, SITs are required in Modes 2 and 3.

Condition Requiring Entry into Shutdown Action Statement

LCO condition [D] requires immediate entry into LCO 3.0.3 if 2 or more SITs are inoperable.

Proposed Modification for Required Actions

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

Change LCO condition [D] wording for [1 or 2] or more unavailable SITs to allow 24 hours to restore all SITs to operable condition prior to LCO 3.0.3 entry. An explicit TS entry for this condition would default to a Mode 4 end state.

Basis for Proposed Change

SIT availability may alter the progression of LOCAs of smaller break sizes, and potentially alter the extent of core damage. However, the impact on the event core damage potential will be negligible. SITs are needed primarily to mitigate the Large 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 AOT is 1.4E-8. Similarly for LERP, the conservative bounding calculation results in an ICLERP of 4.1E-11. These results confirm that the risk impact of the AOT extension is negligible.

A Mode 4 end state is recommended as this operational state provides adequate and redundant systems to ensure core heat removal can reduce unnecessary plant transitions.

Defense-in-Depth Consideration

Unavailability of SITs will compromise the ability of the plant to respond to large 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 systems. It is recommended 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 large LOCA, a 24 hour period to repair or resolve the SIT

inoperability 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 AOT is consistent with the requirements of 10CFR50.46 which require that the license propose immediate steps to "bring plant design or operation" into compliance.

Tier 2 Restrictions

None. Compensatory actions to ensure both LPSIs and all HPSIs are available will partially offset the impact of SIT unavailability.

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, a default 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 SLB 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 SLB. The addition of negative reactivity is designed primarily for the loss of secondary coolant accident where primary cooldown could add enough positive reactivity to achieve criticality and return to significant power with stuck rod.

<u>Condition Requiring Entry into Shutdown Required Action</u>

Inoperability of two HPSI subsystems will result in a default entry into LCO 3.0.3.

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

It is proposed that a condition be added to the LCO addressing actions to be taken following inoperability of both HPSI pumps (or HPSI system). Such actions would allow 4 hours to resolve the inoperability and restore one train of HPSI injection capability (plant dependent) before a commencement of a plant shutdown.

<u>Basis for Proposed Change</u>

Availability of the HPSI system 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) HPSI system inoperability would result in a maximum ICCDP between 1.4E-6 and 2.0E-6 depending on whether or not the plant is equipped with PORVs. The corresponding ICLERP will be on the order of 3.0E-8. Risk associated with system inoperability 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 system 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.

<u>Defense-in-Depth Consideration</u>

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 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 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

potential resolution of minor HPSI system inoperabilities and provides time to prepare for a controlled plant shutdown while increasing very small incremental plant risks.

Extension to the recommended 4 hour interval is consistent with the risk significance of the HPSI system and the intent of 10CFR50.46 which requires the design basis of the ECCS be maintained.

Tier 2 Restrictions

Ensure at least two charging pumps are available during TS entry. Charging pumps may be used to support accident responses to smaller sized pipe failure events and for events with one or more stuck open PORVs, PSVs or SGTRs. Maintenance practices should minimize the simultaneous unavailability of similar equipment (e.g. SITs, LPSIs and swing HPSIs if available).

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 header from each LPSI pump divides into two supply lines, each feeding the injection line to two 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 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. This is accomplished by the HPSI subsystem. After the (RWST) has been depleted, the LPSI is normally shutdown and the ECCS recirculation phase is entered. During ECCS recirculation, the ECCS suction is automatically realigned to the containment sump. The LPSI subsystem increases the inventory in the RPV following events with a severe loss of inventory.

The LPSI pumps also support the shutdown cooling system. However, this function is not considered within the scope of this technical specification.

<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 LPSI must be OPERABLE.

<u>Licensing Basis for LCO</u>

The LPSI subsystem is designed to enhance the reflooding of the core following a large 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 system is 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 Shutdown Required Action

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

Proposed Modification for End State Required Actions

Add separate condition for both LPSI trains inoperable and ECCS flow equivalent less than 100% to allow the immediate shutdown requirement be extended to 24 hours. Explicit definition of this TS condition will result in a default to Mode 4 in final end state.

Basis for Proposed Change

The design basis analysis requires that one train of LPSI be available to suppress the peak fuel temperature heatup during a large LOCA event. In the SONGS PSA, LPSI is also credited in the SGTR event as necessary for Shutdown Cooling (SDC) following the late depressurization of the RCS to isolate the steam generators. Unavailability of the LPSI system for these limited time intervals, will result in a small increase CCDF ~ 4.5E-5 per year in the plant risk associated with large LOCA events. There is no significant impact of 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 plant shutdown with availability of the SDCS will offset any operational increase. A short term unavailability of the LPSI system 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 4.2-1, respectively. These analyses indicate that the ICCDP is 1.2E-7 and the ILERP 3.7E-10 for the proposed 24 hour AOT duration. These results are offset by the risk of transitioning the plant to Mode 4 (> 10.0E-6) (See References 4 and 8).

Defense-in-Depth Consideration

The primary impact of the unavailability of the LPSI system will be the reduction in the capability of the plant to provide RCS inventory makeup to accommodate a large LOCA. In addition, “at power” unavailability of the LPSIs will impair the ability of the plant to maneuver to shutdown cooling. A twenty-four hour AOT/CT is recommended for this inoperability based on the low incremental plant risk associated with continued plant operation and the inadvisability of a plant shutdown without the motive force (LPSI pumps) of the SDC.

Tier 2 Restrictions

For conditions when the LPSI system is unable to support SDC, availability of the AFW system should be assured. SIT availability should be assured to offset the large LOCA risks associated with LPSI system inoperability.

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 AOT for LCO impacted by the risk-informed change is 3.6.1B (Containment Leakage).

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 10CFR50 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 10CFR100 are satisfied following a Maximum Hypothetical Accident (MHA) initiated from full power. 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 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.

<u>Plant Applicability</u>

All

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

Restore inoperable containment in 1 hour or be in Mode 5 in 36 hours. This is an explicit LCO requirement.

<u>Licensing Basis for LCO</u>

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

The containment is intended to perform 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 10CFR50 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 well controlled within limits that assure that the post accident whole body and thyroid dose limits of 10CFR100 are satisfied following a Maximum Hypothetical Accident (MHA) initiated from full power. Inability to meet this leakage limit renders the containment INOPERABLE. Containment operability is defined as maintaining total leakage within specified limits.

Condition Requiring Entry into End State

Containment is declared to be inoperable due to excessive leakage (including leakage from airlocks and isolation valves) for a time period greater than one hour. Declaration results in an implicit 3.0.3 entry.

Proposed Modification for End State Required Actions

Define a specific action to allow 8 hours to restore an inoperable containment to operability. Allow Mode 4 to become a designated end state for correcting containment impairments for conditions where 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 proposed change modifies end state from Mode 5 to Mode 4 for conditions when the containment inoperability is not due to inability to ensure containment isolation. A risk assessment defining the Incremental LERP indicates that when one assumes that in the presence of an impaired containment, all core damage events will proceed to a large early release. Using this conservative approach, a risk-informed AOT (See Section 4.3) may be shown to be approximately 8 hours. This approach assumes a bounding plant core damage frequency of 1.0E-4 per year and an ICLERP limit of 1.0E-7. The actual risk over plant life is much lower since entry into the TS is rare and most containment challenges and containment inoperabilities will not result in large releases. The incremental release probability is offset by the potential decrease in core damage when a lower mode transition is averted.

These recommended changes apply to containment conditions where containment integrity is essentially maintained and adequate ECCS Net Positive Suction Head (NPSH) is expected following an event. 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. If accidents were to occur in Mode 4, resulting containment pressures would be significantly less than the DBA conditions. Hence, leakage would be further reduced. While in Mode 4, the probability of LOCA or MSLB is reduced from Mode 1 levels.

The implied licensing basis assumption that Mode 5 is inherently of lower operational risk than in Mode 4 is not supported by risk evaluations (See Reference 5). 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 the potential risk for containment bypass increased risks including LOCAs. Thus, based on these

PSA insights, remaining in Mode 4 (vs. Mode 5) while the containment excess leakage condition is corrected is an appropriate action. This end state would require more mitigation systems be 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 Containment Isolation Actuation Signal (CIAS) will be available to aid the operator in responding to events that threaten the reactor and/or containment integrity.

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 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. 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 more short term risk beneficial 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. Furthermore should a shutdown be required, Mode 4 would be an acceptably safe end state (See Reference 5).

The TS 3.6.1 requirement to shutdown Mode 5 is rooted in tradition rather than in consideration of risks. Accidents initiated from Mode 4 are far less challenging to the containment than those initiated from Mode 1. The lower energy content in Mode 4 results in containment pressures and potential leakage approximately one half of that associated with Mode 1 releases. Furthermore, by having the plant in a shutdown condition in advance, fission product releases are significantly reduced. Thus, while leakage restrictions should be maintained, Mode 4 leakage in excess of that allowed in Mode 1 can be safely allowed for a limited time sufficient to resolve the inoperability and return the plant to power operation.

From a deterministic perspective, Mode 4 on Steam Generator Heat Removal (SGHR) (vs. Mode 5) would maintain more mitigating systems available to respond to loss of RCS inventory or decay heat removal events and therefore reduce the overall public risk. In Mode 4, SIAS and CIAS will be available to aid the operators in responding to events that threaten the reactor and/or containment integrity. Therefore, the proposed TS end state change does not adversely affect the plant defense-in-depth.

Tier 2 Restrictions

Limitation on containment leakage is still required to ensure that a gross containment inoperability is avoided. This is accomplished in this proposed change by limiting applicability of the TS to conditions where CIVs or air locks are essentially functional (although may be formally inoperable) and have the capability to perform their containment isolation function. Conditions where a prolonged loss of containment

isolation is still expected to result in a Mode 5 end state. The decision to enter Mode 5 will be based on the plant condition and repair strategies and will include a risk assessment as required via paragraph A4 of 10CFR50.65 (Maintenance Rule). Temporary operation of the plant in Mode 4 (as opposed to Mode 5) with an "impaired" containment is not a risk significant action.

5.3 HVAC and Radiological Cleanup Systems

HVAC and radiological cleanup systems provide the plant with 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, 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. These systems were initially incorporated into plants in the belief 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 (CsI) particulates. Consequently, the actual impact of system functionality on actual public doses is negligible.

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 is to keep 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, 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 10CFR100 siting requirement dose rates and supports General Design Criteria (GDC)-19 of 10CFR50 Appendix A (Reference 10) for Control Room (CR) doses (Reference 9). 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 End State

Both ICS trains inoperable.

Proposed Modification for End State Required Actions

Revise LCO to allow 24 hours to take action before entry into LCO 3.0.3 for both ICS trains unavailable and allow Mode 4 as final end state.

Basis for Proposed Change

ICS functions together with the containment spray and containment air recirculation cooling systems following a DBA that causes failure of the fuel cladding, and release of radioactive material (principally iodine) to the containment. The ICS is specifically designed to respond to the 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, it is assumed that the amount of radioactive iodine release is limited by reducing the iodine concentration in the containment atmosphere via use of containment sprays.

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

Modification of the TS to support a Mode 4 end state may avoid 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

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 are filtered and adsorbed prior to exhausting to the environment. 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 End State

Both trains inoperable.

Proposed Modification for End State Required Actions

Revise LCO to allow 24 hours to take action before entry into LCO 3.0.3 for both SBEACS trains unavailable and allow Mode 4 as final end state.

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. 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 selected at 24 hours.

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 large early release. If accidents were to occur in Mode 4, resulting containment pressures would be significantly less than the DBA conditions. Hence, leakage would be further reduced. While in Mode 4, the probability of LOCA and MSLB is reduced from Mode 1 levels.

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 and 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 as 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.

<u>Defense-in-Depth Consideration</u>

See above discussion.

<u>Tier 2 Restrictions</u>

None.

5.3.3 (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.

<u>Plant Applicability</u>

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

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

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

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

<u>Licensing Basis for LCO</u>

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

<u>Condition Requiring Entry into End State</u>

Both Control Room Emergency Air Cleanup System (CREACUS) trains INOPERABLE in Modes 1, 2, 3 or 4.

<u>Proposed Modification of End State Required Actions</u>

Increase the time available to take action under 3.0.3 to 24 hours. Modify allowable end state to be Mode 4.

<u>Basis for Proposed Change</u>

A 24 hour AOT is based on limiting containment challenge probability to 1.0E-6 (See Section 4.4). Operation of CREATCS has no direct impact on ICCDP and ICLERP. Regardless of the system status, the risk of Mode 4 is lower (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 short term shutdown requirement 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 mitigate risk consequences further. These actions include use of portable fans and propping open doors. Several plants have such actions proceduralized.

The CREATCS is needed to protect the CR in a wide variety of circumstances. Long term plant operation in the presence of or unavailable CREATCS should result in the plant being placed in 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.

Tier 2 Restrictions

None. 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. Temporary cooling may also be established via use of portable fans, propping open doors, or similar actions. Also, availability of alternate shutdown panels and local shutdown stations should be ensured.

5.3.4 (ISTS) LCO 3.7.13 ECCS Pump Room EACS

The ECCS pump room EACS 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 pump room EACS 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 pump room EACS and Penetration Room Exhaust Air Cleanup System (PREACS) is combined within the Controlled Ventilation Area (CVAS) Technical Specification.]

Limiting Condition for Operation (LCO)

Default entry into LCO 3.0.3.

Licensing Basis for LCO

ECCS pump room EACS is typically credited in evaluating the ability of the plant to meet 10CFR100 and GDC-19 Appendix A radiation dose limits.

Condition Requiring Entry into End State

Both ECCS PREACS trains INOPERABLE

Proposed Modification of End State Required Actions

Revise LCO wording to allow 24 hours to restore one train of ECCS Pump Room EACS before entry to LCO 3.0.3 and allow Mode 4 as final end state.

Basis for Proposed Change

A 24 hour AOT is based on the likelihood of repair and limiting 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 (or equivalent) to the similar 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 while in Mode 4 is preferred.

Defense-in-Depth Consideration

ECCS pump room EACS only impacts radiation releases to the public when ECCS recirculation is in progress. This system typically is limited to LOCA transients. Releases are typically low as functional recirculation typically implies successful event mitigation. Extension of the AOT/CT to 24 hours provides time to resolve the component inoperability at power. This may potentially avert a plant shutdown and the associated transition risks.

Tier 2 Restrictions

None.

5.3.5 (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 pump room EACS and Penetration Room Exhaust Air Cleanup System (PREACS) is combined within the Controlled Ventilation Area (CVAS) Technical Specification.]

Limiting Condition for Operation (LCO)

Default entry into LCO 3.0.3 and Mode 5 in 37 hours.

Licensing Basis for LCO

The PREACS must be OPERABLE to ensure that the penetration room filtering capability is within the 10CFR100 design basis assumptions. The PREACS filters air from the penetration area between the containment and the auxiliary building.

Condition Requiring Entry into End State

Both PREACS trains INOPERABLE.

Proposed Modification of End State Required Actions

Revise LCO wording to allow 24 hours to restore one train of PREACS before entry to LCO 3.0.3 and allow Mode 4 as final end state

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 penetration room 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 (or equivalent) to the similar Mode 5 operating 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.

5.3.6 (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 PWRs 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 CEOG 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. The required actions for recovery from 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 Loss of Offsite Power (LOOP) and worst single failure. (MSLB is often the limiting accident for containment P/T control). Depending on plant design, 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. Inability to maintain subcooling will prevent ECCS recirculation cooling. For plants with diverse and redundant containment heat removal capability, consisting of both Containment Air Recirculation Coolers (CARCs) and Containment Spray (CS), availability of CARCs will compensate for the unavailability of the CS system. CS also can have the additional function of removing fission products from the post-LOCA atmosphere, in which case 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 End State

Inoperability of both Containment Spray trains.

Proposed Modification for End State Required Actions

Increase the time available to initiate shutdown to 72 hours when a CS system is inoperable and at least one train of CARCs is available.

Increase the time available to initiate shutdown to 12 hours when the CS is inoperable and the CARC's are unavailable for containment heat removal.

Basis for Proposed Change

The design basis of the CS and CARC systems varies among the CE designed PWRs. The plant design bases for many CE designed PWRs require CS and CARC 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 CARC 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 PWRs with diverse containment heat removal capability (employing both CARCs and CSs), unavailability of the CS system poses a negligible plant risk.

CS and CARC are used to support long-term containment heat removal. This heat removal is needed to ensure that 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 CARC operates. Consequently, a minimum containment heat removal capability is required to ensure both long term containment integrity and core damage prevention. CS and CARC 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, 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 LCO for the Palo Verde Units. Thus, the risk of system unavailability is increased. 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 PWRs 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 loss of CS (in the presence of a fully operational CARC 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 following SONGS Units 2 & 3 specific analyses.

For 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/CC trains, the complete PSA model was re-solved, assuming both CS trains and one CARC 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.

Additionally, the PSA model was solved assuming that all CS and CARC trains were unavailable. In this case, the annual CDF increases to 3.73E-3 and the LERF increases to 1.13E-5. This equates to a 24-hour CDP increase of 1.0E-5 and a LERP increase of 3.0E-8. These increases are greater than the acceptance criteria. Hence the 3.0.3 restrictions for loss of all CS and CARC should not be changed.

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 (IPE). Qualitatively, similar conclusions could be drawn for one train of CS. Consequently, in Mode 4 one train of CARC 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 CARCs 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 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

Inoperability of the CS or CARCs will degrade the capability of the plant to respond to a containment threat. 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 occurs (~ 2.0E-5 per week). However, the ability of the plant to cope with the event is not compromised.

Tier 2 Restrictions

None. Entry into a 72 hour AOT should be restricted to conditions where CARCs are available. Otherwise the more restrictive 12 hour AOT would be applicable.

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

PLANT	INOPERABILITY	ACTION
SONGS	2 CS trains or 3 or more CS/CC trains.	Explicit 3.0.3 entry
ANO-2	2 CS trains or 3 or more CS/CC trains.	Default 3.0.3
Calvert Cliffs	3 or more CS/CC trains unavailable	Explicit 3.0.3 entry
FCS	All 3 CS pumps inoperable All 3 containment fan coolers inoperable	Explicit 2.0.1 (3.0.3 equivalent)
WSES	2 CS trains inoperable	Default 3.0.3
MP2	2 CS trains inoperable	Explicit 3.0.3

6.0 SUMMARY

This report justifies modifications to various Technical Specification (TS) Required Action Statements for the conditions that imply 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 extended operating period is recommended to be 24 hours. In specific instances, recommendations for longer and shorter action times 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 and RG 1.177, as appropriate. Risk assessments performed to support these modifications are based on bounding analyses and are applicable to the entire fleet of CE designed PWRs operated in the United States. Furthermore, risks associated with the implementation of these TS changes will be managed in accordance with paragraph a(4) of 10CFR50.65 Maintenance Rule (MR).

The benefit from these changes is that the proposed AOT 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.

7.0 REFERENCES

1. RG 1.174, USNRC, “An Approach for using Probabilistic Risk Assessment in Risk-Informed Decision 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 1, April 1995, 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. 10CFR100, “Reactor Site Criteria,” USNRC, 1991.
10. 10CFR50, “Appendix A: General Design Criteria,” 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.

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.)

Table A-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, then Mode 5 in 8.25 days (1 of 2)	Mode 3 in 6, (2 of 2 inop.) .... Mode 3 in 78, then Mode 5 in 8.25 days (1 of 2 inop.)	NA	**Restore 1 in 48 or Mode 3 & borated in 2 , then 7 days to restore 1 or Mode 5 in 36**
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	NA	Default 3.0.3	Default 3.0.3	(Mode 1-2) **Mode 4 in 6**	NA	Mode 4 in 12	Restore in 72 or Mode 3 in 12	Default 3.0.3	Mode 4 in 12

Table A-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**	NA (no PORVs)	NA (no PORVs)	**Restore 1 in 72 or Mode 3 & ≤ 365F-U1 301F-U2 in 12**	**NA** (no PORVs, but 4 PSVs)	NA (for PORVs) **Restore or close 1 in 1 hr, or Mode 5 in 36** (both BVs inop.)	NA (for PORVs) **Close 1 in 1 hr, or Mode 5 in 36** (both BVs open)	NA (no PORVs)	**Restore 1 in 1 hr or close both block valves & Mode 4 in 42** (PORVs) **Restore 1 in 2 hrs & both in 74 or Mode 4 in 12** (BVs)	**Restore in 72 or Mode 3 & ≤ 365F-U1 301F-U2 in 12**	**Restore 1 in 1 hr or Mode 4 in 12** (PORVs) **Restore 1 in 2 hrs and both in 74 hrs or Mode 4 in 12** (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 A	HPSI (2 of 2)	Implicit 3.0.3	Implicit 3.0.3	Implicit 3.0.3	Implicit 3.0.3	Implicit 3.0.3	Implicit 3.0.3	Implicit 3.0.3	Implicit 3.0.3	Default 2.0.1	Implicit 3.0.3	Implicit 3.0.3

Table A-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 or Mode 4 & Pzr <1700 in 12	Default 3.0.3	Default 3.0.3	Restore 1 in 72 or Mode 3 & Pzr < 1750 psi in 12	Restore 1 in 72 or Mode 3 & Pzr < 1837 psi & < 485F in 12	Default 3.0.3	Default 3.0.3	Default 3.0.3	Default 2.0.1	Restore 1 in 72 or Mode 3 & < 1750 psi in 12	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
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 or Mode 5 in 36** (for both CC)	(Mode 1-3 **Explicit 3.0.3** (for 2 CS or 3 or more CS/CC) **Restore 1 in 72 or Mode 4 in 36** (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 hours & both in 7 days or Mode 5 in 36 hours** (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** (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** (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 or Mode 4 in 12** (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 or Mode 5 in 36** (for both CS or both CC)	(Mode 4 only) **NA** (CS) **Restore 1 in 72 or Mode 5 in 36** (for both CC)	**NA** (CS) **NA** (CC)	(Mode 1-3) **Explicit 3.0.3** (for 3 or more CS/CC) **Restore 1 in 72 or Mode 4 in 12** (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) **Restore 1 in 72 or Mode 4 in 30** (for both CC)	**NA** (CS) **NA** (CC)

Table A-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**
3.6.10 (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]	(Mode 1-3 **Restore in 72[9]**	NA	Restore in 24[10]	**NA**	NA
3.7 Plant Systems														
3.7.10 (Mode 1-4)	3.7.10 (Mode 1-4)	3.7.10	ECW	Default 3.0.3	Default 3.0.3	NA	NA	Default 3.0.3	NA	NA	Default 3.0.3	NA	NA	NA
3.7.11 E (Mode 1-6)	3.7.11 E (Mode 1-6)	3.7.11 D	CREACUS	Explicit 3.0.3	Explicit 3.0.3	Default 3.0.3 (CREVAS)	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	NA	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 13**	NA	NA	NA	Default 3.0.3 (CVAS)	NA	NA	NA

<u>Footnotes to Table A-1</u>

(1) Not used.
(2) Not applicable to all PWR designs.
(3) Fort Calhoun end states are different:

Mode 1 = Operating (Reactor Power ≥ 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).

Table A-2
Technical Specification Numbering Cross-Reference

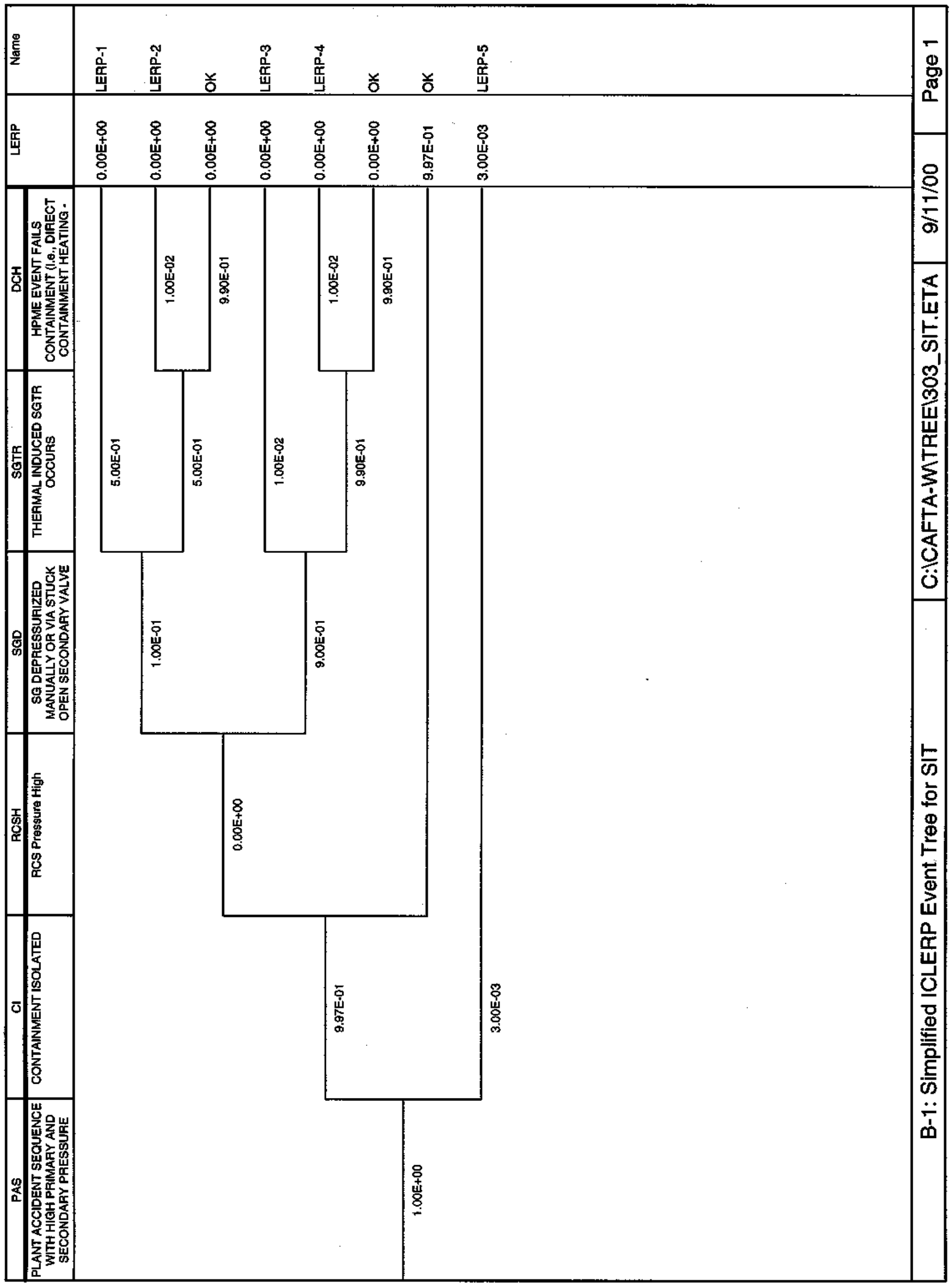
ISTS		SONGS TS #	Title	Current End State								
Analog	**Digital**			**ANO**	**CC**	**Palo Verde**	**SL-1**	**SL-2**	**WSES**	**FCS**	**PAL**	**MP2**
3.1 Reactivity Control System												
None	None	3.1.9	Boration Systems - Operating	3.1.2.2 - flow path 3.1.2.8 - BAT	NA	NA	3.1.2.2 3.1.2.8	3.1.2.2 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	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	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 A	HPSI	3.5.2	3.5.2 A	3.5.3 B	3.5.2	3.5.2	3.5.2			3.5.2
3.5.2 A	3,5,2 A	3.5.2 A	LPSI	3.5.2	3.5.2 A	3.5.3 B	3.5.2	3.5.2	3.5.2			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.D	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.B (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
3.7 Plant Systems												
3.7.10	3.7.10	3.7.10	Emergency Chilled Water System	NA	NA	3.7.10.B (ECW)	NA	NA	3.7.12 (ESCWS)	NA	NA	NA
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 E (CRV)	3.7.6.1b (CREVS)
3.7.12 E	3.7.12 E	NA	CREATCS	NA	3.7.9 D (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

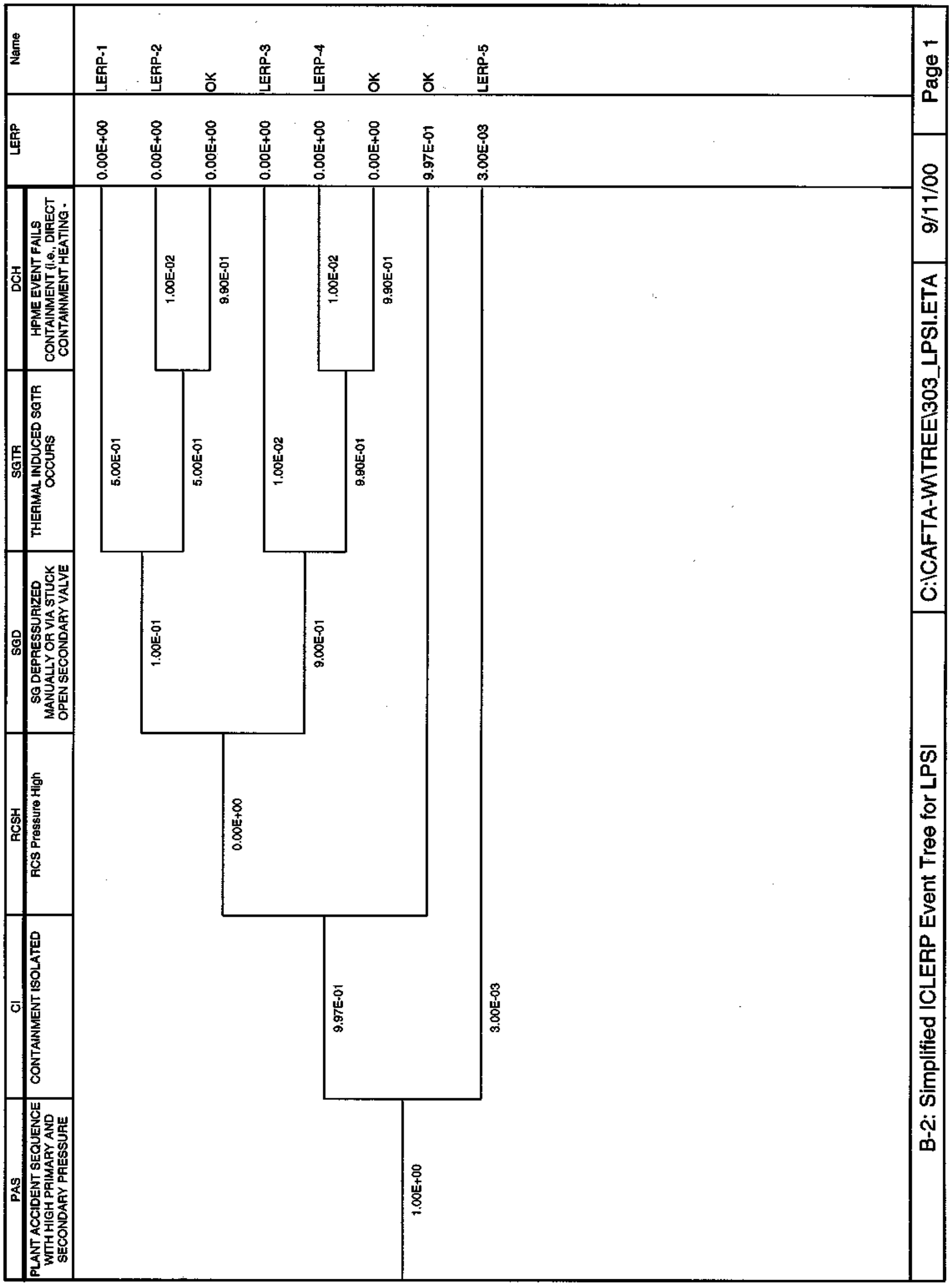
APPENDIX B

System Specific LERF Event Trees

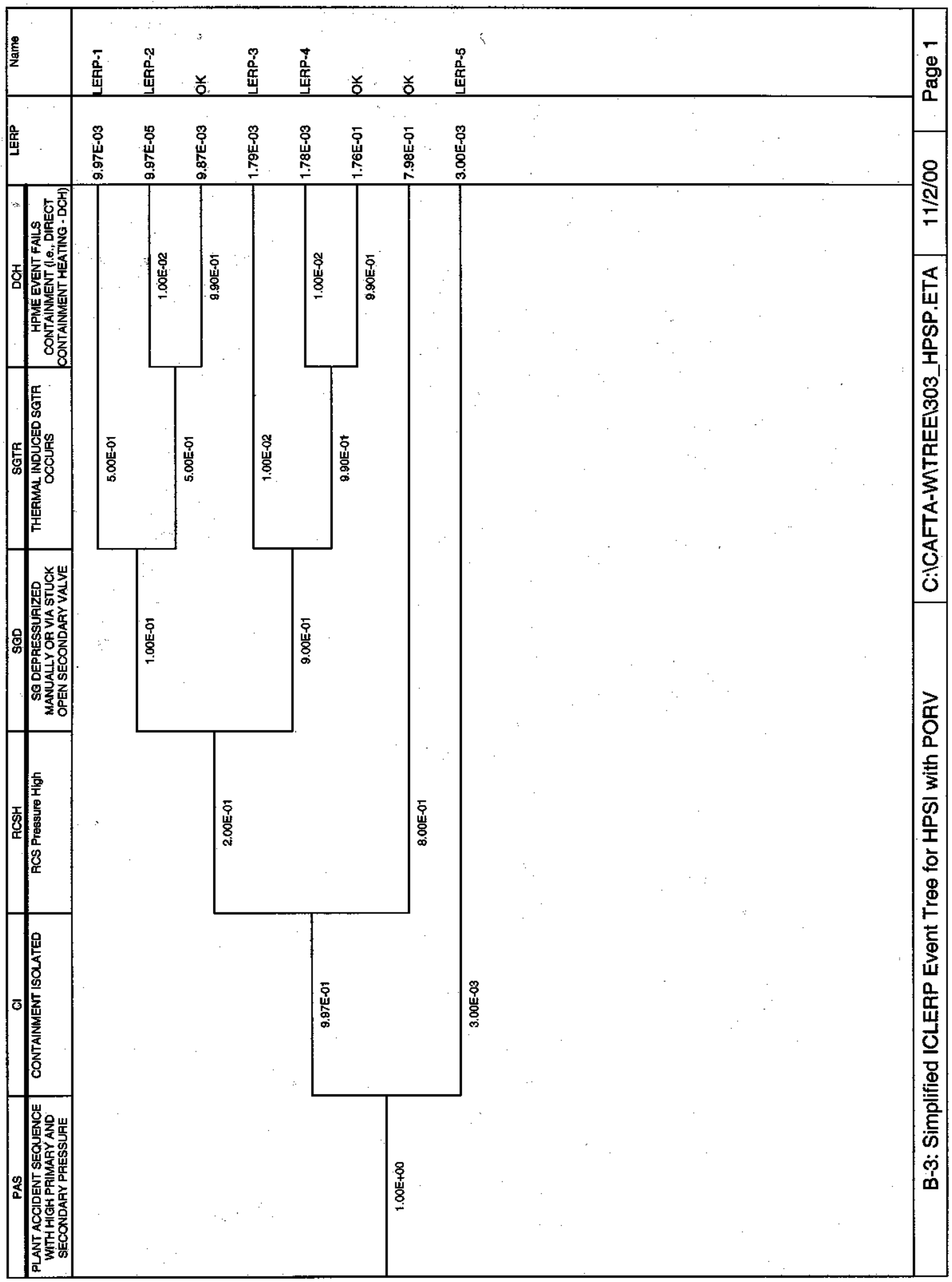
[This appendix contains the simplified Large Early Release event trees for the systems evaluated. The values used to the probability for the event tree scenarios for a normalized ICCDP are also shown.]



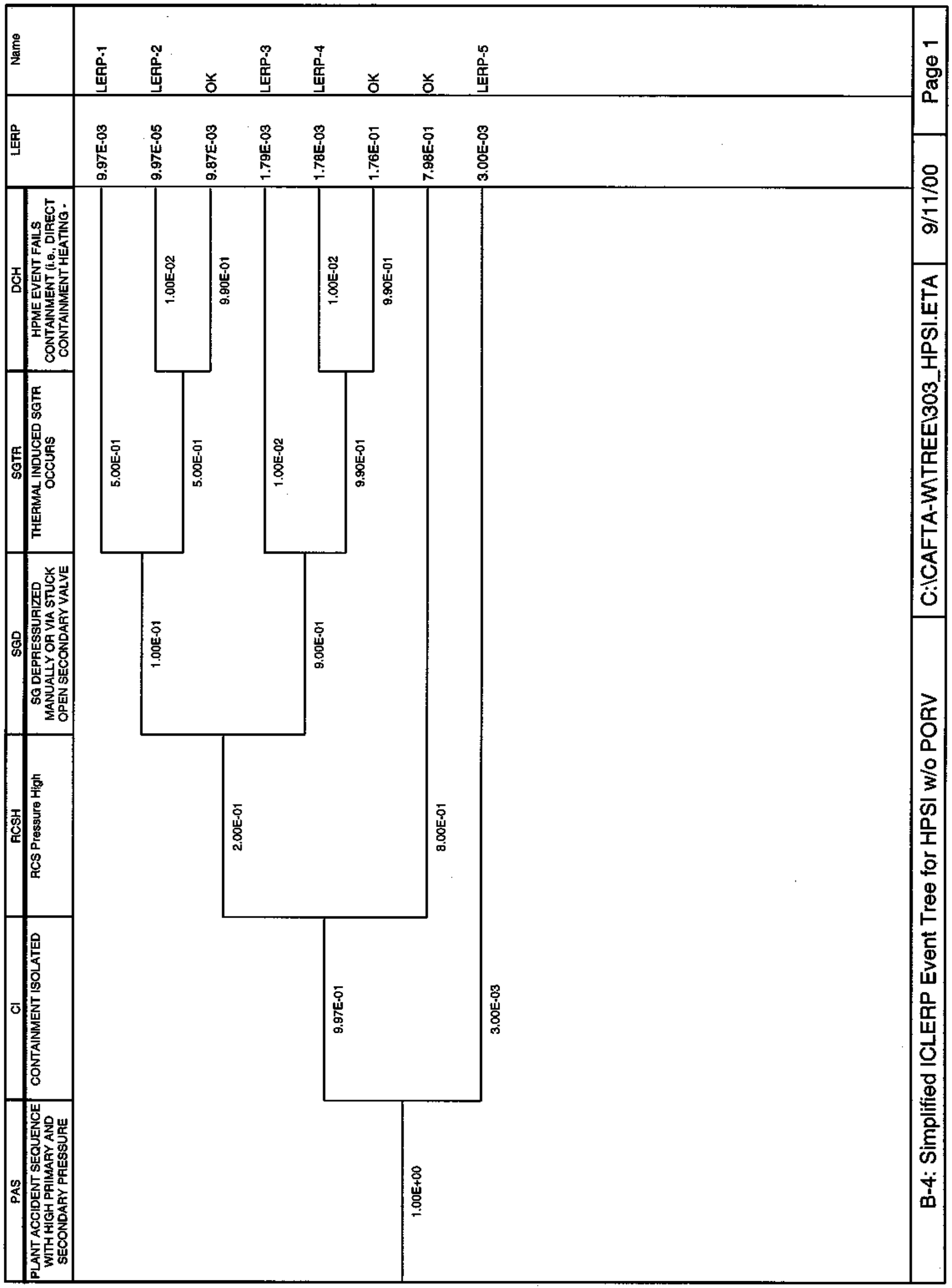
B-1: Simplified ICLERP Event Tree for SIT



B-2: Simplified ICLERP Event Tree for LPSI



B-3: Simplified ICLERP Event Tree for HPSI with PORV


PAS
PLANT ACCIDENT SEQUENCE WITH HIGH PRIMARY AND SECONDARY PRESSURE
CI
CONTAINMENT ISOLATED
RCSH
RCS Pressure High
SGD
SG DEPRESSURIZED MANUALLY OR VIA STUCK OPEN SECONDARY VALVE
SGTR
THERMAL INDUCED SGTR OCCURS
DCH
HPME EVENT FAILS CONTAINMENT (i.e., DIRECT CONTAINMENT HEATING -
LERP
Name
1.00E+00
9.97E-01
3.00E-03
2.00E-01
8.00E-01
1.00E-01
9.00E-01
5.00E-01
5.00E-01
1.00E-02
9.90E-01
1.00E-02
9.90E-01
1.00E-02
9.90E-01
9.97E-03 LERP-1
9.97E-05 LERP-2
9.87E-03 OK
1.79E-03 LERP-3
1.78E-03 LERP-4
1.76E-01 OK
7.98E-01 OK
3.00E-03 LERP-5
B-4: Simplified ICLERP Event Tree for HPSI w/o PORV
C:\CAFTA-W\TREE\303_HPSI.ETA
9/11/00
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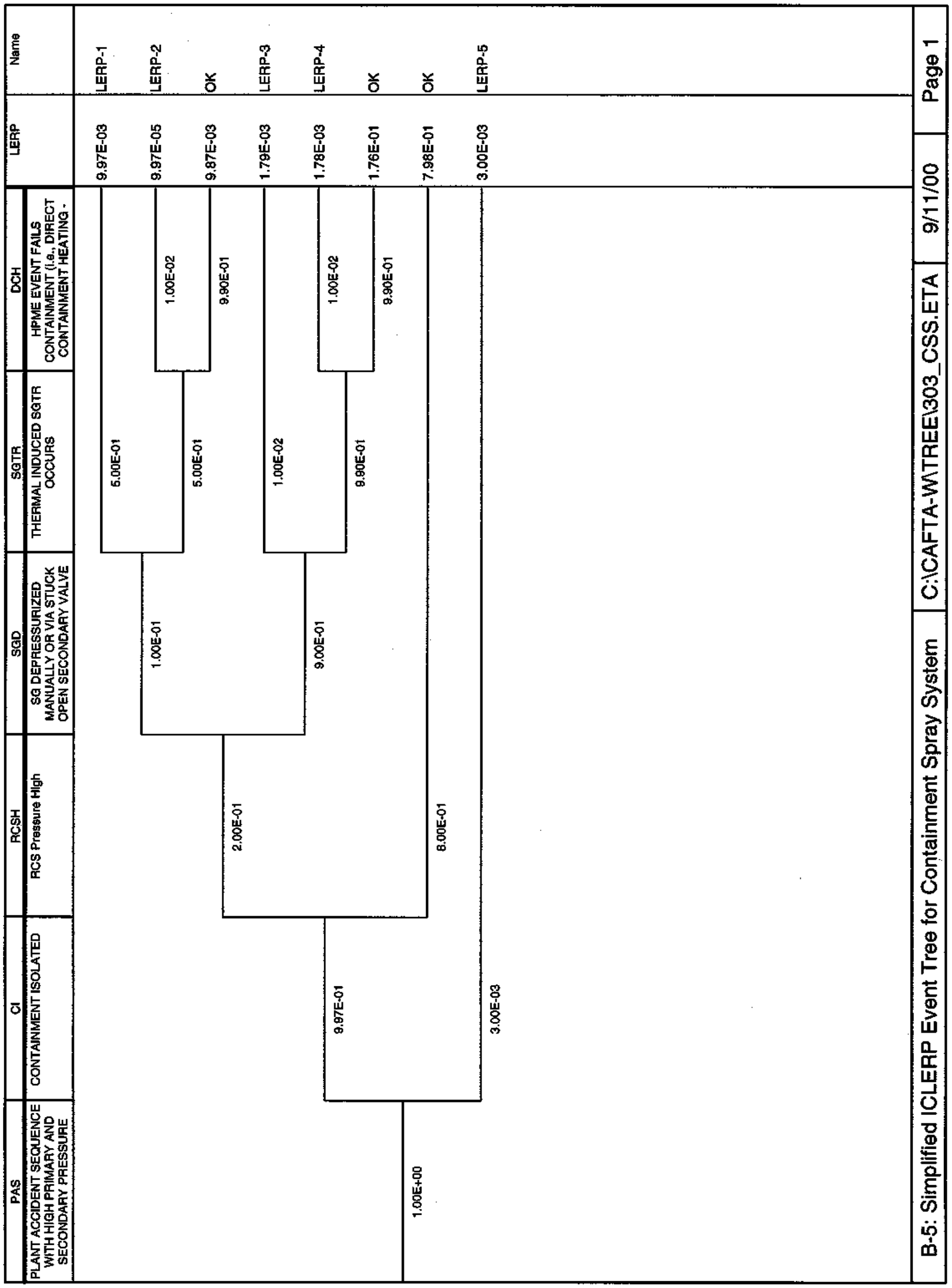
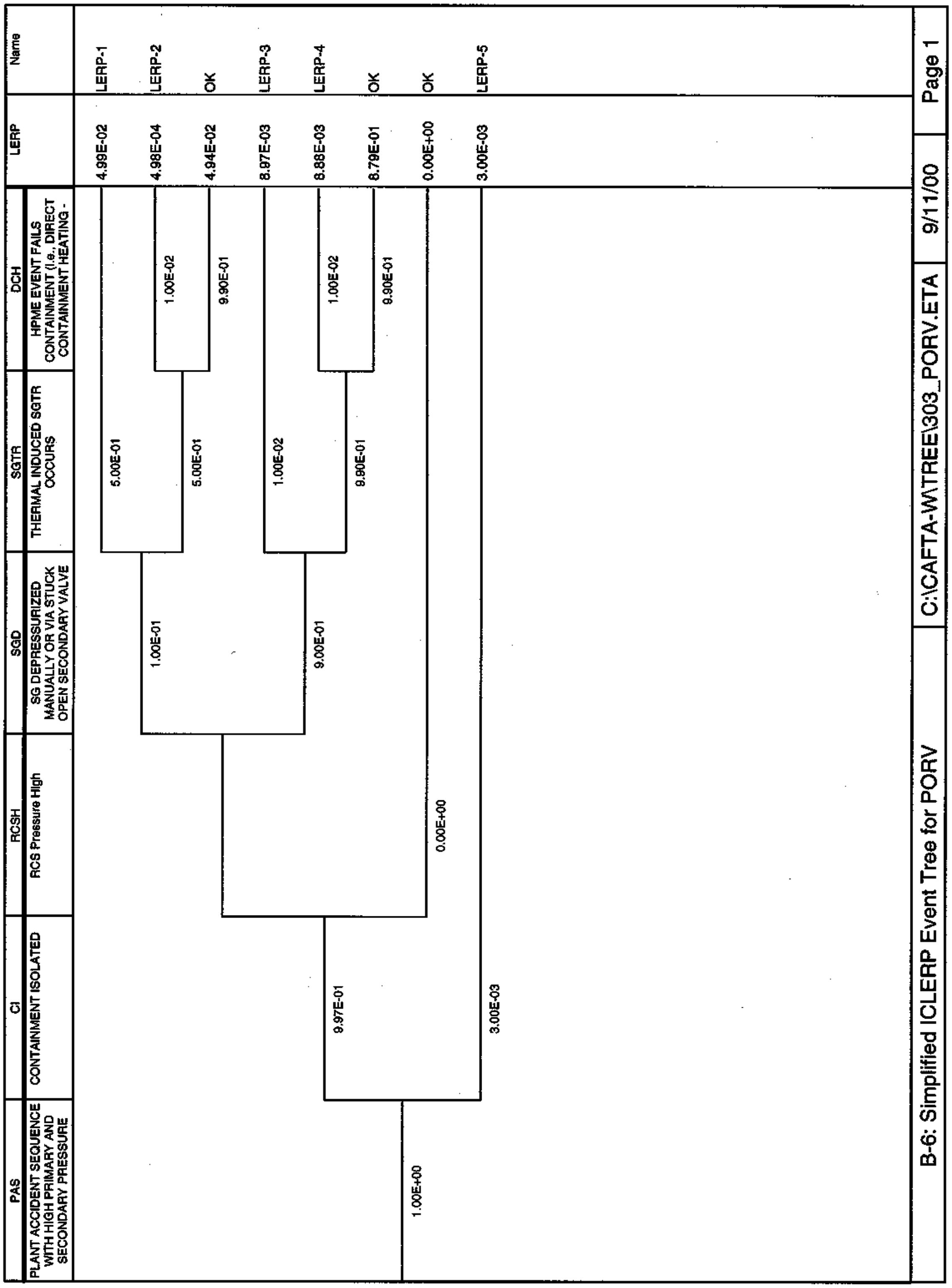
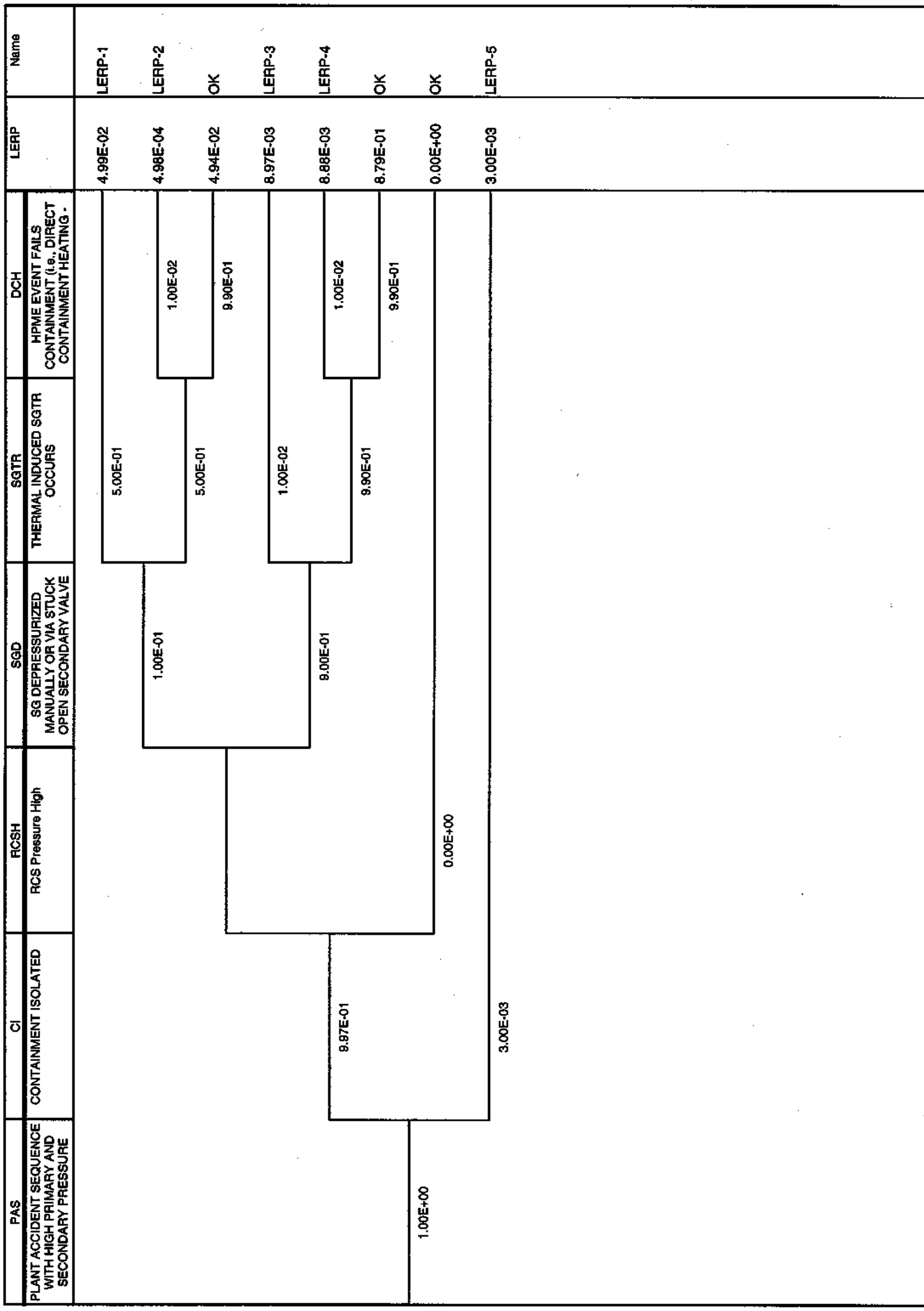
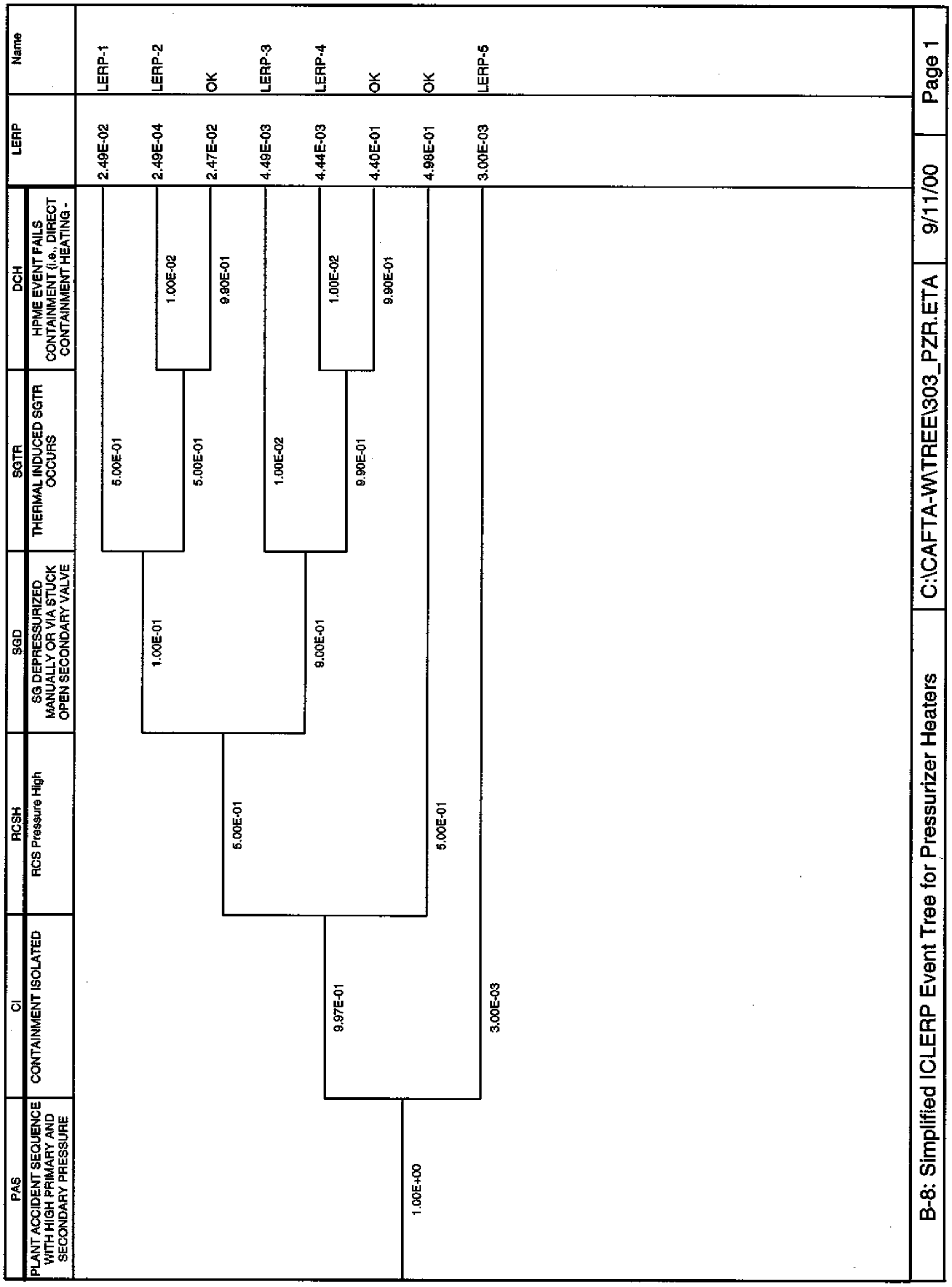


B-5: Simplified ICLERP Event Tree for Containment Spray System


PAS
PLANT ACCIDENT SEQUENCE WITH HIGH PRIMARY AND SECONDARY PRESSURE
CI
CONTAINMENT ISOLATED
RCSH
RCS Pressure High
SGD
SG DEPRESSURIZED MANUALLY OR VIA STUCK OPEN SECONDARY VALVE
SGTR
THERMAL INDUCED SGTR OCCURS
DCH
HPME EVENT FAILS CONTAINMENT (i.e., DIRECT CONTAINMENT HEATING -
LERP
Name
1.00E+00
9.97E-01
3.00E-03
0.00E+00
1.00E-01
9.00E-01
5.00E-01
5.00E-01
1.00E-02
9.90E-01
1.00E-02
9.90E-01
1.00E-02
9.90E-01
4.99E-02
LERP-1
4.98E-04
LERP-2
4.94E-02
OK
8.97E-03
LERP-3
8.88E-03
LERP-4
8.79E-01
OK
0.00E+00
OK
3.00E-03
LERP-5
B-6: Simplified ICLERP Event Tree for PORV
C:\CAFTA-W\TREE\303_PORV.ETA
9/11/00
Page 1



B-7: Simplified ICLERP Event Tree for Boration Systems | C:\CAFTA-W\TREE\303_BOR.ETA | 9/11/00 | Page 1



B-8: Simplified ICLERP Event Tree for Pressurizer Heaters