Westinghouse Non-Proprietary Class 3

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(CE NPSD-1208, Revision 02)

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

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(CE NPSD-1208, Revision 02)

Justification for Risk-Informed Modifications to Selected Technical Specifications for Conditions Leading to Exigent Plant Shutdown (PA-LSC-0364)

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December 2007

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**Electronically approved records are authenticated in the Electronic Document Management System*

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Constellation Energy Group	Calvert Cliffs 1 & 2 (CE)	X	
Constellation Energy Group	Ginna (W)		X
Dominion Connecticut	Millstone 2 (CE)	X	
Dominion Connecticut	Millstone 3 (W)		X
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Dominion VA	North Anna 1 & 2, Surry 1 & 2 (W)		X
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Korea Hydro & Nuclear Power Corp.	Kori 1, 2, 3 & 4 Yonggwang 1 & 2 (W)		X
Korea Hydro & Nuclear Power Corp.	Yonggwang 3, 4, 5 & 6 Ulchin 3, 4 , 5 & 6(CE)	X	
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Nordostschweizerische Kraftwerke AG (NOK)	Beznau 1 & 2 (W)		X
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Spanish Utilities	Asco 1 & 2, Vandellos 2, Almaraz 1 & 2 (W)		X
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EXECUTIVE SUMMARY

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

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

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

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

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ACRONYMS

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

ACRONYMS (CONTINUED)

hrs	-	Hours
ICCDP	-	Incremental Conditional Core Damage Probability
ICLERP	-	Incremental Conditional Large Early Release Probability
ICS	-	Iodine Cleanup System
IEF	-	Initiating Event Frequency
ISTS	-	Improved Standard Technical Specifications
LBLOCA	-	Large Break Loss of Coolant Accident
LCO	-	Limiting Condition for Operation
LERF	-	Large Early Release Frequency
LER	-	Large Early Release
LERP	-	Large Early Release Probability
LOCA	-	Loss of Coolant Accident
LOFW	-	Loss of Feedwater
LOOP	-	Loss of Offsite Power
LPSI	-	Low Pressure Safety Injection
LTOP	-	Low Temperature Overpressure Protection
MBLOCA	-	Medium Break Loss of Coolant Accident
MFW	-	Main Feedwater
MHA	-	Maximum Hypothetical Accident
MP2	-	Millstone Plant Unit 2
MSLB	-	Main Steam Line Break
MSSV	-	Main Steam Safety Valves
MTC	-	Moderator Temperature Coefficient
NC	-	Natural Circulation
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
QHO	-	Quantitative Health Objective
RCS	-	Reactor Coolant System
RG	-	Regulatory Guide
RPS	-	Reactor Protection System
RPV	-	Reactor Pressure Vessel
RWST	-	Refueling Water Storage Tank
SB	-	Shield Building
SBLOCA	-	Small Break Loss of Coolant Accident
SBEACS	-	Shield Building Exhaust Air Cleanup System
SDC	-	Shutdown Cooling
SDM	-	Shutdown Margin
SG	-	Steam Generator

ACRONYMS (CONTINUED)

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

1.0 PURPOSE

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

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

Revision 1 of WCAP-16125 updates and supersedes Revision 0 in its entirety. Specifically, Revision 1 of WCAP-16125 differs from Revision 0 with regard to the following:

1. Inclusion of additional information in support of proposed changes to LCO 3.5.2 "ECCS-Operating," and modifications of the associated required actions.
2. Modification of Control Room Habitability TS conditions LCO 3.7.11 and 3.7.12 to reflect added defense-in-depth requirements.
3. Removal of AOT extension requests for LCO 3.5.2; "HPSI" and LCO 3.6.1 "Containment"
4. Enhanced basis discussion on design basis and potential risks associated with inoperability of fission product cleanup systems, including added discussion on defense-in-depth considerations.
5. Inclusion of a clarification for LCO 3. 4.9, "Pressurizer," discussion regarding post accident operator cooldown procedures without availability of emergency heaters.
6. Extension of applicability of the AOT extension request for LCO 3.6.6.A to LCO 3.6.6.B, as appropriate.
7. Several end state change requests have been removed.

It should be noted that specific changes have been made to the proposed TS change request for LCOs 3.4.11, 3.5.1, 3.5.2, 3.6.8, 3.7.11, 3.7.12 and 3.7.13 to explicitly include compensatory measures within the TS Action Statement framework. The actions included in these statements are based on defense and depth considerations and are intended to significantly reduce the impact of the inoperability. While these actions are not considered necessary to avoid a high risk condition, the actions mitigate the design basis impact of the change and have been identified as Tier 2 action since the TS condition is contained directly in the action statement. Tier 3 considerations typically include actions that reduce the impact of the design change, expedite component repair and/or increase the awareness of the unavailability. TS specific Tier 3 actions are derived from defense-in-depth considerations and are explicitly identified. Changes proposed in this document have also been considered in the development of Revision 1 to TSTF 426.

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2.0 SCOPE OF PROPOSED CHANGES TO TECHNICAL SPECIFICATIONS

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

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

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

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

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

NUREG-1432	SYSTEM/ COMPONENT	CONDITION	CURRENT COMPLETION TIME	PROPOSED COMPLETION TIME TO RESTORE [ONE] TRAIN	PROPOSED END STATE IF ACTION NOT MET (See Note 4, 5, 8)	CCDP (See Notes 1 & 2)	CLERP (See Note 1)
3.1.9 (NA-ISTS)	Boration System	System Inoperable	No Condition defined. Default LCO 3.0.3 entry.	24 hrs	Unchanged	4.7E-8	3.4E-9
3.4.9	Pressurizer Heaters	Two Groups of Class 1E Heaters Inoperable	No Condition defined. Default LCO 3.0.3 entry.	24 hrs	Unchanged	3.0E-7 (See Note 10)	1.1E-8
3.4.11	PORVs	Inability of two PORVs to Open, or Inability of both PORVs to close and block valves to be closed	Separate Condition Entry Allowed for each PORV Mode 4 In 12 hrs	8 hrs. Confirm LCO 3.7.5 met	Unchanged	9.2E-7	6.7E-8
3.5.1	SITs	Two or More SITs Inoperable	Explicit LCO 3.0.3 entry	24 hrs to return all but one SIT to operable status. All ECCS trains are Operable	Unchanged	< 1.4E-8	4.1E-11
3.5.2	LPSI (See Note 3)	Two LPSI subsystems Inoperable	Explicit LCO 3.0.3 Entry	24 hrs if all SITS are Operable. If all HPSI subsystems are not available, LCO 3.0.3 entry is required under 3.5.2 Condition E (revised).	Unchanged	1.2E-7	3.7E-10
3.6.6.A	CSS (See Note 4)	Two Containment Spray Trains Inoperable Two Containment Spray Trains and two Containment Cooling Trains inoperable.	Explicit LCO 3.0.3 entry	72 hours if two Containment Spray Trains inoperable. 12 hours if two Containment Spray Trains and two Containment Cooling Trains inoperable.	Mode 4 in 12 hrs (See Note 9)	7.5E-7 (when CARC not available) Insignificant impact for PWRs with diverse containment cooling systems (See Note 6)	Note 11
3.6.6.B	CSS (See Note 4)	Two Containment Spray Trains and two Containment Cooling Trains inoperable	Explicit LCO 3.0.3 entry	12 hrs.	Mode 4 in 12 hrs (See Note 9)	7.5E-7 (when CARC not available) (See Note 6)	Note 11
3.6.8	SBEACS	Two Trains Inoperable	No condition defined. Default LCO 3.0.3 entry.	24 hrs. Confirm availability of one train of CSS	Mode 4 in 12 hrs (See Note 9)	Note 7	Note 7

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

NUREG-1432	SYSTEM/ COMPONENT	CONDITION	CURRENT COMPLETION TIME	PROPOSED COMPLETION TIME TO RESTORE [ONE] TRAIN	PROPOSED END STATE IF ACTION NOT MET (See Note 4, 5, 8)	CCDP (See Notes 1 & 2)	CLERP (See Note 1)
3.6.10	ICS	Two Trains Inoperable	No Condition defined. Default LCO 3.0.3 entry	24 hrs. Confirm availability of one train of CSS.	Mode 4 in 12 hrs (See Note 9)	0	0
3.7.11	CREACS	Two Trains Inoperable*	Explicit LCO 3.0.3 entry	24 hrs. Confirm LCO 3.4.16 met Upon entry initiate action to implement mitigating actions.	unchanged	0 (See Note 7)	0 (See Note 7)
3.7.12	CREATCS	Two Trains Inoperable in MODE 1, 2, 3, or 4	Explicit LCO 3.0.3 entry	24 hrs.	unchanged	0 (See Note 7)	0 (See Note 7)
3.7.13	ECCS PREACS	Two Trains Inoperable*	No condition defined. Default LCO 3.0.3 entry.	24 hrs. Confirm operability of one train of CREACS.	Mode 4 in 12 hrs (See Note 9)	0 (See Note 7)	0 (See Note 7)
3.7.15	PREACS	Two Trains Inoperable*	No condition defined. Default LCO 3.0.3 entry.	24 hrs Confirm operability of one train of CSS	Mode 4 in 12 hrs (See Note 9)	0 (See Note 7)	0 (See Note 7)

NA – Not applicable * Two trains inoperable for reasons other than inoperable boundary

Notes for Table 2-1:

1 Based on continued "at power" operation for <u>full AOT</u> (for ICCDPs and ICLERPs crediting the current one hour. See Tables 4.1-2 and 4.2-1a, respectively).
2 See Section 4.
3 Mode 5 end state not desirable as SDC is compromised. Mode 4 is low risk end state.
4 CSS proposed AOT applies to both containment cooling TSs.
5 Mode 3 - hot standby; Mode 4 - hot shutdown; Mode 5 - cold shutdown.
6 For plants with non-diverse containment cooling systems, unavailability of CSs is assumed to prevent the establishment of ECCS recirculation and result in core damage (See Table 4.2-1a).
7 AOT reasonable based on low probability of an initiating event occurring during the outage (See Section 4.4).
8 End state consistent with Reference 4.
9 Current LCO 3.0.3 entry requires Mode 5 end state. Mode 4 end states allows for greater resources for event mitigation, and poses an overall lower plant risk that operating in Mode 5 (See Reference 4)
10 Assumes probability of manual RCS pressure control is high.
11 Containment Sprays or Fan Coolers are effective in controlling long term containment challenges and fission product releases. These devices do not play a significant role in averting LERF.

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3.0 BACKGROUND

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

This report provides technical justification for the modification of various TSs to define and/or modify Actions to extend the time required to initiate a plant shutdown from 1 hour (e.g. TS LCO 3.0.3 entry) to a risk-informed time varying from 8 hours to 72 hours, dependent upon the TS system/component and plant design features. In addition, the report proposes, consistent with Reference 4, the modification of selected CT/AOT TS Actions to allow a Mode 4 end state when the time cannot be met.

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

(a) Avoiding unnecessary plant shutdowns.
(b) Minimizing plant transitions and associated transition and realignment risks.
(c) Providing for increased flexibility in scheduling and performing maintenance and surveillance activities.
(d) Providing explicit guidance where none currently exists.

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

1) Accident Prevention
2) Accident Mitigation
3) Control of Radiation Releases to the Environment

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

The second category is comprised of components designed to support accident mitigation. These systems typically impact both the core damage and large early release probabilities. These systems/components are typically highly reliable, and normally available in a standby mode. Systems/components in this category are intended to function during rare, but high consequence, events. This category includes the components of the Emergency Core Cooling System (ECCS), Containment Spray System (CSS) and the pressurizer Power Operated Relief Valves (PORVs)[1]. Functions of the containment cooling systems may also be grouped in this category.

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

Risk assessments performed within the scope of this task are consistent with the general guidance of RGs 1.174 and 1.177. Where possible, risk-informed assessments of the proposed TS modifications are established based on bounding assumptions. All PWROG members with CE designed NSSSs consider the supporting analytical material contained within the document to be applicable to their respective member utilities, regardless of the format of their plant TSs.

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

4.0 RISK-INFORMED EVALUATION OF ALLOWED OUTAGE TIMES

This section presents a methodology for a risk-informed assessment of AOTs when a system and its associated design function is unavailable. The general methods used to support the risk-informed evaluations are based on RGs 1.174 and 1.177. In performing the evaluation, two conditions are 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 a conservative assessment resulting from incomplete knowledge of the cause of the component failure or a partial system failures rather than a deleterious common-cause failure of critical components leading to a functional failure of the entire system.

and,

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

The extended time intervals sought to replace the LCO 3.0.3 entry or one hour Action Statement are relatively short (generally, one day or less), and are expected to be non-repetitive and infrequently entered.

The criteria for the risk-informed assessment of the AOTs were selected based on RG 1.174. Regulatory Guide 1.174 indicates that for plant changes which would result in an increase in Core Damage Frequency (CDF) of less than 1.0E-6 per year and an increase in Large Early Release Frequency (LERF) of less than 1.0E-7 per year, the incremental change is considered small.

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

Several systems contained within the TSs have no contribution, or a relatively indirect contribution, to either core damage or large early release. Such systems include those associated with the control room ventilation envelope, containment ventilation envelope, containment negative pressure protection and containment radionuclide control. While, in some instances, these systems may contribute to long-term public doses, their "risk " impact as assessed via Level 1 and 2 PSAs has consistently proven to be negligible. However, these systems do support the important design objective of meeting 10CFR100.11, and 10CFR50 GDC 19 dose criteria by helping to control the magnitude of radiological releases following an accident. Even so, recommended AOTs for these systems have been established consistent with the desire to ensure that functional loss of radiological cleanup ventilation systems should be limited to 24 hours. AOTs of twenty four hours or less are compatible with the need to take immediate action. This is a conservative position as system challenge is not necessarily associated with core damage or significant radiation releases. That is, while these systems may impact (to some extent) the amount of release, they do not impact the Core Damage Frequency (CDF) or Large Early Release Frequency (LERF).

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 applicable to Modes 1 through 4 (unless otherwise stated) and 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. Defense-in-depth is considered via controlling the outage time for related equipment, restricting activities which may challenge these systems, and where possible, using compensatory measures to limit concurrent unavailabilities and evaluating repair activities and alternatives. In some instances the required defense-in-depth or compensatory measures are directly included in the TS Actions. In some instances recommended actions are to be included in the TS Bases and/or administrative guidance. This guidance will be in addition to that associated with 10 CFR 50.65(a)(4) and associated guidance documents.

4.1 ASSESSMENT OF CORE DAMAGE PROBABILITIES

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

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

The present methodology provides a bounding generic approach for evaluating the incremental Conditional Core Damage Probability (CCDP) where possible. This approach can be implemented for evaluating the risks associated with the unavailability of standby mitigating systems. (A variant of this approach is applied to components whose unavailability impacts the plant trip probability, see Section 4.1.2.) "At power" systems/components of interest that can be grouped in the standby mitigating systems category include the Safety Injection Tanks (SITs), Low Pressure Safety Injection (LPSI), Containment Sprays, 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. No recovery actions are considered. Table 4.1-1 identifies the relationship of the mitigating systems to the initiating event frequencies against which they are designed to protect. Initiating frequencies are established from Reference 7. Detailed table notes provide additional information pertaining to the Initiating Event Frequency (IEF) assessment. In general, it is assumed that the unavailability of the affected system will lead to all associated events progressing towards core damage. Potential mitigating strategies not credited in this analysis and other associated conservatisms are summarized in response to request for additional information question 4 (Reference 21).

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

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

where:

$ICCDP_{goal}$ = 1.0E-6

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

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

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

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

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

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

where:

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

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

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

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

Notes for Table 4.1-1

(a) Data extracted from Table 3-1 and 3-8 of Reference 7.

(b) System/Component is not required to avert core damage for this event.

(c) The frequency of challenging F&B is estimated as the product of the frequency of events that lead directly or indirectly to a loss of Main Feedwater (MFW) and the probability of failing Auxiliary Feedwater (AFW). Events that lead directly or indirectly to a loss of MFW include a total loss of MFW flow, excessive or partial loss of MFW flow, and Loss of Offsite Power (LOOP). Based on information provided in Tables 3-1, 4-7 and 4-1 of Reference 2, the estimated frequency for the loss of MFW events defined herein is 5.03E-01 per year. The AFW failure probability is 2.0E-03, which is a bounding value for CE designed NSSSs. (See Table D-6 of Reference 19.) The estimated frequency of challenging F&B becomes 1.01E-03 per year.

(d) Components may be used as a backup mitigating component, however its risk importance is low in these sequences due to the high reliability of the primary component and the common dependencies.

(e) deleted

(f) This is taken as the product of the initiating event frequency based on the limited set of transients for ATWS and the failure probability of the RPS. The initiating event frequency is 1.4 per year. Using a generic RPS failure probability of 1.2E-5 per demand, the ATWS initiating event frequency becomes 1.68E-5 per year. This frequency is rounded up to 1.7E-5 per year. PORVs may be used to mitigate ATWS events and in a proceduralized manner to effect feed and bleed following a loss of FW events. Assume 50% of ATWS events require PORVs for event mitigation. ATWS events that occur in MOC/EOC do not require PORVs.

(g) Based on the total of applicable initiating event frequencies.

(h) deleted

(i) NA – Not applicable.

(j) Containment heat removal is required to ensure sump cooling. Sump cooling is not required with these events as they may be mitigated using injection resources.

(k) The ATWS values from Table 3-8 of Reference 7 represent CDF due ATWS, rather than the initiating event frequency for ATWS. ATWS frequency is calculated as follows: $ATWS_f = I_T \times RPS = 1.4 \times 1.2E\text{-}5 = 1.68E\text{-}05$ per year (value rounded up to 1.7E-05 per year).

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

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

Notes for Table 4.1-2
(a) Based on incremental time (AOT - 1 hr)
(b) The time is rounded up to the nearest hour.

The above table suggests that the SITs, LPSI, and boration systems are clear candidates for having alternative Required Actions in the Technical Specification. Changes to the CS and PORV TSs are also proposed.

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

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

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

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

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

System/Component Unavailable	ICCDP	ΔCDF (per year)	
		1-In-3 yr. Entry	1-In-5 yr. Entry
SIT	1.31E-08	4.38E-09	2.63E-09
LPSI	1.18E-07	3.94E-08	2.36E-08
CS (no CARC available)	6.91E-07	2.30E-07	1.38E-07
PORV	8.07E-07	2.66E-07	1.60E-07
Boration Systems	4.46E-08	1.49E-08	8.93E-09

4.1.2 Methodology for Estimating CDP for Unavailability of Plant Control Equipment: Assessment of Risk Contribution of the unavailability of Class 1E Pressurizer Heaters

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

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

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

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

In this case, the unavailability of the Class 1E pressurizer heaters is assumed to increase the plant trip potential to 0.05 per day (a typical plant trip probability is normally about 1.5 per year or 0.004 per day). This is considered a conservative estimate in that many potential TS entries may not involve normal pressurizer heater capability (e.g., some entries may be influenced by the status of the emergency power supply) and situations which result in increased difficulty in maintaining and controlling pressure would directly result in plant shutdown. Given the availability of AFW and Emergency Diesel Generators, the conditional core damage probability following a normal plant high/low pressure trip is ≈ 6.0E-6 for a representative CE designed PWR (Reference 18). Substituting a value of 5.0E-2 per day (18.3 per year) for the assumed increase in plant trip potential and a value of 6.0E-6 for CDP/trip in the above expression, the probability of the loss of all pressurizer heaters causing a core damage event is approximately 3.0E-7 over a 24 hour period. Based on the incremental time of 23 hours (i.e., AOT-1), the ICCDP value becomes 2.9E-07. The associated changes in core damage frequency for losing all pressurizer heaters once-in-5 years and once-in-3 years are 5.8E-08 per year and 9.6E-08 per year, respectively. Therefore, as RCS pressure can be controlled manually, the risk of extending the AOT to 24 hours is acceptably small. Such a condition might be expected if non-Class 1E heaters are operational.

4.1.3 Comment on Uncertainty in CDPs

The two preceding methodologies utilized mean values of IEFs with a conservative assumption that system challenges proceeded to core damage. That is, operator recovery and/or actions and the availability of alternative mitigative systems are not credited. Overall, using the upper bound 95th percentile value for IEFs (Reference 7), as shown in Table 4.1-4, would increase the risk values presented in Table 4.1-2 by a factor of approximately four or less.

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

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

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

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

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

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

2. Severe Accidents Accompanied by Loss of Containment Isolation
   These events include any severe accident in conjunction with an initially unisolated containment.

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

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

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

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

4.2.1 Discussion of the ICLERP Model

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

The contribution of incremental core damage frequency is established from Section 4.1. Bounding estimates for ICLERP were developed by using a simplified LER event tree presented in Figure 4.2-1. The LER event tree sums the incremental contributions from: (a) containment bypass events (including

Inter-System LOCAs and induced SGTRs), (b) loss of containment isolation events, and (c) energetic containment failures.

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

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

RCS Pressure – High (RCSH)

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

Steam Generator Depressurized (SGD)

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

Thermally-Induced SGTR Occurs (TI-SGTR)

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

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

RPV Lower Head Failure Results in Containment Failure (DCH)

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

4.2.2 Supporting ICLERP Assumptions for ICLERP Quantification

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

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

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

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

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

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

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

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

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

4.2.3 ICLERP Quantification

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

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

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

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

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

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

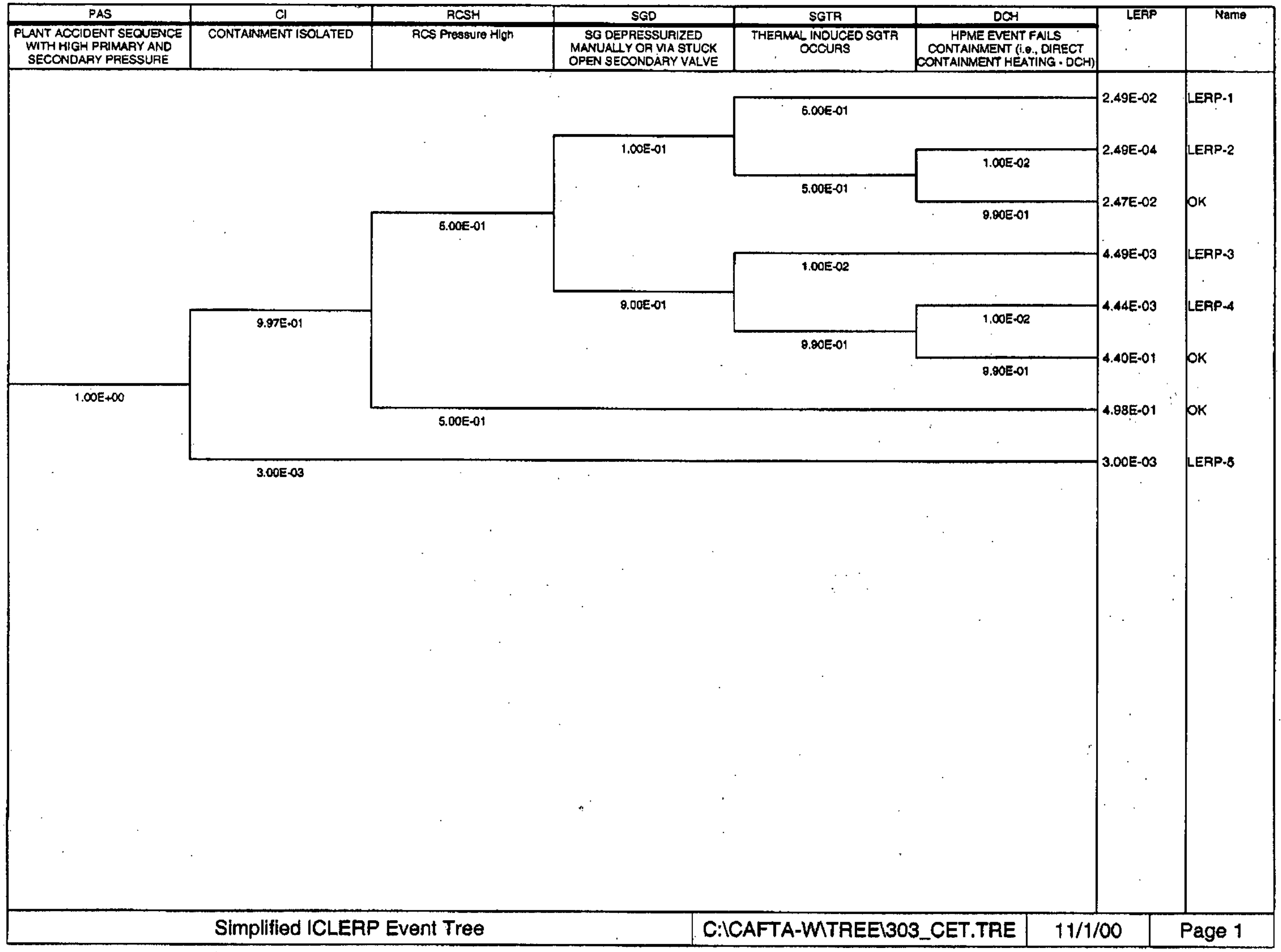

Figure 4.2-1: Simplified Incremental Large Early Release Event Tree

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

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

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

Notes for Table 4.2-1a
(1) See Section 4.1.2
(2) CLERP is defined as the conditional probability that a LER will occur following a core damage event.
(3) CARCS unavailable

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

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

where:

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

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

$$\Delta\text{LERF} = \text{ICLERP} \times f \qquad \text{(Eqn: 4-4)}$$

where:

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

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

Table 4.2-1b: Large Early Release Risk Impact

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

4.2.4 Incremental Conditional LERP Sensitivity Studies

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

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

A thermally-induced SGTR depends on the steam generator design, age, operating history and the time in cycle. Each factor or combination of factors may influence the likelihood of large early releases. In this evaluation, a conservative probability of 0.5 was assumed for failure of a steam generator tube prior to the failure of another RCS structural component (e.g., hot leg or surge line). The 50% SGTR failure probability was based on a severely degraded steam generator. This value also reflects analytical uncertainties which result in inconsistent predictions of this phenomenon. To address this uncertainty, a sensitivity evaluation was performed to determine the impact of variations in TI-SGTR on the large early release probability. This sensitivity involved varying the probability of TI-SGTR from 0.4 to 0.6 and then requantifying the simplified LER event tree to estimate the normalized LERPs for each system. Variations in the probability for TI-SGTR affect the probabilities of large early scenarios LERP-1 and LERP-2 (See Figure 4.2-1) for all CE designed NSSS plants. All of the other probabilities for the remaining large early scenarios are unaffected. The analysis focused on the change in conditional TI-SGTR probability. The results of this sensitivity evaluation are summarized in Table 4.2-2a, for the base case where the inoperable component is the pressurizer heaters. This scenario results in an inadvertent plant trip which has a small probability of leading to a core damage condition. The resulting plant damage state is assumed to be high pressure 50% of the time.

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

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

Note for Table 4.2-2a:
1. A bounding value of 0.01 is used in the calculations for Conditional Containment Failure Probability (CCFP) due to HPME.

Using the thermally-induced SGTR probability of 0.5 as the base case, the results in Table 4.2-2a indicate that the normalized CLERP increases approximately linearly as the thermally-induced SGTR probability increases.

(b) Bounding Assessment of Thermally-Induced SGTR

A bounding case was performed to assess the impact of setting LERP contributors to "bounding" values. For this case, it was assumed that a thermally-induced SGTR occurred. It was also assumed that containment isolation was much less likely to occur, and a containment isolation failure probability of 0.01 was used in place of the baseline value 0f 0.003. The probability of a dry (depressurized) steam generator is assumed to be as high as 0.3. This value is in the same range as the 0.27 value that was used in NUREG-1150. The results of this bounding evaluation are presented in Table 4.2-2b.

Table 4.2-2b: LERP Contributions based on use of Bounding Estimates (SGD=0.3, SGTR=1.0, CI=0.01)

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

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

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

System/Component Unavailable	ICCDP	ΔCDF (per year)		LERP	ICLERP	ΔLERF (per year)	
		1-In-3 yr. Entry	1-In-5 yr. Entry			1-In-3 yr. Entry	1-In-5 yr. Entry
SIT	1.31E-08	4.38E-09	2.63E-09	1.00E-02	1.31E-10	4.38E-11	2.63E-11
LPSI	1.18E-07	3.94E-08	2.36E-08	1.00E-02	1.18E-09	3.94E-10	2.36E-10
CS (no CARC available)	6.91E-07	2.30E-07	1.38E-07	7.23E-02	4.99E-08	1.66E-08	9.99E-09
PORV	8.07E-07	2.69E-07	1.60E-07	3.22E-01	2.60E-07	8.65E-08	5.19E-08
Boration Systems	4.46E-08	1.49E-08	8.93E-09	3.22E-01	1.44E-08	4.78E-09	2.87E-09
Pressurizer Heaters	2.88E-07	9.58E-08	5.75E-08	1.66E-01	4.77E-08	1.59E-08	9.53E-09

This analysis indicates that even if one to were to assume high conditional LERF values, the incremental risk expected to be incurred due to entry into to these exigent TSs would be small (< 10-7 per year, assuming a once in 5 year entry to the PORV TS). For many of the other TS changes, even the bounding risks are negligible.

(c) Steam Generator Depressurized (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 one or more MSSVs failing to close. A sensitivity evaluation was also performed on this parameter to determine the impact on the large early release due to the changes in the probability of a MSSV to close. This study involved varying the probability of a MSSV failing open from 0.05 to 0.2 and then requantifying the simplified LER event tree for a representative event and estimating the normalized LERP. Variations of the probability for a MSSV failing open affect the probabilities of large early scenarios LERP-1 through LERP-4 (See Figure 4.2-1). The probability of large early release scenario LERP-5 (containment isolation) is not affected. The results of this sensitivity evaluation are summarized in Table 4.2-3.

Table 4.2-3: Sensitivity Results for a MSSV Failing Open:
Core Damage Event Resulting from a Plant Trip Following the Unavailability of Pressurizer Heaters (1)

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

Note 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 reasonable basis for the assignment of risks associated with these TS changes and the impact is relatively linear.

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

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

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

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

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

System/Component Unavailable	ICCDP	ΔCDF (per year)		LERP	ICLERP	ΔLERF (per year)	
		1-In-3 yr. Entry	1-In-5 yr. Entry			1-In-3 yr. Entry	1-In-5 yr. Entry
SIT	2.63E-08	8.75E-09	5.25E-09	3.00E-03	7.88E-11	2.63E-11	1.58E-11
LPSI	2.89E-07	9.63E-08	5.78E-08	3.00E-03	8.66E-10	2.89E-10	1.73E-10
CS (no CARC available)	1.38E-06	4.60E-07	2.76E-07	1.68E-02	2.33E-08	7.76E-09	4.65E-09
PORV	2.00E-06	6.66E-07	4.00E-07	7.22E-02	1.44E-07	4.81E-08	2.89E-08
Boration Systems	6.56E-08	2.19E-08	1.31E-08	7.22E-02	4.74E-09	1.58E-09	9.48E-10

As indicate previously, when viewed from the perspective of incremental LERF use of 95% bounds on the initiating event frequency values propagated through the above nominal CET still results in small to negligible estimates of risk.

Final Comments

It should be noted that the ICLERP values presented in Table 4.2-1b are bounded by the ICCDP associated with each event. Using a ΔLERP goal of 1.0E-7/year, the goal is satisfied for the proposed AOT extensions.

4.3 ASSESSMENT OF OTHER DESIGN BASIS SYSTEMS

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

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

The primary purpose of the containment iodine controlled atmospheric cleanup systems is to ensure that accident radiation exposures meet GDC 19 and 10CFR100.11 limits. An assessment of the impact of the unavailability of these systems is presented below. The CREATCS is intended to ensure the post accident control room environment will allow operators and equipment to function within the control room. A discussion of the risk impact of the systems on public health is presented in the following sections.

4.3.1 Comments on the HVAC and Filtration Envelope Systems and ICS

The plant systems included in this section are primarily intended to provide environmental controls following an accident. Air cleanup systems include filtered ventilation paths that are designed to control radiation releases to the public. Control Room systems are designed to maintain a protected environment for operators. These systems are part of the plant radiation protection design basis and therefore they are designed to meet limits of 10CFR100.11 and/or GDC-19 (as appropriate). From a public risk perspective, availability of these systems have no impact on the plant core damage frequency or large early release frequency and as such are not specifically modeled in current generation PRAs. In fact, a large energetic failure of the containment would likely disable the SBEACS and PREACS. Thus, with respect to RG 1.174 metrics, the change in CDF, LERF, ICCDP and ICLERP resulting from the unavailability of these systems are all zero.

While these systems do not influence Quantitative Health Objective (QHO) metrics, they are intended to mitigate the offsite dose from design basis events in the presence of an intact containment and to manage the control room environment following on site radiation releases. The TS for several of these systems (for example 3.7.11 B.1 (CREACS), 3.7.13 B.1 (ECCS PREACS and 3.7.15 B.1 (PREACS), already include an extended AOT for system inoperability in the event of an inoperable barrier. In the presence of an inoperable boundary, these systems cannot perform their intended function and are inoperable. During that condition, the current TS provides a 24 hour allowed outage time/Completion Time for the boundary to be restored. The TS basis for the 24 hour period, as provided in NUREG-1432, is the "low probability of a DBA occurring during that time period." It was also noted that the 24 hour Completion Time is a "typically reasonable time to diagnose, plan and possibly repair, and test most problems" associated with the inoperability. The appropriateness of a 24 hour AOT/CT for assessing boundary impairments for control room systems is also discussed in TSTF-448, Revision 3, "Control Room Habitability."

This report proposes to extend the TS inoperability condition to a 24 hour AOT for system related inoperabilities resulting from two inoperable trains. This extended AOT is intended to be used for the purpose of diagnosis and repair of an inoperable system.

4.3.1.1 Discussion of system challenges

To understand the risk related issues associated with these systems it is useful to consider the plant challenges to which they may be exposed. Table 4.3-1 summarizes the expected iodine releases (as percentage of core inventory) for three categories of events: (1) Large Early Release from the containment, (2) Maximum Hypothetical Accident (per 10CFR100.11) and (3) Design Basis Accidents.

Radiological cleanup systems are intended to mitigate radiation releases following various design basis events. In order to better understand the role of these systems in accident mitigation it is useful to consider the spectrum of "at power" accident scenarios that lead to fuel damage and radiation release. Conceptually, three types of scenarios can be differentiated:

1. Beyond design basis scenarios that lead to Large Early Releases,
2. Maximum Hypothetical Accident,
3. LOCA and Non-LOCA Design Basis Accidents.

Beyond Design Basis Events Leading to Large Early Releases

From the perspective of radiation releases, the current generation of PRAs assesses the public risk of reactor operation using surrogate measures for the USNRC Quantitative Health Objective (QHO). The intent of the QHO is for the USNRC to have a quantitative means of measuring the acceptable level of public safety offered by nuclear power plants. These objectives, as applied to the nuclear industry, indicate that: (1) an individual's risk of an early fatality should be less than one tenth of one percent of the risk imposed from all other accidents, and (2) the increase in the risk of cancer posed by a plant should be less than 0.1 percent of the cancer risk from all other sources (Reference 22). In regulation, the QHO is supported via surrogate PRA measures related to the plant Core Damage Frequency (CDF) and Large Early Release Frequency (LERF). Events that result in large early releases to the environment are characterized by a severe unrecovered core damage condition with significant melting of the reactor fuel in the presence of an unisolated, bypassed or otherwise failed containment. Events considered as contributors to LERF typically result in nearly complete release of iodine from the fuel and iodine releases to the environment greater than 3% of the iodine core inventory (Reference 4). The frequency of PWR events that result in this level of core damage is less than 10^{-4} per year per plant. Associated LERF values are typically less than 10^{-5} per year. Current generation PWRs protect the public from significant releases of radiation via robust containment designs. For intact containment conditions containment sprays provide significant reduction of airborne radionuclide releases.

Maximum Hypothetical Accident

The ventilation systems under consideration are intended to mitigate radiation releases from reactor accidents. The systems are specifically designed to ensure 10CFR100.11 and GDC-19 criteria will be met following a Maximum Hypothetical Accident (MHA). The MHA can be viewed as a beyond design basis LOCA with a short duration uncovery of the fuel resulting from at least a temporary interruption or significant degradation of ECCS injection or recirculation. The event is a very low probability. The loss of cooling is assumed sufficient to result in significant iodine releases (40-50% of core inventory) from the fuel. In evaluating the effectiveness of containment cleanup systems to meet 10CFR100.11, radiation releases from the fuel to the containment are specified per TID-14844 (Reference 23) or NUREG-1465 (Reference 24) dependent upon the analyst's methodology. NUREG-1465 fission product releases represent a more realistic source term and are considered typical of what may be expected for a TMI-2 type event.

The intended use of these ventilation systems is to provide filtration of radionulide releases leaked from the containment into the auxiliary building or secondary containment, or radionuclides transported to the ECCS pump room and subsequently leaked and vaporized during the recirculation phase of a core damage event. MHA environmental releases are evaluated assuming an intact containment with maximum containment design leakage. In evaluating system performance it is assumed one train of CSS (or other containment atmosphere cleanup system) is operable. In estimating releases from the ECCS, the ECCS system is assumed to leak radioactive fluid recirculated from the containment sump at a rate of 1 gpm. (Current regulatory practice for MHA analyses is to increase this leakage by a factor of 2 to account for uncertainties).

From a design perspective, in aggregate, these cleanup systems are designed to confirm the maximum two hour dose to the public remains below the regulatory limit of 300 rem at the location of the Exclusion Area Boundary (EAB). An approximate assessment of the doses resulting from equipment inoperabilities indicate that, should an unmitigated LOCA unfold during the period of system inoperability, doses could exceed regulatory limits (see Table 4.3-1). However, when one considers realistic meteorological assumptions (i.e., X/Q values) doses would be less than regulatory limits. Note that as this event emerges from an unmitigated LOCA, much of the dose release results from ECCS recirculation leakage in the presence of an inoperable ECCS PREACS.

NUREG-1465 describes an MHA as being roughly equivalent to the TMI-2 event. Using this equivalency, the frequency of an MHA can be approximately established with reference to TMI-2.

According to NUREG 5750 from 1969 to 1997 PWRS and BWRs combined operated approximately 1500 reactor years. Since that time the United States nuclear industry logged on the order of 1000 additional reactor years. In that time only one event (TMI-2) emerged as a mitigated core damage event in the presence of an intact containment. Even if one assumes no improvement in reactor operation since that time the approximately frequency of an MHA can be estimated to be 1/2500 reactor years or 4 x 10^{-4} per year. When one considers improvements in procedures and operation of nuclear units in the past three decades and the positive operational experience of similarly designed international units, a more realistic estimate for the MHA would be closer to 1×10^{-4} per year.

LOCA and Non-LOCA Design Basis Events

While the containment, penetration room, and ECCS pump room ventilation systems, operating in conjunction with the containment sprays, have been designed to limit radiological releases following an MHA, these systems also contribute to dose mitigation resulting from a spectrum of design basis reactor accidents with potential fuel damage. These events include both LOCAs and non-LOCAs that are accompanied by successful core cooling. While these events lead to a successful conclusion from a PRA perspective, they may still result in failed fuel. Events considered in this category include design basis LOCAs, steam line breaks, locked rotor, and rod ejection accidents. Iodine releases from these events generally result from fuel gap releases from failed fuel rods (including the potential for localized fuel melting). Releases from such events vary from well under 0.1% of iodine inventory for LOCA scenarios to about 1% of iodine inventory for limiting non-LOCA transients[2]. In assuring adequate protection of plant staff, DBAs must also meet limits identified in 10CFR100 and GDC 19 or a fraction of those limits. As with the MHA the DBA analysis credits an intact containment with maximum design leakage. Note that the ECCS PREACS is only applicable to the LOCA events. In estimating releases from the ECCS, the ECCS system is assumed to leak radioactive fluid at a rate of 1 gpm with a factor of 2 uncertainty factor.

Events included in this category range from mitigated SGTRs (with a mean frequency of occurrence of less than 0.01 per reactor year) to very low probability locked rotor and control rod ejection scenarios.

4.3.1.2 Comparison of Relative Performance of Containment Systems

To understand the relative importance of various ventilation systems it is instructive to compare the various categories of accident with respect to curies released to containment and the environment given availability of various combinations of radiation cleanup systems. Table 4.3-1 presents results of a scoping study curies of iodine-131 released and thyroid doses based on a hypothetical 3800 Mwt PWR. The selection of I-131 as a surrogate for dose impact is generally consistent with design basis practice. The table calculations assume: (1) an iodine source term based on a high power core with 1 x 10^{8} curies of iodine (2) a containment with a leakage of 0.5 volume %/day (3) an ECCS leakage of 1 gpm (and factor of 2 uncertainty factor) and (4) bounding dose conversion factors based on limiting meteorological assumptions (i.e., X/Q = 10^{-3} sec/m^{3}). Note realistic dose estimates reflect X/Qs one and two orders of magnitude lower. In addition, the ECCS PREACS iodine loading also assumed a re-evolution factor of 0.10. In practice iodine re-evolution of iodine is expected to be far lower (typically less than a few percent). Additional assumptions regarding the effective decontamination factors for the systems included the following: (1) CS, ICS and SBEACS were each credited with a DF of 10. Since the PREACS does not filter leakage from the entire containment it was only credited with a DF of 2. The ECCS PREACS was credited with a DF of 20.

2 The iodine inventory estimate for LOCAs is based on 1% damaged fuel releasing iodine from the fuel gap containing 10% of the rod iodine. For non-LOCA events it conservatively assumed that the 10% of the fuel rods are damaged.

Several items should be noted in reviewing the table. Radiation releases to the environment arise from two sources: leakage from the containment and leakage from the auxiliary building as a result of leakage of vaporized leakage of recirculated sumpwater containing radionuclides. The above calculations includes consideration of SGTR releases in the evaluation of "containment leakage", however, none of the systems considered in Table 4.3-1 can mitigate such releases. For MHA and DBA releases (as defined above), radionuclide releases from the containment are relatively well controlled via the presence of an intact containment and the use of sprays. In fact, so long as sprays are available the impact of additional containment filtering capability such as that afforded by the SBEACS, PREACS and ICS are relatively small (less than 10% of the total release). In interpreting the Table 4.3-1 values it should further be noted that with regard to containment leakage contributions all assessments assume a 0.5 vol % per day containment leakage rate even though most CE plant designs will have containment leakage below 0.1% per day.

In the above calculations the ECCS PREACS is assumed inoperable. As a consequence, recirculation of the sumpwater following an accident dominates the environmental release, contributing in excess of 80% of the offsite dose. Radiation releases from the recirculation system arise as a result of re-evolution of I-131 from sumpwater leakage. As discussed above, in estimating these releases it is assumed that a 2 gpm leakage occurs and that the iodine partition factor is 0.10. Experience indicates that use of realistic partition factors and consideration of plate out of iodine in the auxiliary building would reduce unfiltered release predictions by more than a factor of 5.

A final factor in estimating public doses should be discussed and that is the selection of the atmospheric dispersion factor (X/Q). The table considers bounding (typical of design basis) and expected values of such factors. Bounding values typically will arise fewer than several days per year. More realistic estimates are expected to be between one and two orders of magnitude lower. Therefore, dose estimates are provided for both bounding and more typical atmospheric conditions.

4.3.1.3 Comments on the Auxiliary Building Cooling Functions of the Air Cleanup Systems

The various air cleanup systems include a ventilation and cooling capability. A review of the systems indicates that the primary purpose of these systems is to maintain the post accident humidity and temperature of the system charcoal filters. In general, availability of these systems ventilation capability is not explicitly required for cooling of other safety systems. Of particular interest is the role of cooling associated with the ECCS PREACS. The ECCS PREACS technical specification basis description indicates that these systems may also be used to support pump room cooling. One plant in the CE fleet noted that a portion of the ECCS PREACS is credited in SI pump cooling. For the plants proposing to implement this change, the ECCS PREACS or equivalent systems are not credited for cooling the ECCS pump room and do not factor into the SI pump operability determination. Several plants noted that ECCS rooms are cooled via "spot chillers".

4.3.1.4 Summary of Assessment of Other Design Basis Systems

As shown in Table 4.3-1, the results of these studies clearly indicate that accident releases anticipated under the operational conditions proposed by the TS change to the ICS, SBEACS, PREACS and ECCS PREACS are expected to be well below the LERF releases. Furthermore, offsite doses would be expected to be near or below regulatory limits for all but the rarest of events. Even for the MHA, when one assumes realistic meteorological conditions, doses are within regulatory limits for all but the most adverse combinations of equipment availability. Provided containment sprays are available, the more frequent (yet still low probability) LOCAs and non-LOCA transients that challenge fuel integrity are expected to result in 2 hour doses, at the EAB, below the regulatory limits for transient events.

Table 4.3-1: Comparison of Radionuclide Releases for Various Combinations of Mitigating Systems (Based on Inventory of 1E+8 Curies Iodine-131)

	Large Early Release associated with QHO	MHA (containment intact)	Releases due to DBA associated with limited fuel damage (containment intact and/or releases via steam generators)
Iodine Released from Fuel	100% of Core Inventory	40-50% of Core Inventory	< 1% of core inventory (non-LOCA) < 0.1% of core inventory (LOCA) (i.e., all or a fraction of the fuel gap inventory for damaged fuel)
Iodine releases to the Environment (24 hours)	2.5 to > 30%	< 0. 1%	< 0.002%
Approximate 2 hr curie release (no operable dose mitigation system)	2.5 to 30 x 10^{+6}	< 1 x 10^{+4}	< 2 x 10^{+2}
Approximate 2 hr curie release (only CSS operable) (all other radiation mitigation systems inoperable)	2.5 to 30 x 10^{+6}	< 3.5 x 10^{+3}	< 20
Approximate 2 hr curie release (CSS and SBEACS or PREACS or ICS operable)	2.5 to 30 x 10^{+6}	< 3 x 10^{+3}	< 10
Approximate 2 hr curie release (CSS and one train of ECCS PREACS operable)	2.5 to 30 x 10^{+6}	< 1 x 10^{+3}	< 20
Approximate 2 hr curie release (all mitigation systems operable)	2.5 to 30 x 10^{+6}	< 1.5 x 10^{+2}	< 10
EAB 2 HR dose based on bounding X/Q with sprays operational (REM Thyroid) (X/Q = 10^{-3})	> 10^{+6}	60 to 1300 depending on equipment availability	< 10
EAB 2 HR dose based on realistic X/Q with sprays operational (REM Thyroid) (10^{-5} < X/Q < 10^{-4})	10^{+4} to 10^{+5}	0.6 to 130 depending on equipment availability	< 1
Event Frequency (per year)	10^{-6}	10^{-4}	< 10^{-2} to 10^{-6}(3)

Notes for Table 4.3-1:
1 EAB –Exclusion Area Boundary
2. X/Q is the meteorological dispersion factor. Bounding values are based on 10^{-3} sec/m^3 while realistic values are one to two orders of magnitude lower. Specific dose conversion factor is 0.385 rem per Curie released; taking into account the assumed X/Q and breathing rate.
3. Higher frequency events refer to SGTRs, lower frequency events include rod ejection accidents and RCP locked rotor events.

4.3.2 Challenges to Control Room Ventilation and HVAC Systems

Control room ventilation and HVAC systems are designed to provide a protected environment for the reactor operator to control the plant following an uncontrolled release of radioactivity, chemicals, or toxic gas. Accident releases from the containment are largely mitigated via the containment sprays and the ventilation systems discussed above. While these releases will challenge the control room environment, operability of the ventilation systems significantly reduces the severity of the release.

The control room is also subject to radiation releases associated with steam line breaks and steam generator tube ruptures. These releases are particularly of concern during high specific activity (i.e., dose equivalent I-131 > LCO limit) plant operation. (Note: that TSs allow for a 48 hour period where plant specific activity for I-131 can significantly exceed the nominal operational value (see TS 3.4.16)). While operating with that LCO, the plant is required to demonstrate limits of 3.4.16-1 are not exceeded and to take activity samples. This mode of operation is accommodated in regulatory dose evaluations by including a short duration factor of five increases on the post accident iodine evolution. The activity source for this challenge is the primary coolant. For these other events, inoperability of both trains of control room ventilation system can potentially subject the plant staff to dose levels exceeding 5 rem and therefore actions to mitigate the potential challenge is important to supporting an AOT extension to 24 hours.

To support the temporary inoperability of the CR ventilation system, a review of the causes of radiological challenges to the CR was performed. Based on the review, it was noted that the severity of the potential challenge to the inoperable systems can be reduced by having the plant staff:

(1) Verify RCS Specific Activity is within the LCO 3.4.16 limit

and

(2) Implement mitigating actions such as:

- Increase plant awareness to the inoperability, verify operability of radiation monitors, and verify availability of mitigating actions, such as SCBAs.
- Protect operability of the containment spray and ventilation systems (See Section 4.3.1, as appropriate to each plant)

In evaluating actions, Item (1) above will be considered explicitly as conditions for entry into the technical specification. In addition, a condition will be added to the TS to implement mitigating actions. Note that the 24 hour AOT requested herein is consistent with TSTF-448 which provides for a 24 hour period to evaluate the impact of the inoperability of this system due to a breach of the control room envelope. The basis for that extension, like the basis for the requested extension, is the low probability of a DBA occurring in that time frame.

Control room temperature control systems are also used to control a post accident environment. Temperature control is important to ensure the operators can man and properly respond in the control room and to ensure mitigating equipment will function post accident. To ensure the risk of loss of post accident temperature control in the control room is negligible the plant must implement actions to ensure the control room can accommodate a post accident environment for a 24 hour period. These plant actions must be preplanned. Actions can include use of portable fans, temporary opening of doors, or use of normal HVAC. To support this change, administrative guidance will be provided to monitor the control room temperature to ensure habitability of control room and operability of TS equipment. In addition it will be noted in the bases that the plant should have administrative guidance in place to establish alternate cooling in the event of an accident or an evaluation of the control room which demonstrates the ability of the control room to remain habitable for a 24 hour period following an event. In establishing alternative cooling the plant should consider re-establishment of "normal HVAC" or use of portable fans. If compensatory measures requires that the plant temporarily violate the CR envelope the plant staff should verify that containment and auxiliary building post accident air cleanup systems discussed in Section 4.3.1 are operational.

Assessment of the Impact of the Proposed AOT on System Challenge Probability

It is important to note that challenge frequency or probability discussed in this section are not regulatory metrics, but rather a means of providing a quantitative measure of the term "low probability". The most significant expected challenge to these systems where the system significantly contributes to dose mitigation is the MHA. Using the above information one can estimate the expected challenge probability and consequences for the systems under consideration.

That is,

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

In this context the system Challenge Frequency (CF) is assumed to be equivalent to the significant containment radiation release frequency as measured by the frequency of the Maximum Hypothetical Accident. As discussed above, the MHA is generally analogous to a recovered core damage event.

Using this method, the reasonableness of the proposed 24 hour AOT for the ICS and components of the HVAC and filtration envelope (with the exception of the ECCS PREACS) can be demonstrated. It is assumed that these systems will be challenged during all core damage events (approximately 1.0E-4 per year, see discussion in Section 4.3.1.1). The resulting system challenge probability for these systems (assuming a 24 hour AOT) is presented in Table 4.3-2.

The role of the ECCS PREACS is different from the other systems in that it only is intended to function during recirculation operation. As a consequence, the system is in service when SI is available and functioning. Typically the ECCS PREACS is intended to cleanup airborne iodine volatilized from leakage from the ECCS pumps during the recirculation phase of an accident. These events limit iodine releases to a fraction of the iodine contained in the fuel rod gap of the damaged fuel rods. Since ECCS is generally operable, for credible events, radionuclide releases are expected to be well below MHA levels.

For purposes of estimating a challenge frequency for the ECCS PREACS system, it is assumed that recirculation is required for all large and medium LOCAs (4.5E-5 per year) and that all events, regardless of the presence of mitigation systems, result in significant radionuclide releases to the sump. The challenge was limited to these events since long term recirculation cooling is generally not needed for the higher frequency smaller LOCA breaks and design basis LOCAs of small size will have very limited fuel failure. Using the nominal LOCA frequency, the resulting system challenge probability is 1.2E-7. (Note that while containment spray operation will reduce the airborne concentration of fission products following a fuel damage accident, the spray operation offers no direct benefit to the ECCS PREACS operation as all the radionuclides scrubbed from the containment atmosphere are washed into the sump.) Radionuclide generation can be minimized by maximizing the availability of inventory makeup resources in the event of a LOCA. Availability of inventory makeup will reduce the extent of fuel damage and hence limit the radionuclide concentration in the sumpwater.

Table 4.3-2: Probability of MHA System Challenge Given Extended Entry into Proposed AOT

System	Proposed AOT for Inoperable System (hrs)*	Challenge Frequency (per year)	Probability of System Challenge Given Extended Entry into Proposed AOT
Iodine Cleanup System	24	1.0E-4 +	2.7E-7
Shield Bldg. EACS	24	1.0E-4 +	2.7E-7
CR EACS/EATCS	24	1.0E-4 +	2.7E-7
PREACS	24	1.0E-4 +	2.7E-7
ECCS – PREACS	24	4.5E-5 ++	1.2E-7
CS#	72	1.0E-4 +	8.1E-7

\+ Representative Bounding Estimate of Total Core Damage Frequency.
++ Frequency of entry into recirculation
* Both trains inoperable
\# With Emergency Grade CARC operable

4.4 TRANSITION RISK CONSIDERATIONS

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

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

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

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

4.5 END STATES AND SHUTDOWN RISKS

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

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

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

changes are specifically requested for LCO 3.6.6A and 3.6.6B, "Containment Spray and Cooling Systems," 3.6.8, "Shield Building Exhaust Air Cleanup System (SBEACS)", 3.6.10, "Iodine Cleanup System (ICS)," 3.7.13, "ECCS PREACS" and 3.7.15, "Penetration Room EACS". As these system's importance diminishes during shutdown, maintaining the plant in Mode 4 is not adversely affected by the component inoperability (compared to mode 5), and the Mode 4 is a more resource rich state, consequently, mode 4 operation is expected to be lower risk than Mode 5 operation for short duration outages.

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

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

4.6 MAINTENANCE RULE

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

4.7 10CFR20 CONSIDERATIONS

The amendment changes requirements with respect to the installation or use of facility components located within the restricted area as defined in 10CFR part 20. The TS changes proposed herein will not significantly change the types of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure resulting from the proposed AOT extensions.

5.0 SYSTEM EVALUATION

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

i) Description
ii) Plant Applicability
iii) Limiting Condition for Operation
iv) Licensing Basis for LCO
v) Condition Requiring Entry into a Shutdown Action Statement
vi) Proposed Modification to Actions
vii) Basis for Proposed Change
viii) Defense-in-Depth Considerations
ix) Tier 2 Restrictions
x) Tier 3 Recommendations

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

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

Consistent with Reference 4, in establishing the modified TS end state actions it was assumed that:

- The purpose of the Action is to complete a short duration repair of the inoperable system/component.
- When a Mode 4 end state is recommended, hot shutdown is the preferred means of heat removal since the steam generators are available for heat removal.
- Mode 5 end states are supported by a fully functional shutdown cooling system.
- Times for end state entry are as follows:
    - Transitions from Mode 1 to Mode 3 are required to be less than 6 hours.
    - Transitions from Mode 1 to Mode 4 are required to be less than 12 hours.

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

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

Conditions of entry, and compensatory measures and/or required actions to be taken when the LCO is not met are divided into prudent actions to be taken consistent with defense-in-depth considerations, avoidance of high risk conditions (Tier 2 requirements) and "good practices" (Tier 3 recommendations). RG 1.174 defines a Tier 2 condition as a potentially high-risk configuration that could exist if equipment in addition to that associated with the change were to be taken out of service simultaneously, or other risk-significant operational factors such as concurrent system or equipment testing were also involved. Tier 2 requirements are included as a TS action. In practice, none of the TS changes proposed in this package rigorously result in the high risk condition associated with the need for Tier 2 actions. However, in some instances defense-in-depth actions that can affect the significance of an accident while the plant is within the LCO are included within the Tier 2 actions sections to emphasize the importance of the action and the desire to include the action as a condition of entry or operation within the LCO. Tier 3 actions are generally associated with configuration risk management actions and are embedded within the plant maintenance rule program. In addition, in several instances explicit Tier 3 recommendations are provided where that information was considered important to include in the plant administrative guidance. When no Tier 3 recommendations are identified, it is tacitly assumed that the plant maintenance rule program includes actions to increase awareness of the unavailability of these systems, limit the duration of the outage and ensure the scope of these unavailabilities are considered in maintenance rule assessments.

5.1 STANDBY SAFETY SYSTEMS

5.1.1 (Non-ISTS) LCO 3.1.9 – Boration Systems - Operating

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

<u>Plant Applicability</u>

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.

Licensing Basis for LCO

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

Condition Requiring Entry into a Shutdown Action Statement

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

Proposed Modification to Actions

Increase the time available to take action to restore one boration flow path to operable status to 24 hours for the cases in which both boration paths are inoperable.

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), the Chemical and Volume Control System injection function is modeled only for small-small LOCA, SGTR and ATWS. The impact of charging flow on LOCAs and SGTRs is small, since both types of initiating events may be effectively mitigated via HPSI. However, HPSI is not an effective backup for ATWS events since ATWS events will rapidly repressurize above the HPSI shutoff head.

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

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

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

Defense-in-Depth Consideration

In the event that a loss of redundancy of charging pumps occurs, the impact on plant risk will be very small since boration (and injection) may be provided by other injection equipment (e.g., HPSI pumps) for many events. Therefore, the availability of HPSI during this interval ensures the plant Defense-in-depth is maintained.

Tier 2 Restrictions

None.

Tier 3 Recommended Actions

Upon entry into LCO take actions to verify availability of HPSI subsystem.

5.1.2 (ISTS) LCO 3.4.9 – Pressurizer Heaters

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

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

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

Plant Applicability

All except St Lucie-2

Limiting Condition for Operation (LCO)

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

Licensing Basis for LCO

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

Condition Requiring Entry into a Shutdown Action Statement

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

Proposed Modification to Actions

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

Basis for Proposed Change

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

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

Pressurizer heaters are beneficial in assisting the recovery from a SGTR and for pressure control during natural circulation cooldown to shutdown cooling entry conditions. 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 natural circulation cool down following an extended Loss of Offsite Power. Also, while the unavailability of pressurizer heaters may complicate post-trip cooldowns, a successful cooldown is expected with a minimal impact on plant risk due to the ability to control pressure by adjusting the cooldown rate, adjusting RCS make-up flow rate and cycling the RV head vents to eliminate RV head voids. During the cooldown, the operator is directed to continuously monitor for the presence of voids. Void formation may cause RCS pressure to remain high and prevent RCS cooldown. Any time it is found that voiding is inhibiting RCS depressurization, then an attempt at elimination of the voiding should be made. The EOPs include instructions to monitor for the presence of voiding and to eliminate them. In the case of a void in the reactor vessel, the pressurization and depressurization cycle will produce a fill and drain of the reactor vessel. The pressurization can be accomplished by lowering RCS cooldown rate or increasing inventory.

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

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

Defense-in-Depth Consideration

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

Without the emergency pressurizer heaters, a natural circulation cooldown may be required (as 20 °F subcooling may not be assumed). Such cooldowns may be conducted using RCS makeup, RCS heat removal (feeding and steaming a SG to the atmosphere) and RV head vents to minimize voiding in the RV head.

The NRC's November 3, 2006 letter to the TSTF on TSTF-426 requested that the Bases for the Pressurizer Required Action include a discussion on plant pressure control constraints if pressurizer level and temperature cannot be controlled, as stated in the associated section of the Safety Evaluation. That information will be added to TSTF-426.

Tier 2 Restrictions

None.

Tier 3 Recommendations

No specific tier 3 recommendations are applicable to this LCO.

5.1.3 (ISTS) LCO 3.4.11 PRESSURIZER PORVS & ASSOCIATED BLOCK VALVES

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

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

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

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

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

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

Plant Applicability

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

Limiting Condition for Operation (LCO)

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

Licensing Basis for LCO

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

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

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

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

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

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

Condition Requiring Entry into a Shutdown Action Statement

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

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

Proposed Modification to Actions

Revise ISTS 3.4.11 Condition E (or equivalent) CT to be consistent with other CE designed NSSSs (with PORVs) to allow 8 hours to restore one PORV to operable status for conditions where a PORV is unable to re-close once challenged, but may be isolated. 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. The LCO Bases state that a leaking PORV is inoperable. Therefore, Actions B or E would apply. Both Actions require closing the associated block valve. If the block valve cannot be closed, an immediate plant shutdown is required. Therefore, the existing TS enforce the Topical Report conditions that in order to apply the revised Actions, that a leaking PORV must be isolated by a block valve and that an inoperable PORV be isolable following a demand.

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

Basis for Proposed Change

The PORV functions as an automatic overpressure protection device and limits challenges to the Primary Safety Valves. However, overpressure protection is provided by the Primary Safety Valves, and the analyses do not take credit for the PORV opening for accident mitigation. Section 4.1 indicates that the increased CDP associated with extending the CT/AOT to 8 hours for inoperable PORVs (unable to open) is small, 8.1E-07 and is primarily attributed to the non-design basis safety function which credits the PORVs in feed and bleed cooling of the RCS following a total loss of feedwater event.

Defense-in-Depth Consideration

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

A core heat removal application of PORVs was identified post-TMI in the event of a total loss of feedwater. This feature is risk significant and impacts core damage. To minimize impact of this risk an action will be added to the TS to confirm LCO 3.7.5, Auxiliary Feedwater System, is met when the extended completion time is used.

Tier 2 Restrictions

The 8 hr AOT/CT does not apply to PORVs that are leaking and: (1) that cannot be isolated by block valves or (2) that are not expected to be isolable following a demand. This is true for plants that have requirements consistent with NUREG-1432. Plants with TS not consistent with NUREG-1432 should verify that these restrictions are implemented in their TS.

See defense-in-depth discussion for required concurrent actions.

Tier 3 Recommendations

No specific tier 3 recommendations are applicable to this LCO.

5.1.4 (ISTS) LCO 3.5.1 – Safety Injection Tanks

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

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

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

Plant Applicability

All

Limiting Condition for Operation (LCO)

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

Licensing Basis for LCO

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

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

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

Condition Requiring Entry into a Shutdown Action Statement

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

Proposed Modification to Actions

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

Insert a new Condition C to allow 24 hours for two or more inoperable SITs, provided both trains of ECCS are verified to be operable within one hour

Basis for Proposed Change

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

Defense-in-Depth Consideration

The unavailability of the SITs will compromise the ability of the plant to respond to Large Break LOCA events. In this same instance, the unavailability of 2 or more SIT(s) will result in an extended fuel heatup and affect 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. Event mitigation and long term core cooling will be assured via availability of the plant's LPSI and HPSI subsystems. It is proposed that the current requirement for an "immediate" response be extended to include the risk-informed interval of 24 hours. As a result of the low anticipated frequency of occurrence of a Large Break LOCA, a 24 hour CT/AOT to restore SIT operability is appropriate. At the end of this period, the operator will be instructed to exit the LCO either by resolving the condition causing the inoperability, or take actions to bring the plant to Mode 3 with pressurizer pressure < [700] psia.

The proposed CT/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 by ensuring the defined outage time is commensurate with the risk significance of the system. Availability of both LPSIs and both trains of HPSIs will limit the impact of SIT unavailability. Maintenance rule assessment per 10 CFR 50.65(a)(4) will ensure the integrated risk of this inoperability is small.

Tier 2 Restrictions

While the risk impact of SIT unavailability is small, use of this Condition will be restricted to when both ECCS trains are operable. Thus, in order to utilize the 24 hour AOT provision in the TS, upon entry into this Condition, the operator must "verify all ECCS trains are OPERABLE. Availability of both LPSI and HPSI trains will minimize the impact of the SIT unavailability and limit potential core uncovery times should a large LOCA event occur.

Tier 3 Recommendations

No specific tier 3 recommendations are applicable to this LCO.

5.1.5 (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 and 2 and MODE 3 with pressurizer pressure ≥ [1700] psia. Each train consists of a High Pressure Safety Injection (HPSI) subsystem and a Low Pressure Safety Injection (LPSI) subsystem.

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

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

The LPSI pumps also support the shutdown cooling system. However, this function is not considered within the scope of this technical specification. The shutdown cooling functions of the LPSI pumps are addressed by LCO 3.4 (RCS Loops) the RCS Loop specifications and requirements for RCS and SDC loop operability, which encompasses feedwater, cooling water, instrumentation and control, etc.

Plant Applicability

All

Limiting Conditions For Operation (LCO)

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

Licensing Basis for LCO

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

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

Condition Requiring Entry into a Shutdown Statement

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

Proposed Modification to Actions

It is proposed that the allowed outage time or completion time for the return to service of two inoperable LPSI subsystems be increased to 24 hours. Upon entry into the Required Action with the 24 hour Completion Time, the operator must verify operability of the SITs. In addition, in order to use the proposed Required Action, both HPSI subsystems must be OPERABLE.

Basis for Proposed Change

The design basis analysis requires that one subsystem of LPSI be available to suppress the peak fuel temperature heatup during a large LOCA event. Section 4.1 shows that the unavailability of the LPSI subsystems for this limited time interval will result in a contemporaneous increase in CDF of 4.5E-5 per year for the plant risk associated with large and medium LOCA events. There is no significant impact of the unavailability of LPSI following SGTR events as for many systems the LPSI would be required to be aligned to the SDC to affect entry into shutdown cooling. The risk impact of a plant shutdown with availability of the SDCS will offset any operational increase. A short term unavailability of the LPSI subsystems will result in a negligible incremental increase in the plant risk associated with large LOCA events.

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

Defense-in-Depth Consideration

The primary impact of the unavailability of the LPSI subsystems in Modes 1, 2 and 3 will be the reduction in the capability of the plant to provide RCS inventory makeup to accommodate a large LOCA. A twenty-four hour AOT/CT is proposed for this condition based on the low incremental plant risk associated with continued plant operation.

A best estimate CEFLASH-4AS analysis performed for a representative CE PWR indicates that LOCAs with break sizes up to the size of the largest Branch Line Pipe Break (BLPB) can be mitigated using SITs and one HPSI pump. These analyses credit use of the 1979 ANS decay heat standard, a nominal plant ASI and best estimate HPSI delivery curves. As CE PWRs have implemented Leak before Break (LBB) strategies random catastrophic failure of main coolant piping typical of that of the larger LOCAs is considered remote (less than 5.0E-6/year). The high reliability of this system has been reflected in allowing plants with these capabilities to be exempt from GDC 4 and allow internal component design to be based on BLPB sizes. In implementing this AOT, a Tier 2 restriction is applied to verify the availability of two trains of HPSI and all SITs. Additional availability of spare HPSIs (not available at all CE PWRs) and charging pumps should also be considered.

CE PWRs are equipped with leak before break detection capability. As LPSIs are primarily needed to mitigate the larger LOCAs, availability of leak before break detection should provide additional assurance of the very low potential of a rupture of main coolant piping.

Availability of alternate inventory makeup sources (such as charging pumps) and spare HPSIs (when they exist at a particular plant) can provide additional resources to mitigate the consequences of a LOCA and support long term core cooling.

Tier 2 Restrictions

The 24 hour AOT is restricted to conditions where the remaining components of the ECCS system are operable. Therefore, a required action to verify that all SITs are OPERABLE will be added to the TSs when two LPSI subsystems are inoperable.

If a HPSI subsystem is inoperable concurrent with the proposed Condition of both LPSI subsystems inoperable, Condition E (Less than 100% of the ECCS flow equivalent to a single OPERABLE train available for reasons other than Condition B) is applicable. Condition E directs entry into LCO 3.0.3.

Availability of SITs and HPSI subsystems will mitigate the unavailability of the LPSI subsystem and, in the event of a LOCA, the available systems will avert an extended core uncovery.

Tier 3 Recommended Actions

None.

5.2 HVAC AND RADIOLOGICAL CLEANUP SYSTEMS

HVAC and radiological cleanup systems provide the plant with the capability to protect the control room personnel and control radiological exposure to site personnel and the public. These devices are typically not credited for core damage mitigation/prevention and do not impact the probability of a large early release. There are ancillary impacts of these systems on some of these functions particularly those that protect Control Room (CR) staff. Furthermore, the control of long-term releases is an important design basis function. The proposed 24 hour AOTs for these systems were confirmed to be of low risk based on a zero CDF and LERF impact. Furthermore, the appropriateness of the 24 hour AOT/CT is consistent with the low probability of the beyond design basis and limiting DBA challenges.

5.2.1 (ISTS) LCO 3.6.8 Shield Building Exhaust Air Cleanup System (SBEACS)

The SBEACS provides radionuclide removal capability for fission products leaked into the shield building. The SBEACS consists of two separate and redundant trains. Each train includes a heater, cooling coils, a prefilter, a moisture separator, a High Efficiency Particulate Air (HEPA) filter, an activated charcoal adsorber section for removal of radioiodines in gaseous form, and a fan. Ductwork, valves and/or dampers and instrumentation also form part of the system. The alternate identification of this system for Waterford 3 is the Shield Building Ventilation System (SBVS).

<u>Plant Applicability</u>

St Lucie 1 & 2, Waterford 3 and Millstone 2.

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

Default entry to LCO 3.0.3.

<u>Licensing Basis for LCO</u>

The SBEACS is required to ensure that radioactive material leaking from the primary containment of a dual containment facility into the Shield Building (SB) (secondary containment) following a DBA is filtered and adsorbed prior to being exhausted to the environment. The SBEACS meets 10CFR50, Appendix A, GDC 41, "Containment Atmosphere Cleanup." The loss of the SBEACS could cause site boundary doses, in the event of a DBA, to exceed the values given in the licensing basis. Only the upstream HEPA filter and the charcoal adsorber section are credited in the analysis. The system initiates and maintains a negative air pressure in the shield building by means of filtered exhaust ventilation of the shield building following receipt of a Safety Injection Actuation Signal (SIAS).

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

Both trains inoperable.

<u>Proposed Modification to Actions</u>

Allow 24 hours to restore at least one SBEACS train when both SBEACS trains are inoperable and allow Mode 4 as the final end state if the action is not met.

Basis for Proposed Change

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

Changes in the unavailability of the SBEACS have no direct impact on ICCDP or ICLERP. This system does impact the magnitude of long-term radionuclide releases. The resulting risk-informed AOT is proposed to be 24 hours.

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

Defense-in-Depth Consideration

Modification of the TS to support a Mode 4 end state if the action is not met avoids the risks associated with an unnecessary mode transition and allows for the increased redundancy and diversity of RCS heat removal equipment in Mode 4. The implied licensing basis assumption that Mode 5 is inherently a lower operational risk than in Mode 4 is not supported by risk evaluations. Mode 5 risks are either about equal to or likely greater than equivalent Mode 4 risks 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 PRA insights, remaining in Mode 4 (vs. Mode 5) is an appropriate action while the SBEACS inoperability is corrected. This end state would maintain more mitigation systems available to respond to any event that could lead to a loss of RCS inventory or decay heat removal. Furthermore, in Mode 4 the SIAS and CIAS will be available to aid the operator in responding to events that threaten the reactor and/or containment integrity.

To minimize the impact of unavailability of the SBEACS, it is recommended that actions be taken to ensure availability of the CSS. Containment spray can effectively scrub the post accident containment atmosphere of fission products and therefore reduces the reliance on downstream air cleanup systems. Consistent with ensuring adequate defense-in-depth, an action will be added to the TS which upon entry into this Condition, the operator will confirm the operability of at least one train of the containment spray system. Availability of other plant air cleanup systems will further marginally reduce impact of SBEACS unavailability.

Tier 2 Restrictions

None. See the defense-in-depth discussion for concurrent required actions.

Tier 3 Recommendations

No specific tier 3 recommendations are applicable to this LCO.

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

The purpose of the ICS is to remove activity from the post-accident containment atmosphere. The system was initially installed based on the understanding that radiological iodine releases would be predominantly in elemental form. Decades of research have indicated that most iodine will be released in the form of cesium iodide (CsI) particulates. Consequently, the impact of the system on public doses is reduced. Airborne particulates will settle naturally; and within 24 hours, the sedimentation will be nearly complete. Operation of the system will enhance particulate removal, but particulate removal will occur within a 24 hour time-frame even without it.

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

Plant Applicability

Calvert Cliffs 1 & 2, St Lucie 1 & 2

Limiting Conditions for Operation (LCO)

Default entry into LCO 3.0.3.

Licensing Basis for LCO

For several PWRs, the ICS contributes to meeting 10 CFR 100 (Reference 9) siting requirement dose limits and supports General Design Criteria (GDC)-19 of 10 CFR 50 Appendix A (Reference 10) for Control Room (CR) doses. These design basis calculations assume a high concentration of elemental iodine in the fission product release (See References 11 and 12). Two ICS trains are provided to meet the requirement for separation, independence and redundancy. The moisture separators function to reduce the moisture content of the airstream. The ICS also supports meeting GDC 41, "Containment Atmosphere Cleanup". Note that if the NUREG-1465 source term (Reference 13) has been adopted as an Alternative Source Term (AST), these systems may still be credited for particulate removal (and for the removal of a small amount of elemental iodine).

Condition Requiring Entry into Shutdown Action Statement

Both ICS trains inoperable.

Proposed Modification to Actions

Add a condition which allows 24 hours to restore one ICS train when both ICS trains are inoperable before requiring a shutdown. Allow Mode 4 as final end state if the action is not met.

Basis for Proposed Change

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

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

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

Modification of the TS to support a Mode 4 end state if the action is not met avoids the risks associated with an unnecessary mode transition and the increased redundancy and diversity of RCS heat removal equipment relative to that required in Mode 4. The implied licensing basis assumption that Mode 5 is inherently a lower operational risk than in Mode 4 is not supported by risk evaluations. Mode 5 risks are either about equal to or likely greater than equivalent risks in Mode 4 and therefore produce radiation releases to containment on par with those of Mode 4. Furthermore, plant shutdown actions that require entry into Shutdown Cooling (SDC) introduce potential containment bypass risks including LOCAs. Thus, based on these PRA insights, remaining in Mode 4 (vs. Mode 5) is an appropriate action while the ICS inoperability is corrected. A Mode 4 end state would maintain more mitigation systems available to respond to any event that could lead to a loss of RCS inventory or decay heat removal. Furthermore, in Mode 4 the SIAS and CIAS will be available to aid the operator in responding to events that threaten the reactor and/or containment integrity.

Defense-in-Depth Consideration

Containment sprays can effectively scrub the post accident containment atmosphere of fission products and therefore reduces the reliance on downstream air cleanup systems. Consistent with ensuring adequate defense-in-depth, an action will be added to the TS which upon entry into this Condition, the operator will confirm the operability of at least one train of the containment spray system. Availability of other plant air cleanup systems will marginally reduce impact of system unavailability.

Tier 2 Restrictions

None. See the defense-in-depth discussion for concurrent actions.

Tier 3 Recommendations

No specific Tier 3 recommendations are applicable to this LCO.

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

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

Plant Applicability

All

Limiting Condition for Operation (LCO)

In Modes 1, 2, 3, 4, [5 and 6] and during movement of [recently] irradiated fuel, the CREACS must be operable to limit operator exposure during and following a DBA. In Mode 5, the CREACS is required to cope with the release from a rupture of an outside waste gas tank or external toxic gas challenges. During movement of irradiated fuel assemblies [and CORE ALTERATIONS], the CREACS must be OPERABLE to cope with the release from a fuel handling accident.

Licensing Basis for LCO

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

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

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

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

Condition Requiring Entry into a Shutdown Action Statement

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

Proposed Modification to Actions

Note that the existing TS allows for a 24 hour inoperability of the CREACS resulting from an inoperability of the boundary. This inoperability renders the CREACS system inoperable as the leakage may be in excess of that allowed for adequate event mitigation. This inoperability has been judged acceptable based on the low probability of the DBA and implementation of compensatory measures.

It is recommended that a similar 24 hour period be provided for conditions other than boundary inoperabilities.

Basis for Proposed Change

Operation of the CREACS has no direct impact on CDF and LERF as analyzed in the plant's PRA. Personal operator radiation protection equipment is available to the operators. SCBAs provide a partial radiation protection or chemical protection backup suits and can help to control post-accident exposure. Furthermore, the probability that an event would occur challenging this system during the extended outage period is very low.

Defense-in-Depth Consideration

The CREACS provides a protected environment from which operators can control the plant following an uncontrolled release of radioactivity, chemicals or toxic gas. The CREACS is designed to ensure that the whole body dose to the operators following a DBA is < 5 REM or the equivalent to any part of the body. To maximize protection to the operators under conditions with an inoperable CREACS, mitigating actions should be taken to lessen the effect on the CRE occupants from potential hazards. This would include actions to minimize the potential for a significant release of radioactivity and to ensure alternate operator protection is available. During the period that the CREACS trains are inoperable, the technical specifications will require that action be initiated to implement mitigating actions to lessen the effect on CRE occupants from the potential hazards of a radiological or chemical event or a challenge from smoke (for example, restricting transport of hazardous materials, verifying availability of radiological monitoring instrumentation, and verifying availability of operator personal radiological protection systems).

As SGTR events are a significant contributor to potential CRE radiological hazards, entry into this condition will require verification that LCO 3.4.16 "RCS Specific Activity" is met.

Implementation of paragraph (a)(4) of 10 CFR 50.65 (Maintenance Rule) will assure proper plant configuration control. Other technical specifications (e.g., 3.3.8 (analog), 3.3.9 (digital)) require the availability of equipment to identify the onset of a radiological challenge to the control room (or if applied to non-radiation atmospheric cleanup, a toxic gas release). TS 3.3.12 requires the availability of alternate shutdown panels and local shutdown stations should remote actions become necessary.

Tier 2 Restrictions

Upon entry into the condition, the operator will confirm that the RCS specific activity is within the LCO 3.4.16 limit. Operation within the LCO 3.4.16 will limit potential environmental releases should a SGTR occur during this outage period. In addition, the TS will contain a requirement to implement mitigating actions. Mitigating actions are intended to lessen the effect on CRE occupants of a CREACUS system inoperability resulting from the potential hazards of a radiological or chemical event or a challenge from smoke. These actions will be captured in administrative guidance. Such actions include restricting transport of hazardous materials, ensuring the OPERABILITY of at least one Containment Spray train, verifying availability of radiological monitoring instrumentation, verifying availability of operator personal radiological protection systems. In the event of a DBA, the mitigating actions will ensure that the exposure of CRE occupants will be controlled to the maximum extent practical and that CRE occupants are protected from exposure to hazardous chemicals and smoke.

Tier 3 Recommendations

None.

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

The CREATCS provides temperature control for the control room following isolation of the control room. The CREATCS consists of two independent, redundant trains that provide cooling and heating of recirculated control room air. Each train consists of heating coils, cooling coils, instrumentation and controls to provide for control room temperature control. (Note that for several plants the control room temperature and cleanup functions are combined in a single system).

Plant Applicability

Fort Calhoun, Palisades, PVNGS 1, 2 & 3, Waterford 3, San Onofre 2 & 3 and ANO 2.

(Note: St. Lucie Units 1 & 2 and San Onofre Units 2 & 3 include this cooling function within the air cleanup system. A discussion of the CR cooling functions of those systems are included in this section.)

Limiting Condition for Operation (LCO)

Two CREATCS trains shall be OPERABLE in Modes 1, 2, 3, 4, [5 and 6] and during movement of [recently] irradiated fuel assemblies.

Licensing Basis for LCO

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

Condition Requiring Entry into a Shutdown Action Statement

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

Proposed Modification to Actions

Modify Condition E to allow 24 hours to restore one CREATCS train to operable status before requiring a plant shutdown.

Basis for Proposed Change

Provided actions are in place to provide alternate cooling, unavailability of the CREATCS will not directly impact on ICCDP and ICLERP. Some plants have analyses that demonstrate that post-accident control room heatup to levels will not challenge containment personnel or mitigation equipment for periods up to 24 hours.

In addition to the above, it should also be noted that the probability that an event would occur challenging this system during the extended outage period is very low.

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 Completion Time is based on the low risk of system inoperability, compared to the associated risks of plant shutdown. In the event that inoperability of the system the plant staff would monitor CR temperature to ensure the CR remains habitable and that electrical cabinets are not exposed to excessive temperatures. This could include actions such as use of normal (i.e., non-safety) ventilation systems, opening cabinet doors, use of fans or ice vests, and opening CR doors or ventilation paths. Note that if the control room envelope is breached in order to provide CR cooling, the Actions of Specification 3.7.11, CREACS, would also apply.

Tier 2 Restrictions

None. See defense-in-depth discussion for required concurrent actions.

Tier 3 Recommendations

Administrative guidance/procedures should be in place for alternate means of establishing temporary CR cooling. See Defense-in-depth discussion for typical actions.

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

The TS definition of the ECCS PREACS notes that the system "filters air from the area of the active ECCS components during the recirculation phase of a loss of coolant accident. The ECCS PREACS, in conjunction with other, normally operating systems, also provides environmental control of temperature and humidity in the ECCS pump room area and the lower reaches of the Auxiliary Building. The primary function of the ECCS PREACS is to filter volatized iodine resulting from leakage of post-accident recirculation fluid. The ECCS PREACS consists of two independent, redundant trains of equipment that provide filtering of air in the ECCS pump rooms during post-LOCA recirculation cooling.

<u>Plant Applicability</u>

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

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

Two ECCS PREACS trains shall be operable in Modes 1, 2, 3 and 4.

<u>Licensing Basis for LCO</u>

ECCS PREACS filters volatized radioiodine released from leaks in the ECCS pumps following reactor accidents that lead to ECCS recirculation. The ECCS PREACS is typically credited in evaluating the ability of the plant to meet 10CFR100 and Appendix A GDC-19 radiation dose limits (or the equivalent limits when the alternative NUREG-1465 source term have been adopted). Design basis dose calculations assume a stylized radiological source term associated with a damaged, but recovered core (radionuclide distribution dependent upon methodology selected). All released radionuclides are assumed instantaneously swept into the sump, recirculated, and leaked at an assumed rate, usually 1 gpm.

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

Both ECCS PREACS trains inoperable for reasons other than an inoperable PREACS penetration room boundary.

<u>Proposed Modification to Actions</u>

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

<u>Basis for Proposed Change</u>

While the ECCS Pump Room Emergency Air Clean-up System (PREACS) affects the magnitude of post-accident radioiodine releases, operation of ECCS pump room PREACS has no direct impact on ICCDP and ICLERP as analyzed in the PRA. The ECCS PREACS is only expected to be used during long-term mitigation of the larger LOCAs, where long-term cooling is afforded via recirculation cooling.

This system is not available at all CE PWRs. Some plants do credit portions of the ECCS PREACS to support pump room cooling. Those plants are not currently pursuing this AOT extension. Waterford 3 indicated that it does not rely on this system for pump room cooling and therefore, the system does not provide a risk-significant function at their plant. That is, failure of this system will not prevent post-accident ECCS pump cooling. Pump room cooling is accomplished via independent systems including "spot" chillers. St. Lucie units also indicate that they do not require these systems to be operable to maintain ECCS pump room cooling. Plants implementing this TS change must confirm that ECCS PREACS cooling functions are not credited in the plant design basis or PRA.

Regardless of the system status, the risk of Mode 4 is lower than (or equivalent to) the Mode 5 operating state since more mitigating systems are available in Mode 4 to respond to an event and there are additional risks associated with the transition to Mode 5 from Mode 4. Since the risk of a transition to SDC and subsequent Mode 5 operation is greater than that incurred by continued operation in Mode 4, and the likelihood of a LOCA initiated from Mode 4 is low, restoring the ECCS PREACS in Mode 4 is preferred.

A 24 hour AOT/CT for this inoperability is reasonable based on the low probability of a DBA.

Defense-in-Depth Consideration

The ECCS PREACS is specifically designed to filter volatilized radioiodine leaked from the ECCS system that may occur within the pump room. Leakage is estimated at 1gpm. Since the system is focused on the recirculation mode of ECCS operation, operation of the ECCS PREACS can only impact radiation releases to the public when ECCS recirculation is in progress. While no system can specifically back up the ECCS PREACS, the significance of the system can be reduced by taking actions that minimize fuel damage following a LOCA and by ensuring control room air cleanup systems are in place. Actions to verify concurrent availability of LOCA mitigation equipment (LPSI, HPSI, etc.) will reduce the impact of ECCS PREACS unavailability by reducing the extent of core damage and the associated radiological load within the sump. Control of the sump water pH above 7.0, will also reduce the impact of ECCS PREACS availability. The ECCS PREACS is designed to mitigate a core damage event with iodine releases on the order of 50% of the iodine inventory. More realistic DB LOCA events will have far less fuel damage (< 1%) and with all equipment available, releases could be far less. More realistic credit for retaining radioiodine in the leaked sump water (including recognition of the positive aspects of fission product cesium in raising sump water pH) would also point to much smaller releases of volatile radioiodine in the ECCS pump rooms.

On site impact of the release could be mitigated by verifying the Control Room Emergency Air Cleanup System (CREACS) is operable.

Extension of the ECCS PREACS AOT/CT to 24 hours provides time to restore component operability during power operation. This may potentially avert a plant shutdown and the associated transition risks.

The plant risk due to ECCS PREACS unavailability is negligible. Due to defense-in-depth considerations, the TS will require verification of the operability of one train of CREACS. Availability of the CREACS during this timeframe ensures that any ECCS volatilized releases will be filtered prior to entering the control room.

Tier 2 Restrictions

None. See defense-in-depth for required concurrent actions.

Tier 3 Recommendation

Design basis assessments for this system assume radiological releases based on a severe core damage condition. Availability of the ECCS will minimize the potential fuel damage from such an event and consequently will limit the radioiodine in the recirculated sumpwater. This will minimize the impact of the ECCS PREACS inoperability. The ability to keep the sump water pH > 7.0 will reduce the volatility of any radioiodine that may be present. As a result of the low likelihood of random simultaneous outage of these components and the onset of a concurrent event that could impact the ECCS PREACS, explicit consideration of this action for unanticipated occurrence is not considered necessary.

Administrative guidance will be provided in the maintenance rule procedures to note the importance of LCO 3.5.2, "ECCS Operating" and LCO 3.5.5, "Trisodium" when in this ECCS PREACS condition.

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

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

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

Plant Applicability

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

Limiting Condition for Operation (LCO)

Two PREACS trains shall be operable in Modes 1, 2, 3 and 4.

Licensing Basis for LCO

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

Condition Requiring Entry into Shutdown Action Statement

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

Proposed Modification to Actions

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

Basis for Proposed Change

While the PREACS affects the magnitude of the post-accident radionuclide releases, operation of the PREACS has no direct impact on ICCDP and ICLERP as analyzed in the PRA.

There is no significant risk impact of extending the potential system inoperability to 24 hours (see Table 4.3-1). The system does not provide a preventive function with respect to core damage events. As the PREACS filters containment leakage into the penetration room, an operable containment spray system will limit the radiological load on the system. Table 4.3-1 indicates that post accident, the presence of an operational CSS will control the containment releases. Unavailability of the PREACS will not have a risk significant impact on radiological releases to the public or to the control room.

Regardless of the system status, the risk of Mode 4 is lower than (or equivalent to) the similar Mode 5 end state, since more mitigating systems are available in Mode 4 to respond to an event and there is additional risk associated with the transition to Mode 5 from Mode 4. Since the risk of a transition to SDC and subsequent Mode 5 operation is greater than that incurred by continued operation in Mode 4, restoring the PREACS while in Mode 4 is preferred.

The 24 hour AOT/CT extension is supported by the low probability of a DBA during the period of inoperability.

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. To minimize the impact of inoperability of the PREACS, the operator should ensure availability of at least one train of Containment Spray System to mitigate leakage of airborne radionuclides from the containment. Availability of other plant cleanup systems will also marginally contribute to dose reductions.

The current TS requires operability of PREACS from Modes 1 through 4.

To minimize the impact of unavailability of the PREACS, it is recommended that actions be taken to ensure availability of at least one train of Containment Spray System.

Tier 2 Restrictions

None. See defense-in-depth discussion for required actions.

Tier 3 Recommendations

None.

5.2.7 (ISTS) LCO 3.6.6A & B Containment Spray System & LCO 3.6.6.1 Containment Sprays/ Coolers

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

Plant Applicability

All.

Limiting Conditions for Operation (LCO)

Two containment spray trains and two containment cooling trains shall be operable in Modes 1, 2, 3 and 4.

Licensing Basis for LCO

The Standard Technical Specifications (STS) requirements of NUREG-1432 distinguish between containment spray systems that are credited in containment activity removal and containment spray systems that are not credited in containment activity removal (ISTS 3.6.6A and 3.6.6B, respectively). The Required Actions for restoring inoperable containment spray systems that are not credited for activity removal are less stringent than the requirements for containment spray systems that are credited for activity removal. The LCO 3.6.6A Required Actions are revised to be consistent with the changes being proposed to the LCO 3.6.6B Required Actions.

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

Some plants include dedicated Iodine Cleanup Systems (ICS) consisting of recirculation filter units. These units are separately discussed in Section 5.2.2.

Condition Requiring Entry into a Shutdown Action Statement

Inoperability of both Containment Spray trains.

* Also known as Containment Air Recirculation Coolers (CARCs). The designation of the fan cooler system at Waterford 3 is the Containment Fan Cooler System (CFCS).

Proposed Modification to Actions

For LCO 3.6.6.A, increase the time available to initiate shutdown to 72 hours when the Containment Spray system is inoperable and at least one train of CACs is operable.

For LCO 3.6.6.A and 3.6.6.B, extend the time available to initiate shutdown to 12 hours when the CS system is inoperable and both trains of CAC are inoperable for containment heat removal. See Tables 5.2.3-1 and 5.2.3-2 for details.

Basis for Proposed Change

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

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

Containment Spray and CAC are used to support long-term containment heat removal. This heat removal is needed to ensure that the ECCS recirculation mode can continue to effectively remove decay heat. Containment analyses performed for San Onofre indicates that successful containment heat removal occurs when at least one CS train <u>or</u> one CAC operates. Consequently, a minimum containment heat removal capability is required to ensure both long-term containment integrity and core damage prevention. Containment Spray and CAC are also considered in the PRA 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 activity removal. Therefore, the unavailability of the CS system will compromise both post-accident containment integrity and ECCS recirculation cooling. To limit the risk associated with the loss of the containment/sump cooling function of the CSS, the Completion Time for both trains inoperable is limited to 12 hours.

Risk-Informed Assessment

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

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

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

Based on representative plant analyses performed in support of PSA containment success criteria, containment integrity may be established via use of a single fan cooler as documented in the SONGS 2 & 3 PRA. Qualitatively, similar conclusions could be drawn for one train of CS. Consequently, in Mode 4 one train of CAC or one train of CS assures adequate heat removal capability.

For the case of CACs and CS unavailable, Table 4.1-2 indicates a CDP impact of 7.5E-7 for a 12 hour unavailability. ICLERP impacts will also be acceptable since these systems have a limited impact on prevention of early containment failures. As a result of the low challenge frequency, infrequent expected use and short exposure time, 12 hours provides a reasonable AOT and results in a sufficiently low risk impact.

The 72 hour AOT for conditions where containment heat removal can otherwise be maintained is appropriate in recognition of the low likelihood of system challenge. This AOT is similar to that already granted in TS 3.6.6.B. As sump cooling will be supported by containment fan coolers, the risk impact during the operational mode is negligible.

Defense-in-Depth Consideration

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

Risk-significant impacts of unavailability of both systems included the inability to cool the containment sump and to remove heat from the containment. Loss of sump cooling will result in failure of ECCS sump recirculation mode of operation. Unavailability of containment heat removal following a LOCA or other inventory release sequence will result in a long-term challenge to the containment. However, cooling to the core will be maintained and core damage may be averted provided inventory is available to the RWST. The impact of loss of recirculation cooling is mitigated by procedures to refill the RWST. Such procedures are available through implementation of Severe Accident Management Guidance.

Tier 2 Restrictions

None.

Tier 3 Recommended Actions

Note that plants implementating this TS change have proceduralized actions, equipment and inventory for refilling the RWST.

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

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

Note for Table 5.2.3-1:
1. Default and implicit actions result in 3.0.3 or equivalent entry.

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

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

* This Completion Time should be extended to 7 days based on similarity to 3.6.6B and the risk-based Completion Time for two trains inoperable.

6.0 SUMMARY

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

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

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

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7.0 REFERENCES

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5. NUREG-1570, "Risk Assessment of Severe Accident Induced Steam Generator Tube Rupture," USNRC, March 1998.
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8. CE NPSD-1045-A, "Joint Application Report: Modifications to Containment Spray System, and the Low Pressure Safety Injection System Technical Specifications," CE Owner's Group, March 1998.
9. 10 CFR 100, "Reactor Site Criteria," USNRC, 1991.
10. 10 CFR 50, "Appendix A: General Design Criteria 79," USNRC, 1991.
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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.
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14. Not used.
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17. Not used.
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20. Arkansas Nuclear One – Unit 2 Letter, 2CAN030003, "Proposed License Change For Cycle 14 Risk-Informed Operation," from Craig Anderson (ANO-2) to U.S. Nuclear Regulatory Commission, dated March 9, 2000.

21. Westinghouse Letter, LTR-ESI-02-39, "Response to NRC RAIs Concerning Topical Report CE NPSD-1208, Justification of Risk-Informed Modifications to Selected Technical Specifications for Conditions Leading to Exigent Plant Shutdown" (TSTF-360), dated February 20, 2002.
22. Safety Goals for the Operation of Nuclear Power Plants,, Federal register (51 FR 20844), August 4, 1986
23. TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," U.S. Atomic Energy Commission, March 23, 1962. [8202010067]
24. NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," U.S. Nuclear Regulatory Commission, February 1995.

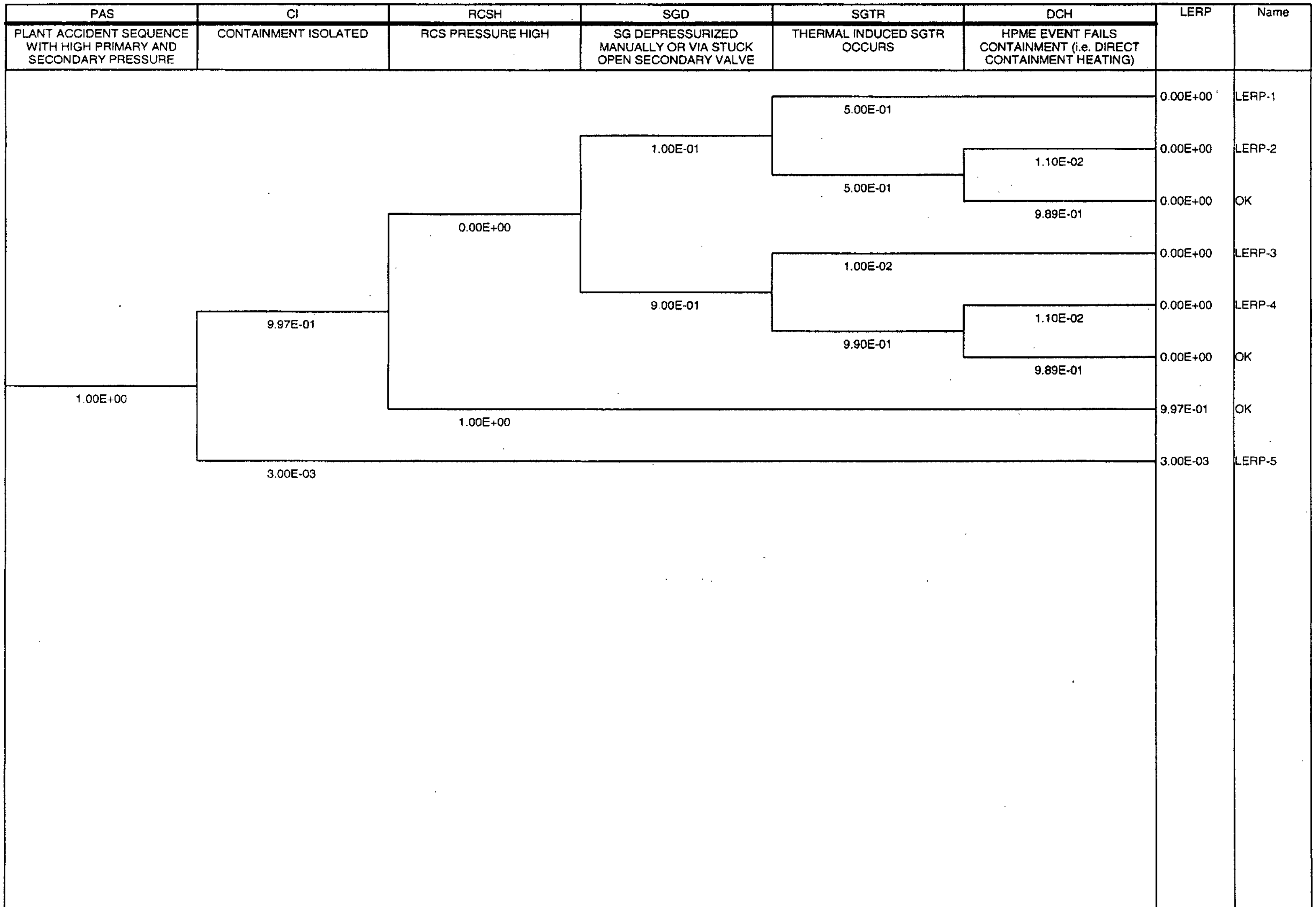
APPENDIX A

Technical Specification Cross-Reference

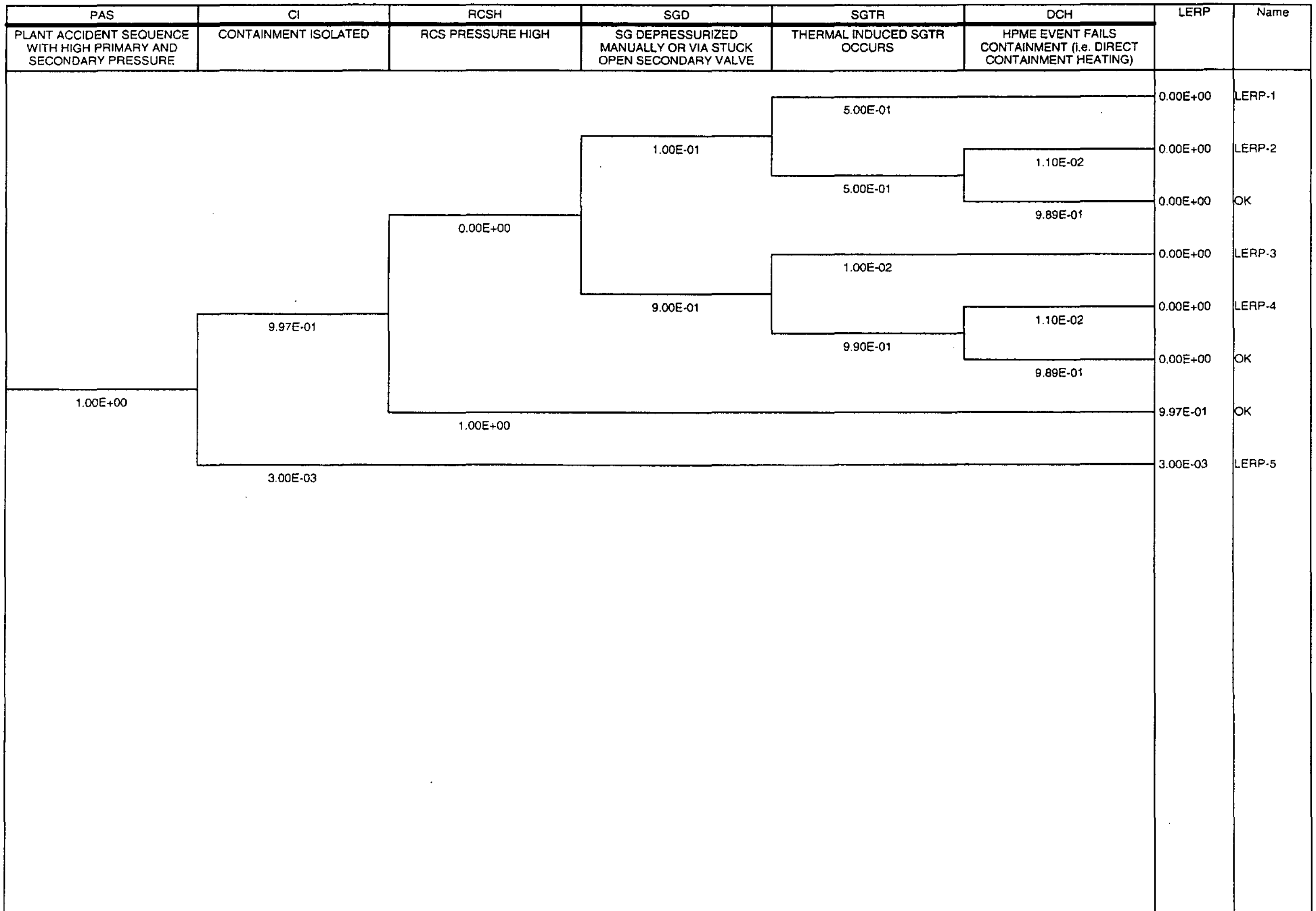
This Appendix contained archival information that has been deleted.

APPENDIX B

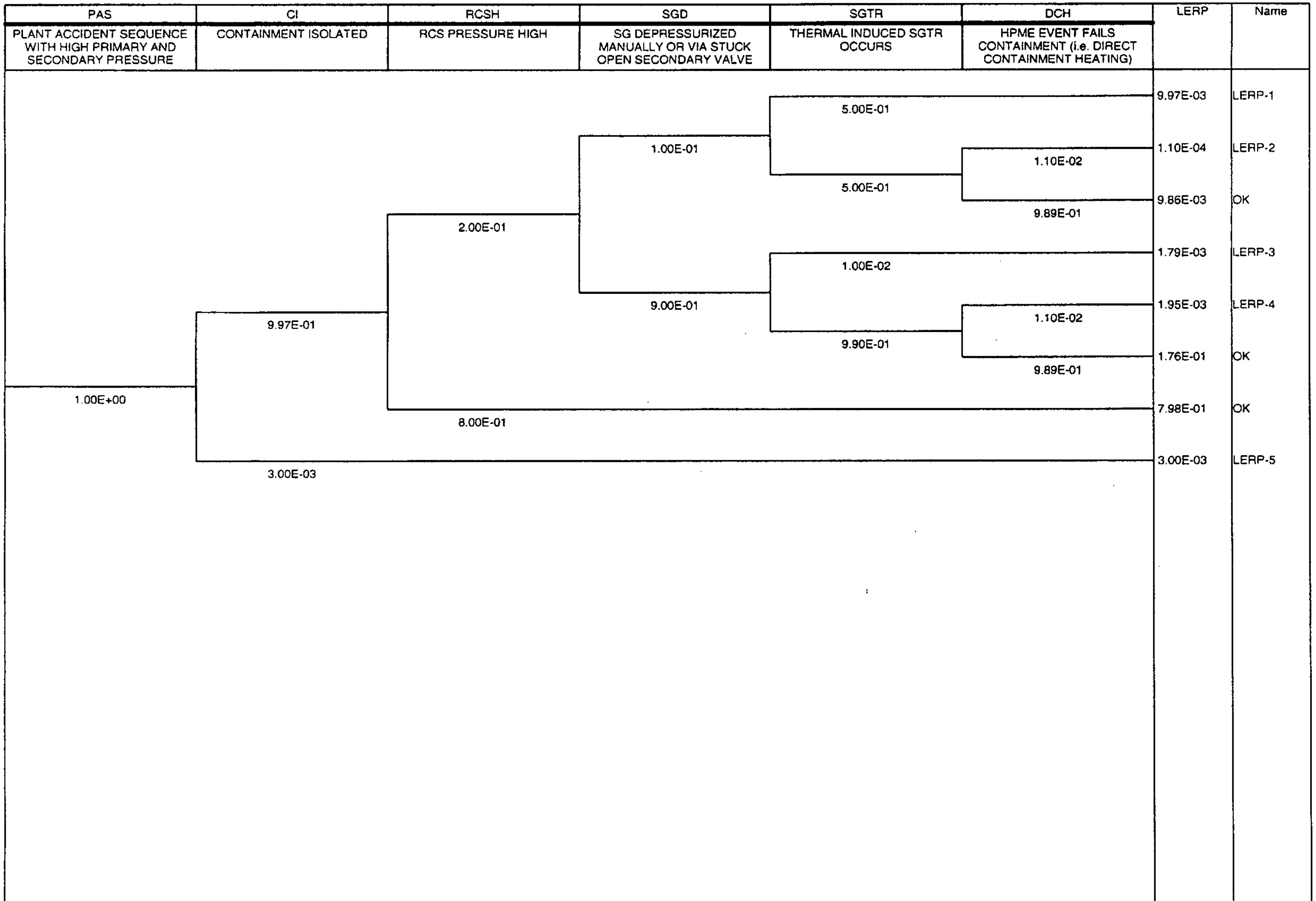
System Specific LER Event Trees



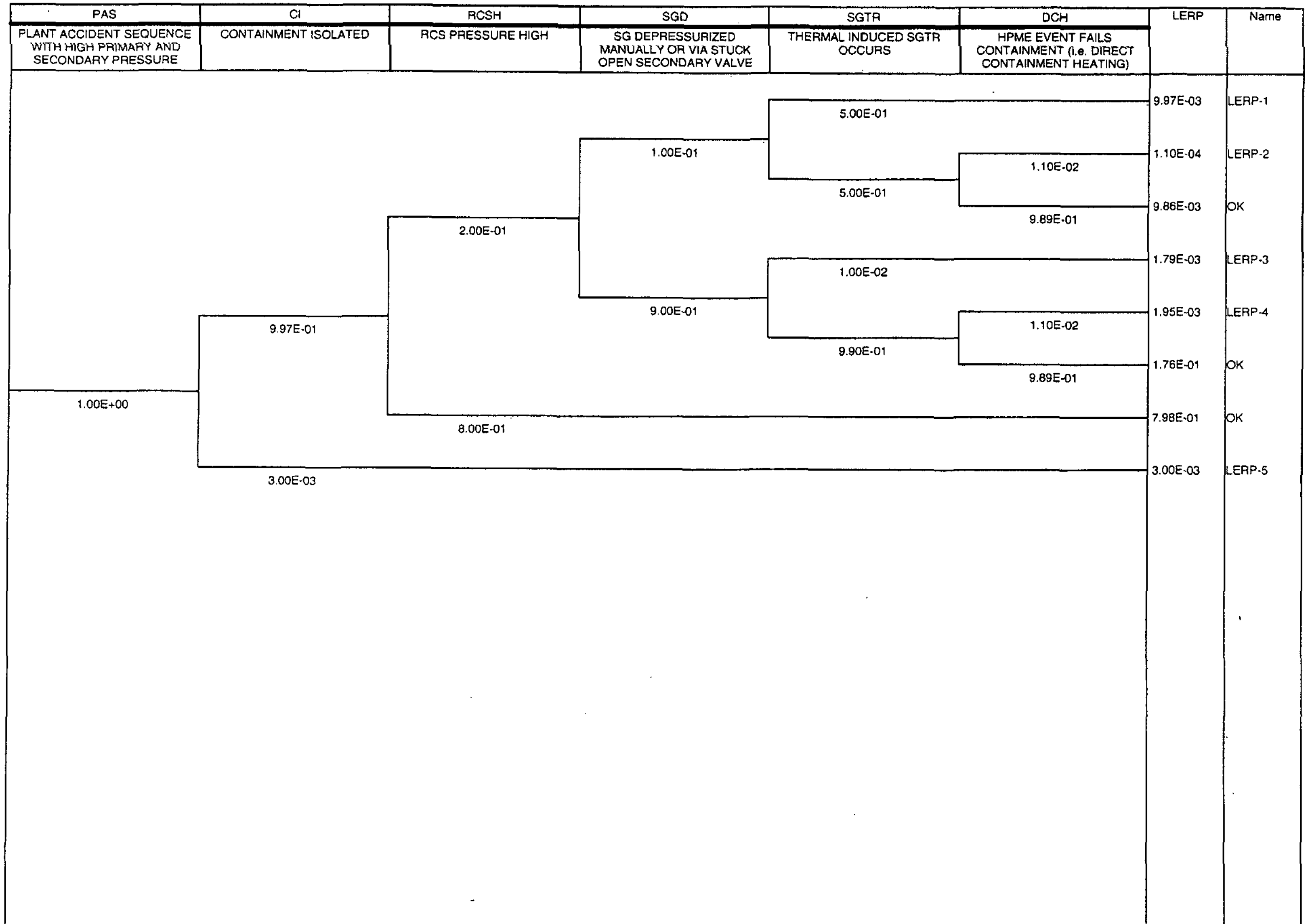
B-1: Simplified CLERP Event Tree for SIT	D:\RUPERT\303_SIT.eta	8/4/03	Page 1



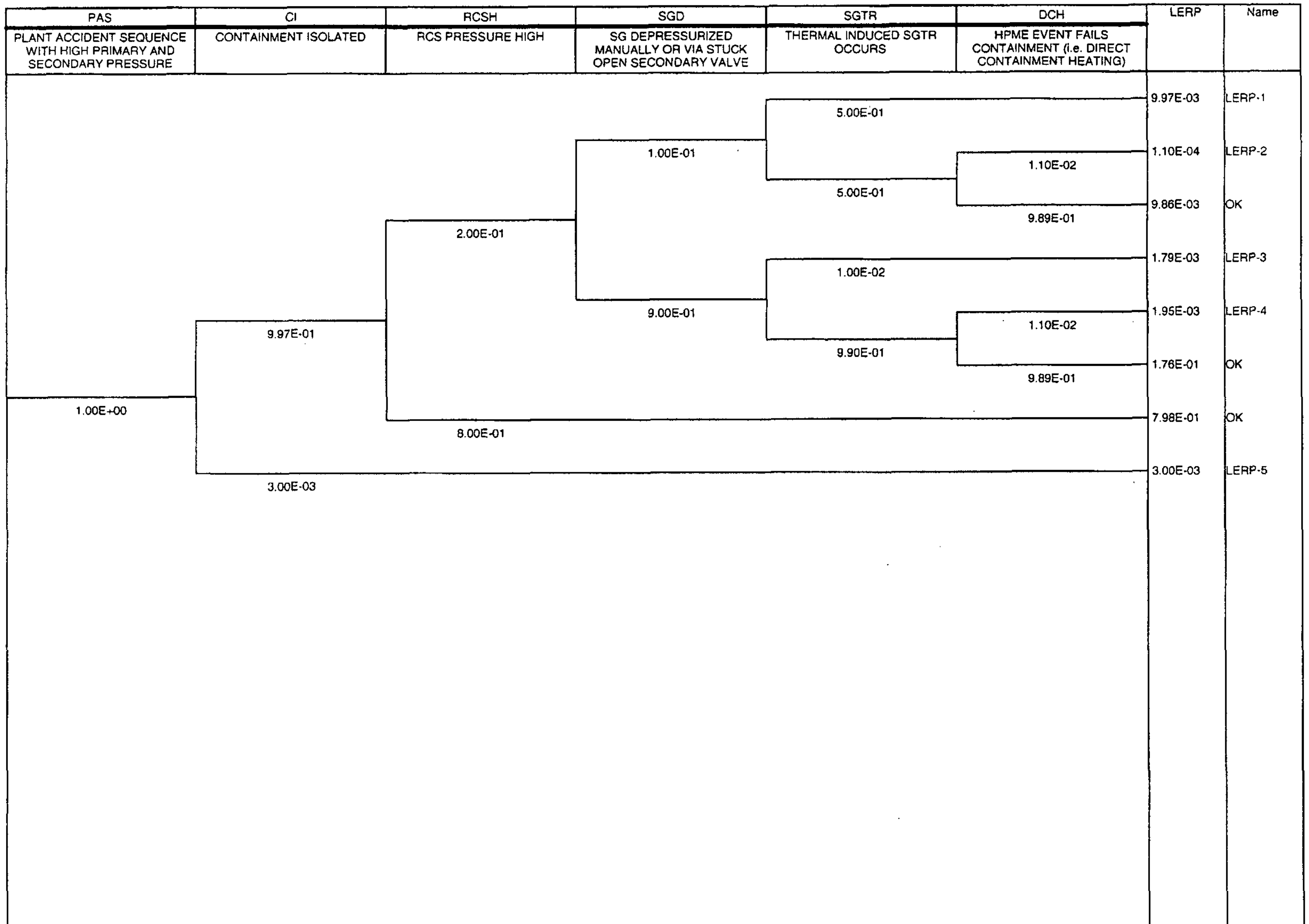
B-2: Simplified CLERP Event Tree for LPSI	D:\RUPERT\303_LPSI.eta	8/4/03	Page 1



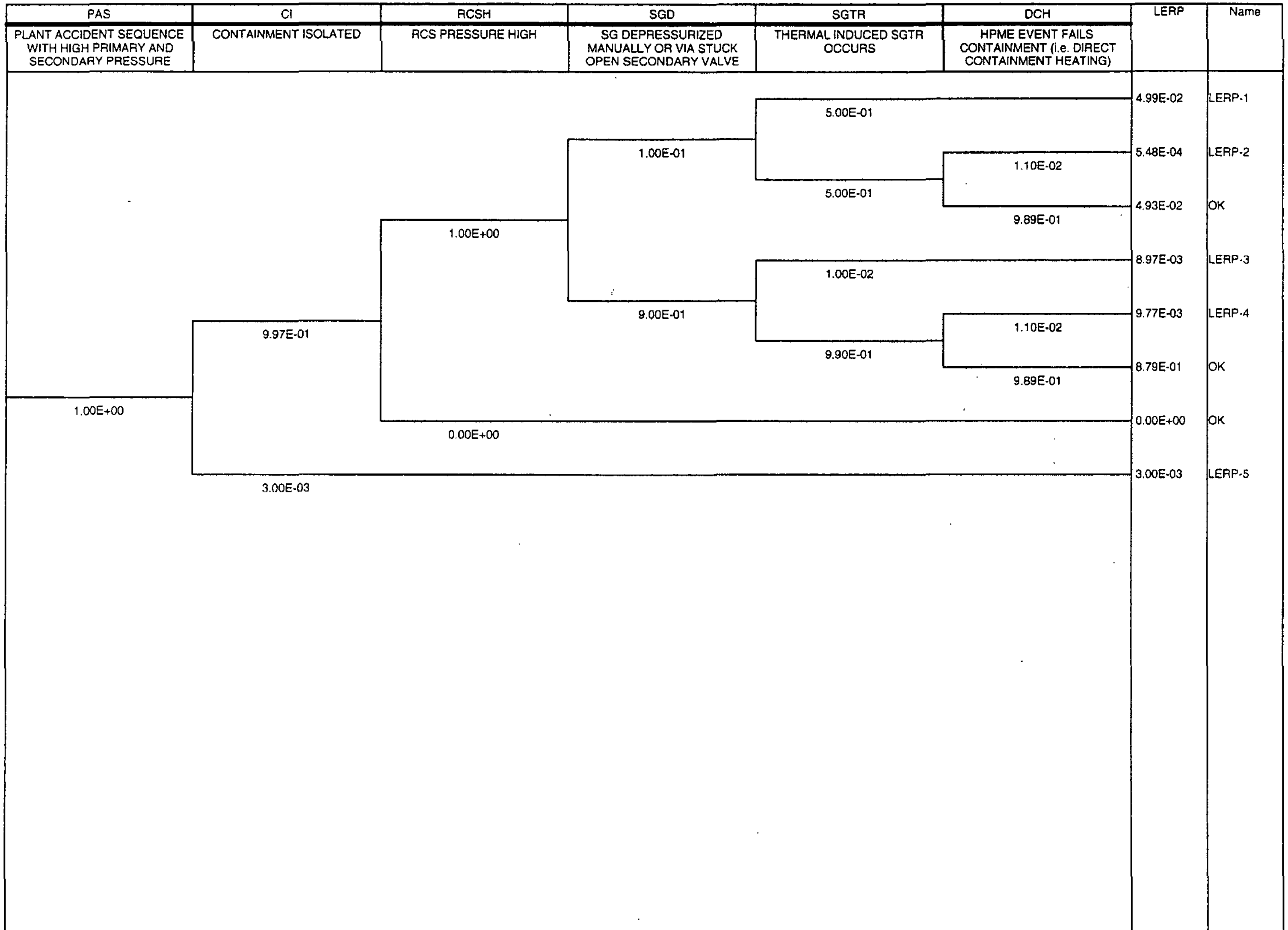
B-3: Simplified CLERP Event Tree for HPSI with PORV	D:\RUPERT\303_HPSP.eta	8/4/03	Page 1



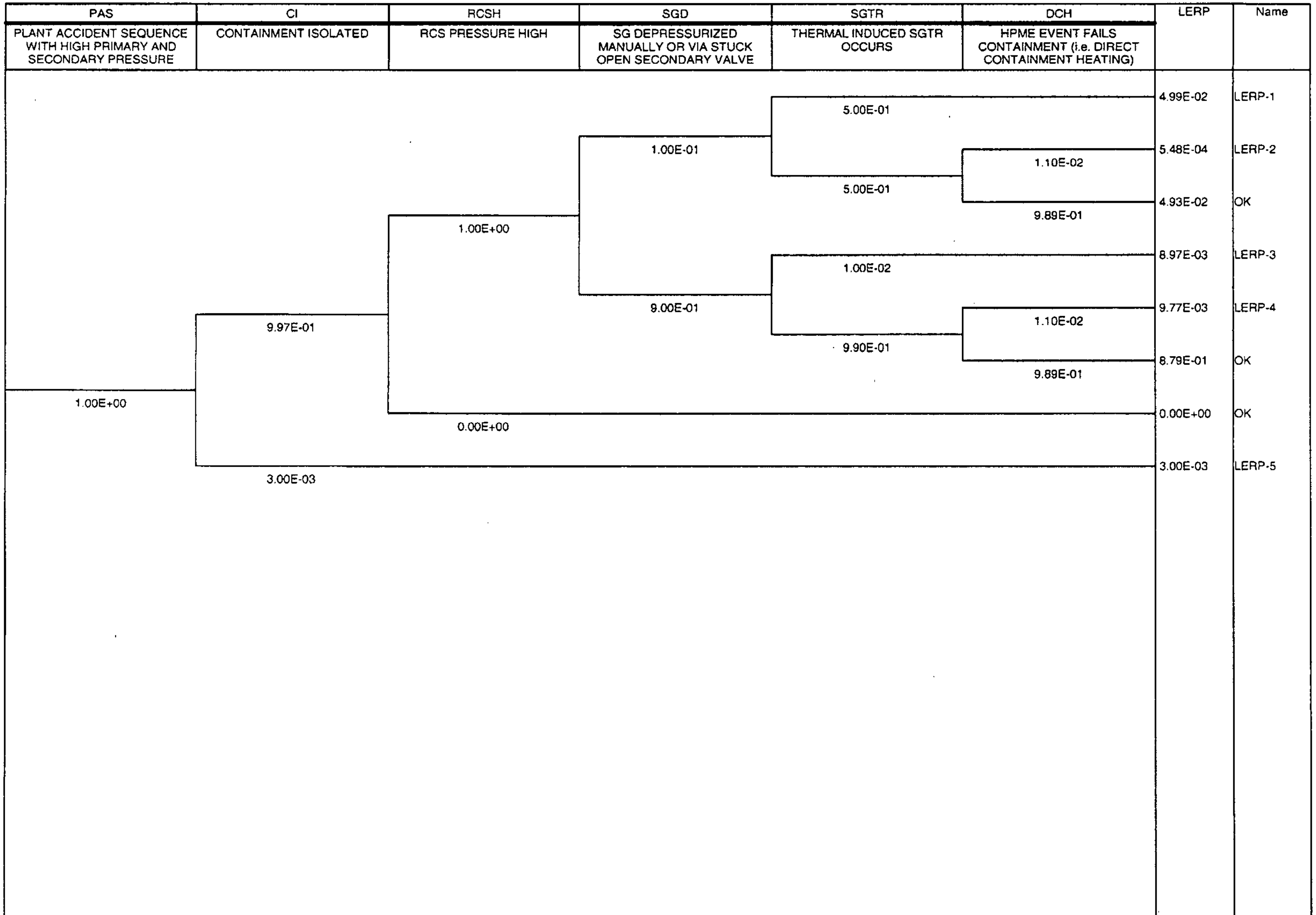
B-4: Simplified CLERP Event Tree for HPSI w/o PORV	D:\RUPERT\303_HPSI.eta	8/4/03	Page 1



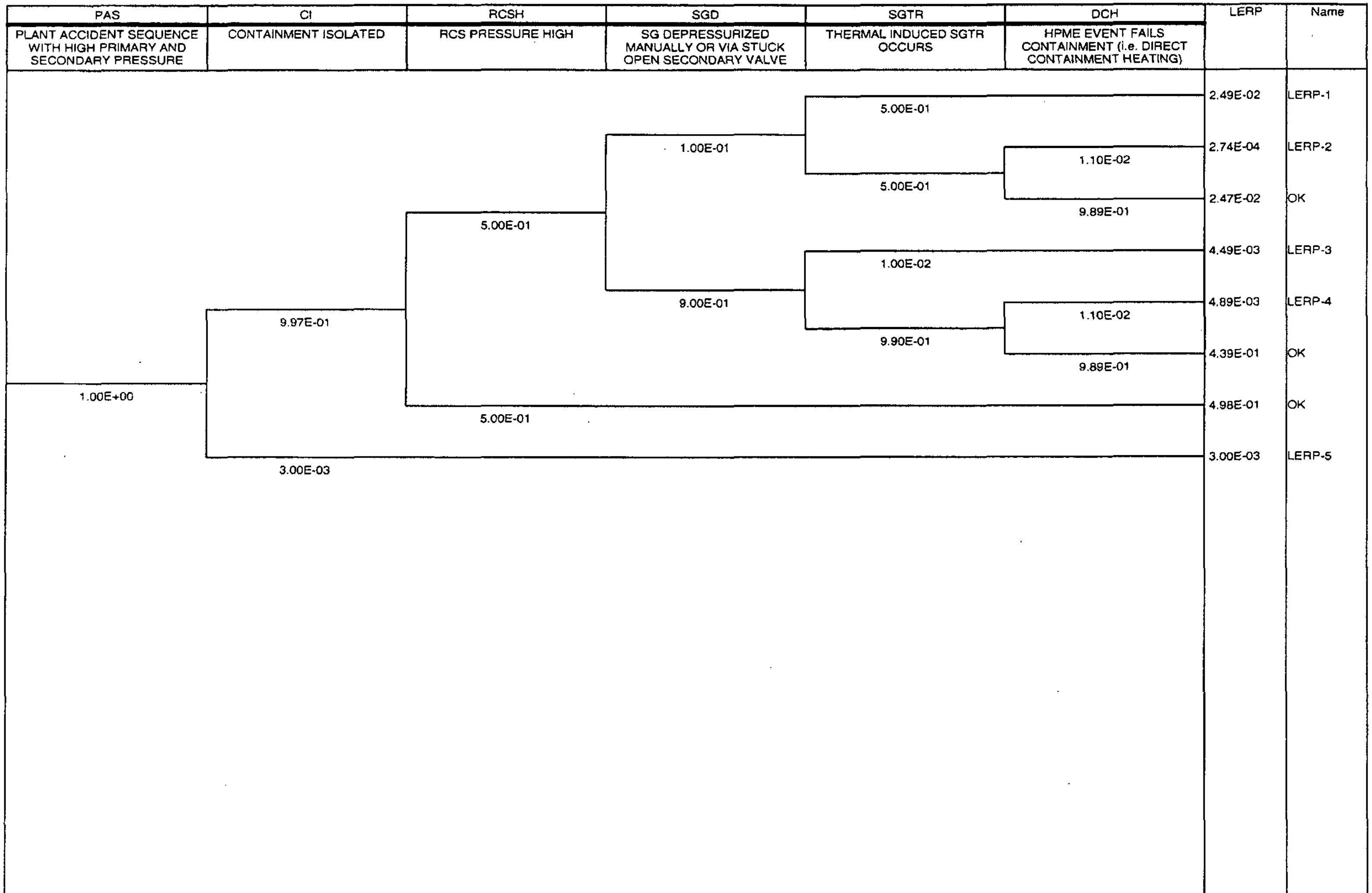
B-5: Simplified CLERP Event Tree for Containment Spray System	D:\RUPERT\303_CSS.eta	8/4/03	Page 1



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



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



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

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