Westinghouse Non-Proprietary Class 3
WCAP-16125-NP-A, Revision 2
(CE NPSD-1208-A, Revision 03)
August 2010

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

Westinghouse

WCAP-16125-NP-A, Revision 2
(CE NPSD-1208-A, Revision 03)

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

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August 2010

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

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

May 24, 2010

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PWROG
Project Office

Mr. Anthony Nowinowski, Manager
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SUBJECT: FINAL SAFETY EVALUATION FOR PRESSURIZED WATER REACTOR OWNERS GROUP TOPICAL REPORT WCAP-16125-NP, REVISION 2, "JUSTIFICATION FOR RISK-INFORMED MODIFICATIONS TO SELECTED TECHNICAL SPECIFICATIONS FOR CONDITIONS LEADING TO EXIGENT PLANT SHUTDOWN" (TAC NO. MD8138)

Dear Mr. Nowinowski:

By letter dated January 7, 2008, the Pressurized Water Reactor Owners Group (PWROG) submitted Topical Report (TR) WCAP-16125-NP, Revision 1, to the U.S. Nuclear Regulatory Commission (NRC) staff for review. In response to staff requests for additional information, TR WCAP-16125, Revision 2, was submitted on May 8, 2009. Additional clarifications and changes to the scope of the TR were submitted on July 8, 2009. By letter dated October 19, 2009, an NRC draft safety evaluation (SE) regarding our approval of WCAP-16125-NP, Revision 2, was provided for your review and comments. By letter dated November 11, 2009, the PWROG commented on the draft SE. The NRC staff's disposition of the PWROG's comments on the draft SE are discussed in the attachment to the final SE enclosed with this letter.

The NRC staff has found that TR WCAP-16125-NP, Revision 2, is acceptable for referencing in licensing applications for nuclear plants with Combustion Engineering (CE) designed nuclear steam supply systems to the extent specified and under the limitations delineated in the TR and in the enclosed final SE (changes to specifications 3.5.1 and 3.5.2 are not approved). The final SE defines the basis for our acceptance of the TR, with the identified exceptions.

Our acceptance applies only to material provided in the subject TR. We do not intend to repeat our review of the acceptable material described in the TR. When the TR appears as a reference in license applications, our review will ensure that the material presented applies to the specific plant involved. License amendment requests that deviate from this TR will be subject to a plant-specific review in accordance with applicable review standards.

In accordance with the guidance provided on the NRC website, we request that the PWROG publish the accepted non-proprietary version of this TR within three months of receipt of this letter. The accepted version shall incorporate this letter and the enclosed final SE after the title page. Also, it must contain historical review information, including NRC requests for additional information and your responses. The accepted version shall include an "-A" (designating accepted) following the TR identification symbol.

If future changes to the NRC's regulatory requirements affect the acceptability of this TR, the PWROG and/or licensees referencing it will be expected to revise the TR appropriately, or justify its continued applicability for subsequent referencing.

Sincerely,

Thomas B. Blount

Thomas B. Blount, Deputy Director
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

Project No. 694

Enclosure:
Final SE

cc w/encl: See next page

FINAL SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

TOPICAL REPORT WCAP-16125-NP, REVISION 2

"JUSTIFICATION FOR RISK-INFORMED MODIFICATIONS TO

SELECTED TECHNICAL SPECIFICATIONS FOR CONDITIONS

LEADING TO EXIGENT PLANT SHUTDOWN"

PRESSURIZED WATER REACTOR OWNERS GROUP

PROJECT NO. 694

1.0 INTRODUCTION AND BACKGROUND

On January 7, 2008, the Pressurized Water Reactor (PWR) Owners Group submitted topical report (TR) WCAP-16125, Revision 1, "Justification for Risk-Informed Modifications to Selected Technical Specifications for Conditions Leading to Exigent Plant Shutdown," (Reference 1) for the U.S. Nuclear Regulatory Commission (NRC) staff review and approval. In response to NRC staff requests for additional information (RAI), TR WCAP-16125, Revision 2 (Reference 2) was submitted on May 8, 2009. Additional clarifications and changes to the scope of the TR were submitted on July 8, 2009 (Reference 3). The TR provides justification for risk-informed technical specifications (TS) initiative 6 for nuclear plants with Combustion Engineering (CE) designed nuclear steam supply systems.

The proposed changes would address conditions involving the inoperability of both redundant trains of selected plant systems, which is a loss of function of the system, leading to an exigent plant shutdown, either due to applicability of limiting condition for operation (LCO) 3.0.3 or a specific short duration shutdown action requirement in the individual TS LCO. The intent of these modifications to the TSs is to provide a risk-informed alternative to the existing shutdown requirements to permit resolution of the loss of function condition while the plant continues to operate. The completion times (CT) associated with the proposed actions are specified. The extended CTs are intended to allow for the potential restoration of the system to an operable condition, thereby avoiding the risk associated with an immediate controlled shutdown.

Table 1 summarizes the proposed changes to the standard TS (NUREG-1432).

Enclosure

Table 1				
TS LCO	SYSTEM/COMPONENT	CONDITION	CURRENT CT	PROPOSED CT
3.1.9	Boration System	System inoperable	None (3.0.3)	24 hours
3.4.9	Pressurizer Heaters	Two groups of class 1E heaters inoperable	None (3.0.3)	24 hours
3.4.11	Power-operated Relief Valves (PORV)	Inability of two PORVs to open, or inability of both PORVs to close and block valves to be closed		8 hours
3.5.1	Safety Injection Tanks (SIT)	Two or more SITs inoperable	None (3.0.3)	24 hours[c]
3.5.2	Low Pressure Safety Injection (LPSI)	Two LPSI subsystems inoperable	None (3.0.3)	24 hours[c]
3.6.6.A	Containment Spray (CS) System	1) Two CSS trains inoperable, or 2) Two CSS and two Containment Cooling System (CCS) trains inoperable [a]	None (3.0.3)	1) 72 hours [b] 2) 12 hours
3.6.6.B [a]	CS System	Two CSS trains inoperable, or two CSS and two CCS trains inoperable	None (3.0.3)	12 hours
3.6.8	Shield Building Emergency Air Cleanup System (SBEACS)	Two trains inoperable	None (3.0.3)	24 hours
3.6.10	Iodine Control System (ICS)	Two trains inoperable	None (3.0.3)	24 hours
3.7.11	Control Room Emergency Air Cleanup System (CREACS)	Two trains inoperable for reasons other than an inoperable boundary	Explicit 3.0.3	24 hours
3.7.12	Control Room Emergency Air Temperature Control System (CREATCS)	Two trains inoperable (modes 1 – 4)	Explicit 3.0.3	24 hours

Table 1				
TS LCO	SYSTEM/COMPONENT	CONDITION	CURRENT CT	PROPOSED CT
3.7.13	Emergency Core Cooling System Pump Room Emergency Air Cleanup System (ECCS PREACS)	Two trains inoperable for reasons other than an inoperable boundary	None (3.0.3)	24 hours
3.7.15	Penetration Room Emergency Air Cleanup System (PREACS)	Two trains inoperable for reasons other than an inoperable boundary	None (3.0.3)	24 hours

[a] This proposed change was withdrawn by Reference 3.
[b] This proposed CT was revised from 72 hours to 24 hours by Reference 3.
[c] This proposed CT is not approved.

2.0 REGULATORY EVALUATION

2.1 Applicable Regulations

In Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.36, "Technical Specifications," the Commission established its regulatory requirements related to the content of TSs. Pursuant to 10 CFR 50.36, TSs are required to include items in the following categories related to station operation: (1) safety limits, limiting safety system settings, and limiting control settings; (2) LCOs; (3) surveillance requirements; (4) design features; and (5) administrative controls. The rule does not specify the particular requirements to be included in a plant's TSs. As stated in 10 CFR 50.36(c)(2), "Limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When a limiting condition for operation of a nuclear reactor is not met, the licensee will shut down the reactor or follow any remedial action permitted by the technical specifications until the condition can be met."

Most TS LCOs provide a fixed time interval, referred to as the allowed outage time (AOT) or CT, during which the LCO may not be met, to permit a licensee to perform required testing or maintenance activities, or to conduct repairs. Upon expiration of the CT, the requirement to shut down the reactor or follow remedial action is imposed. The proposed changes in the TR provide a means for the licensee to make unanticipated emergent repairs to a system, where both trains are inoperable, rather than immediately shutting down the reactor as required by current TS requirements.

2.2 Applicable Regulatory Criteria/Guidelines

The yearly risk impact is represented by the delta-core damage frequency (ΔCDF) and delta-large early release frequency (ΔLERF) metrics referenced in Regulatory Guide (RG) 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," (Reference 4). The risk of each individual application of a proposed risk-informed completion time is represented by the incremental conditional core damage probability (ICCDP) and the incremental conditional large early release probability (ICLERP), referenced in RG 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications," (Reference 5). The guidelines in RG 1.177 for ICCDP and ICLERP are intended to assure that each entry in a TS action results in no more than a very small incremental risk. However, since these proposed changes to CTs are not intended to be used as an operational convenience that permits voluntary removal of redundant systems or components from service in lieu of other alternatives that would not result in redundant systems or components being inoperable, the frequency of entry into the TS actions is expected to be very rare, therefore, the ICCDP and ICLERP metrics are not evaluated to determine the acceptability of the proposed changes.

General guidance for evaluating the technical basis for proposed risk-informed changes is provided in Section 19.2, "Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance," of the NRC Standard Review Plan (SRP), NUREG-0800 (Reference 6). More specific guidance related to risk-informed TS changes, including changes to TS CTs, is provided in SRP Section 16.1, "Risk-Informed Decisionmaking: Technical Specifications," (Reference 7).

Chapter 19.2 of the SRP states that a risk-informed application should be evaluated to ensure that the proposed changes meet the following key principles:

- The proposed change meets the current regulations, unless it explicitly relates to a requested exemption.
- The proposed change is consistent with the defense-in-depth philosophy.
- The proposed change maintains sufficient safety margins.
- When proposed changes increase core damage frequency or risk, the increase(s) should be small and consistent with the intent of the Commission's Safety Goal Policy Statement.
- The impact of the proposed change should be monitored using performance measurement strategies.

3.0 TECHNICAL EVALUATION

The staff evaluated the TR for conformance with the five key principles of SRP 19.2.

3.1 Compliance with Current Regulations

Regulations in 10 CFR 50.36 permit either a plant shutdown or other remedial actions specified by TSs when an LCO is not met. The proposed changes provide new action requirements for conditions of equipment inoperability which currently require an immediate plant shutdown. Since such remedial actions are permitted per 10 CFR 50.36, the proposed changes in the TR continue to comply with current regulations, and therefore, satisfy this key principle.

3.2 Defense-in-Depth

The proposed changes address conditions where both trains of a system are inoperable, resulting in a loss of that system's function and a temporary reduction in the defense-in-depth capabilities of the plant. Each proposed change addresses the remaining available alternative system(s) capable of providing mitigation of events, and, where applicable, includes requirements to assure these required backup systems are operable. The reduced level of defense-in-depth is retained by verification that both trains (if applicable) of the backup system are operable. Therefore, this key principle is satisfied by the unique requirements identified for each proposed TS change.

3.3 Safety Margins

The proposed changes do not have any impact on the use of NRC-approved codes and standards, nor do the changes impact any acceptance criteria used in a plant's licensing basis. Under the current TSs, if an accident occurs during the 6-hour controlled shutdown time of LCO 3.0.3 caused by two trains of these systems being unavailable, it could potentially result in offsite dose limits that do not meet NRC regulatory limits. Since the changes proposed by TR

WCAP-16125 do not modify the design basis of the systems evaluated, extending the AOT to 24 hours would have no quantitative effect on the dose consequence as compared to the existing condition. As such, the proposed changes would not significantly reduce a nuclear power plant's available safety margin, and this key principle is satisfied.

3.4 Performance Monitoring

The proposed changes would permit continued plant operation for short periods to address emergent equipment failures. Degradation of equipment performance could lead to excessive use of the new action requirements. This is adequately addressed by equipment performance monitoring required by 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," and therefore, this key principle is satisfied.

3.5 Risk Assessment

The risk of each of the TS LCOs for which action requirements are proposed is evaluated in the TR by three methods, as described below.

1. For those TSs governing systems or components which provide mitigation of core damage and large early releases, ΔCDF and ΔLERF metrics are calculated using a simplified generic method, and the results are compared to the acceptance guidelines of RG 1.174. This applies to TS 3.1.9 Boration Systems, 3.4.11 PORVs, 3.5.1 SITs, and 3.5.2 ECCS (LPSI).

   For calculations of ΔCDF, a bounding approach was applied to evaluate loss of function of a system by identifying the initiating events for which the system provides mitigation, and assuming that the event goes directly to core damage. No credit was taken for alternate mitigation strategies, and the baseline CDF was effectively assumed to be zero. The initiating event frequencies were taken from NUREG/CR-5750 (Reference 8).

   For ΔLERF, a simplified approach using an event tree was developed to calculate the fraction of core damage events which result in large early releases. The event tree assessed containment isolation status, reactor coolant system (RCS) pressure, secondary side depressurization via the steam generators, thermally-induced steam generator tube rupture (SGTR), and reactor pressure vessel (RPV) lower head failure. Assumptions related to the potential impact on LERF for each of these events, and the associated basis for probabilities used in the analysis, are discussed below:

   <u>Containment Isolated</u> – This event defines containment integrity prior to the core damage event. If containment is not isolated, then a large early release will occur concurrent with core damage. A probability of 3.0E-3 was applied for an unisolated containment, which is identified as the upper end of the range used in the CE Probabilistic Risk Assessment (PRA) models.

   <u>RCS Pressure – High</u> – This event defines the RCS pressure at the time of core damage. If the pressure is low, then large early releases are assumed not to occur (except via an unisolated containment); otherwise, thermally-induced SGTR and high pressure melt ejection events are further evaluated. All core damage events involving loss-of-coolant

accidents (LOCAs) are assumed to result in low or intermediate RCS pressure, and all other events result in high RCS pressure.

Steam Generator Depressurization – This event defines the status of the secondary side, and affects the next event which is the potential for induced SGTR. Depressurization of the secondary side occurs either due to prior operator response or due to failure of a safety relief valve. Based on NUREG-1570 (Reference 9), a probability of 0.9 is assigned for secondary depressurization.

Thermally-induced SGTR Occurs – This event represents a loss of steam generator tube integrity due to thermal stresses during a severe accident, which is assumed to result in a large early release. Two values are used, based on the status of the prior event, for steam generator depressurization. A probability of 0.5 is assigned when the steam generators are depressurized, and 0.01 otherwise. These values are conservative, based on the assumptions regarding tube age and integrity and based on neglecting operator actions to depressurize the RCS after core damage.

RPV Lower Head Failure Results in Containment Failure – This event represents a high pressure failure of the lower head, with an energetic discharge of the molten fuel and direct containment heating, leading to failure of containment. Based on NUREG/CR-6338 (Reference 10), the conditional containment failure probability given the event for CE-designed plants is 0.01, which is considered to be a bounding value.

None of the assessed initiating events include either SGTRs or other containment bypass events because the systems being evaluated do not mitigate these events. The NRC staff concludes that the simplified LERF event tree is reasonable and acceptable to support the evaluation of LERF for the scope of the TR.

2. For TS 3.4.9 Pressurizer Heaters, an evaluation of the increased likelihood of a plant trip due to degraded pressure control is made in order to calculate ΔCDF. The ΔLERF calculation for this TS is the same simplified approach described above for case 1.

3. For all remaining systems associated with mitigation of radiological releases with magnitudes less than those associated with LERF, there is no impact to either CDF or LERF, as the systems are provided to meet design basis dose limits. An evaluation of the frequency of events which challenge the systems is made and compared to the acceptance guidelines of RG 1.174 applicable to ΔLERF in order to characterize the risk of these lesser releases. The TR provides additional justification based on the availability of other systems which provide a degree of defense-in-depth for prevention of these releases. The following systems are addressed in this manner: TS 3.6.10 ICS, TS 3.6.8 SBEACS, TS 3.7.15 PREACS, TS 3.7.13 ECCS PREACS, TS 3.7.11 CREACS, and TS 3.7.12 CREATCS.

   The primary purpose of these systems is to ensure that accident radiation exposures meet the limits as defined in General Design Criteria 19, "Control Room," 10 CFR 100.11, "Determination of Exclusion Area, Low Population Zone, and Population Center Distance," or 10 CFR 50.67, "Accident Source Term," for plants that have adopted the alternative source term. The design basis dose consequence analyses that credit these systems are based on the assumption of an intact containment. In addition, since these systems are designed only

to be used in the event of an accident condition, the operability of these systems has no impact on normal releases of radioactive material as governed by 10 CFR Part 20, "Standards for Protection Against Radiation."

To assess the impact of the unavailability of these systems, the TR examined the expected iodine releases for three categories of events:

- Beyond design basis scenarios that lead to large early releases,
- Maximum Hypothetical Accident (MHA), and
- LOCA and Non-LOCA Design Basis Accidents (DBA).

The purpose of this assessment is to show that, using worst case assumptions, the potential accident releases anticipated under the short term operational conditions proposed by the increased CT for the ICS, SBEACS, PREACS and ECCS PREACS will be well below and bounded by a large early release. For clarity, the TR evaluation was limited to the release of iodine. For each category, iodine releases were estimated assuming various combinations of system availability. The results of this assessment are shown in Table 4.3-1 of the TR, supplemented by additional information in Reference 2. The NRC staff reviewed the assumptions and methodology used to determine the bounding iodine release quantities and resulting dose consequences and found that in all cases appropriately conservative assumptions were used. In many cases the assumptions used were more conservative than the applicable regulatory guidance. For instance, the Table 4.3-1 MHA analyses assume a containment leak rate of 0.5% per day which is significantly higher than the TS allowed leak rate normally used in dose consequence analyses. The evaluation of the iodine release contribution from ECCS leakage was based on a conservative iodine partition factor of 0.01. This is considered to be a conservative assumption as suggested in the applicable regulatory guidance; however, in various licensing actions, the NRC staff has accepted more realistic values when supported by a sound technical basis.

In most of these cases evaluated, the offsite doses for the MHA and DBA, with various combinations of systems inoperable, are near or below regulatory limits. If more realistic yet still conservative assumptions were incorporated into the dose consequence analysis, it could easily be shown that offsite doses for the MHA and DBA accidents would be below regulatory limits for all the cases evaluated. Furthermore, when realistic meteorological conditions are assumed and containment sprays are available, calculated doses for all the evaluated accident conditions including the MHA are shown to be within regulatory limits.

To reduce the impact of an increased CT for the CREACS, the TR proposes to add conditions to verify that RCS specific activity is within limits and to verify that dose mitigating actions are available in the control room. For limited durations, such as the short term operational conditions proposed by the increased CT for the CREACS, the NRC has accepted credit for the use of respirators and potassium iodide on an interim basis to demonstrate that control room dose limits can be met.

Similarly, the TR proposes to add pre-planned actions to ensure that the impact of loss of post accident temperature control associated with an increased CT for the CREATCS is mitigated. Actions can include use of portable fans, temporary opening of doors or use of normal heating, ventilation, and air conditioning systems. To support this change,

administrative controls will be provided to monitor the control room temperature to ensure control room habitability and operability of TS equipment. If compensatory measures impact the CR envelope, the operability of containment and auxiliary building post accident air cleanup systems will be verified. The 24 hour CT proposed in the TR for the CREACS and the CREATCS is consistent with the allowed 24-hour period for the evaluation of a breach of the control room envelope provided in Technical Specification Task Force (TSTF)-448.

Based on an evaluation of the methods and assumptions used to produce the results shown in Table 4.3-1, the NRC staff has reasonable assurance that the postulated accident releases calculated for the short-term operational conditions proposed by the increased CT for the ICS, SBEACS, PREACS and ECCS PREACS will be well below the LERF releases. In addition, the NRC staff has reviewed the bases for the increased CT for the CREACS and the CREATCS and has determined that the proposed conditions and compensatory measures provide reasonable assurance that control room habitability will be adequately maintained during the proposed 24-hour CT.

External events, including internal fires and floods, were not evaluated in the TR. None of the systems being evaluated provide a primary mitigating function for external events, and therefore these events are not significant to the risk-informed decision.

The TR also evaluated sensitivity studies for key areas of uncertainty in the analyses. Specifically, the TR considered uncertainties in the initiating event frequencies which are the input to the CDF calculations and showed that even assuming a 95% upper bound frequency would not result in excessive risk. These were also propagated into the LERF calculations with similar results. The TR also addressed uncertainties in the thermally-induced SGTR assumptions and steam generator depressurization assumptions, and demonstrated that the LERF results are not significantly impacted. These sensitivity studies performed to evaluate the key sources of uncertainty in the risk analyses adequately demonstrate the robustness of the results to support the proposed TS changes.

Each individual proposed change to TS is discussed in detail in Sections 3.5.1 – 3.5.10.

3.5.1 LCO 3.1.9 – Boration Systems – Operating

The boration systems are required to ensure that adequate shutdown margin exists to bring the plant to cold shutdown with the most reactive control element assembly not fully inserted into the reactor core and the decay of all xenon poison. The systems also mitigate main steam line breaks and reactor coolant pump restart by preventing a return to power scenario (due to cold water injection), and also mitigate anticipated transients without scram (ATWS) events. (Only the ATWS mitigation function is typically included in PRA models because the other functions are not directly related to core damage events.) These systems are not included in the current standard TSs (NUREG-1432) because they do not satisfy any of the criteria of 10 CFR 50.36.

The plant-specific TSs in applicable CE plants do not provide any action requirements for two inoperable boration paths, and therefore TS 3.0.3 applies, which requires an immediate plant shutdown. The proposed change provides for a 24-hour CT to restore at least one boration path to operable status, to permit continued operation under an existing action requirement.

A risk assessment of the proposed 24-hour CT was provided which conservatively assumed that all ATWS events would result in core damage, with the following results:

ΔCDF	RG 1.174 Guidance	ΔLERF	RG 1.174 Guidance
1.55E-8/yr	<1.0E-6/yr	1.12E-9/yr	<1.0E-7/yr

The ΔCDF and ΔLERF were assessed based on an assumed once per three year entry into the proposed action requirement and assumed that the entire 24-hour duration of the CT is used. The risk metrics are well below the acceptance guidelines of RG 1.174 as noted in the table.

A reduced level of defense-in-depth is retained by verification of the operability of the high pressure safety injection (HPSI) system and the PORVs. The HPSI can provide boration of the reactor after RCS pressure is reduced to less than [1350 psig]; operability is required by TS 3.5.2 and 3.5.3 and 3.4.11, which assure operability of at least one HPSI train and one pressurizer PORV and its associated block valve.

Based on the conservatively calculated risk impact being within the acceptance guidelines of RG 1.174, and on defense-in-depth provided by an operable HPSI train, the NRC staff finds the proposed new action requirement and 24-hour CT are acceptable.

3.5.2 LCO 3.4.9 – Pressurizer Heaters

The pressurizer and the class 1E electrical heaters maintain a liquid-to-vapor interface to permit RCS pressure control during normal operations and in response to anticipated design basis transients. The class 1E heaters, with their power provided by emergency AC power busses, are used to maintain RCS subcooling during a natural circulation cooldown, and the unavailability of the heaters will extend the time to reach entry conditions for the shutdown cooling system. The unavailability of the class 1E heaters may complicate steady-state RCS pressure control and may increase the potential of an unplanned reactor trip. However, the availability of additional heaters beyond the two groups required by this TS LCO permit continued RCS pressure control.

The standard TSs do not provide any action requirements for two inoperable pressurizer heater groups, and therefore TS 3.0.3 applies which requires an immediate plant shutdown. The proposed change provides for a 24-hour CT to restore at least one pressurizer heater to operable status, to permit continued operation under an existing action requirement.

The unavailability of the class 1E pressurizer heaters would not have any significant impact on plant transient response, and so there is no quantifiable impact to CDF or LERF. While mitigation of a SGTR is enhanced by the availability of pressurizer heaters, the non-class 1E heaters can also function if offsite power is available, and plant procedures provide for mitigation of a SGTR without pressurizer heaters, if necessary.

Conservatively, the risk impact due to increased likelihood of a reactor trip was calculated by assuming an order-of-magnitude increase in the reactor trip frequency when both class 1E heaters are inoperable. The risk impact is then calculated based on the conditional core damage probability given a reactor trip with no other complications:

ΔCDF	RG 1.174 Guidance	ΔLERF	RG 1.174 Guidance
1.0E-7/yr	<1.0E-6/yr	3.8E-9/yr	<1.0E-7/yr

The ΔCDF and ΔLERF were assessed based on an assumed once per three year entry into the proposed action requirement and assumed that the entire 24-hour duration of the CT is used. The risk metrics are well below the acceptance guidelines of RG 1.174 as noted in the table.

Minimum pressurizer heater capability is supplemented by the normal availability of non-class 1E heaters for normal plant pressure control, and the availability of plant procedures which provide plant shutdown and cooldown guidance with or without pressurizer heaters. If the available heaters are sufficient to maintain RCS pressure control, normal plant operations can continue. Because unavailability of class 1E and non-class 1E heaters would physically result in plant shutdown, the NRC staff does not consider it necessary to specify additional TS or administrative requirements for the non-class 1E heater availability.

Based on the conservatively calculated risk impact being within the acceptance guidelines of RG 1.174, and the limited impact of plant shutdown and cooldown without pressurizer heaters, the NRC staff finds the proposed new action requirement and 24-hour CT are acceptable.

3.5.3 LCO 3.4.11 – Pressurizer PORVs and Associated Block Valves

The pressurizer PORVs and block valves are required to be operable to minimize the potential for a small break LOCA through a PORV pathway. The PORVs automatically open for RCS pressure control to avoid challenging the primary safety relief valves, and may be manually opened by the operator to control pressure. In the event of a total loss of feedwater to the steam generators, one or more PORVs may be opened manually to provide for feed-and-bleed cooling of the reactor using once-through cooling from high pressure injection to the RCS. The PORVs may also be used for low temperature overpressure protection during heatup and cooldown. The PORV may be manually operated to depressurize the RCS in response to normal or abnormal transients. The PORV may be used for depressurization when pressurizer spray is not available, a condition that may be encountered during loss of offsite power. The PORVs can be manually operated to reduce RCS pressure in the event of a steam generator tube rupture (SGTR) with offsite power unavailable.

The standard TS is proposed to be revised to provide an 8-hour CT to restore at least one PORV or one block valve to operable status, to permit continued operation under an existing action requirement. This action may only be applied provided the PORV is isolable by its block valve.

This constraint is already provided for in the TS action requirements to close the associated block valve, and if this cannot be accomplished, an immediate plant shutdown is still required.

The risk impact of unavailable PORVs or block valves is primarily attributable to the non-design basis function of providing for feed-and-bleed cooling:

ΔCDF	RG 1.174 Guidance	ΔLERF	RG 1.174 Guidance
1.5E-7/yr	<1.0E-6/yr	1.1E-8/yr	<1.0E-7/yr

The ΔCDF and ΔLERF were assessed based on an assumed once per three years entry into the proposed action requirement and assumed that the entire 8-hour duration of the CT is used. The risk metrics are well below the acceptance guidelines of RG 1.174 as noted in the table.

The primary safety relief valves provide the design basis pressure control function, controlled by TS requirements. The non-design basis feed-and-bleed function is considered to be risk-significant, and the proposed change includes a TS requirement to confirm that the LCO for auxiliary feedwater (AFW) is met, which requires both trains to be operable. A reduced level of defense-in-depth is retained by verification of the operability of the both AFW pumps. These requirements assure that mitigation capability is available for those DBAs or anticipated operational occurrences (AOOs) requiring the pressure control and heat removal functions of the PORVs.

In addition, the new 8-hour CT does not apply in the standard TSs to PORVs which are leaking and unisolable. Plants implementing the proposed change must verify that their TS changes are consistent with the standard TS with regards to leaking and unisolable PORVs.

Based on the risk impact being within the acceptance guidelines of RG 1.174, and the additional restriction on operability of both AFW trains in the TS action, the NRC staff finds the proposed new action requirement and 8-hour CT are acceptable.

3.5.4 LCO 3.5.1 – Safety Injection Tanks (SIT)

The SITs provide for rapid refill of the RCS following a large break LOCA. The SITs are passive devices; no component actuation or operator control is required for the components to perform their safety function. The standard TSs explicitly require entry into LCO 3.0.3 when two or more SITs are inoperable. The proposed change would provide a 24-hour CT to restore at least all but one SIT to operable status, to permit continued operation under an existing action requirement.

The unavailability of the SITs will compromise the ability of the plant to respond to Large Break LOCA events. Additionally, 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. The availability of both trains of the ECCS (HHSI and LHSI) limits the impact of unavailable SITs and provides a reduced level of mitigation for the certain LOCA break sizes. However, because these defense-in-depth systems do not provide accident mitigation criteria required by 10 CFR 50.46, the staff does not find the loss of the design basis SIT function to be acceptable.

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

The LPSI system refloods the RCS following a large break LOCA during the injection phase of ECCS operation. The system is normally shut down after the refueling water storage tank is depleted, at which point the recirculation phase of ECCS operation is entered, using the HPSI system only.

The standard TSs explicitly require entry into LCO 3.0.3 when both LPSI trains are inoperable. The proposed change would provide a 24-hour CT to restore at least one LPSI train to operable status, to permit continued operation under an existing action requirement.

The availability of the SITs and both trains of the HPSI limits the impact of having no operable LPSI system by providing mitigation capability for some LOCA scenarios normally mitigated by the LPSI system. The TR stated that best-estimate analysis using the CEFLASH-4AS code on a representative CE PWR showed that LOCAs with break sizes up to the size of the largest branch line pipe break can be mitigated using one HPSI pump and the SITs. However, these defense-in-depth systems do not provide accident mitigation criteria required by 10 CFR 50.46. Furthermore, the NRC staff is reviewing the applicability of LOCA leak-before-break in regards to 10 CFR 50.46 criteria and it is premature to consider this for the LPSI loss-of-function. Therefore, the NRC staff does not find the loss of the design basis LPSI function acceptable.

3.5.6 LCO 3.6.8 – Shield Building Exhaust Air Cleanup System
LCO 3.6.10 – Iodine Cleanup System
LCO 3.7.15 – Penetration Room Exhaust Air Cleanup System

The SBEACS (also called the Shield Building Ventilation System), ICS, and PREACS function to assure radioactive material released from containment leakage following a design basis accident is filtered prior to being exhausted to the environment. Each system includes two redundant trains with high efficiency particulate air filters, moisture absorbers, and charcoal adsorbers in the flowpath. The SBEACS filters leakage from containment into the shield building for dual containment facilities. The PREACS filters leakage from containment into the penetration room between the containment and the auxiliary building. The ICS removes elemental iodine directly from the containment atmosphere. The design basis for these systems is a postulated MHA involving a LOCA with a short duration uncovery of fuel, resulting from a temporary interruption, or significant degradation, of the ECCS flow. The event is assumed to result in significant iodine releases (40 - 50% of core inventory) from the fuel into the containment. The containment remains intact, with no more than the design basis leakage permitted by TSs. Releases associated with the MHA are significantly below the release which would occur for a postulated large early release (at least two orders of magnitude lower). None of these systems provide any mitigation capability for preventing either core damage or large early releases.

The current standard TSs do not address the condition of two inoperable trains of these systems; therefore, a default LCO 3.0.3 entry is required, resulting in an immediate plant shutdown. The proposed change would provide a 24-hour CT to restore at least one train of the affected system to operable status, to permit continued operation under an existing action requirement.

As noted above, these systems do not provide any core damage or large early release mitigation. Therefore, the risk metrics are zero for these systems. However, it may be conservatively assumed that if any of these systems are unavailable following a postulated core damage event, then some radioactive release above design limits, but well below the large early release level, would occur. A bounding estimate for CDF of CE plants was identified as 1E-4/year, so that over a 24-hour period the probability of a significant core damage event which would require the unavailable system would be:

(1E-4/year) x (24 hours) x (year/8760 hours) = 2.7E-7

Assuming a once per three year entry into the new TS would result in a frequency of a "less than LERF" release of about 9.0E-8/year. This frequency is within the acceptance guidance of RG 1.174 applicable to large early releases, and therefore provides a context for consideration of the risk impact for smaller releases.

As noted in the TR, there are also higher frequency design basis accidents (e.g., rod ejection and reactor coolant pump locked rotor) which are assumed to result in fuel damage and therefore rely upon these systems to filter any containment leakage. These accidents are associated with releases from the fuel into containment two or more orders of magnitude below those associated with the MHA described above, and four or more orders of magnitude below large early releases.

Containment spray can effectively scrub the post-accident containment atmosphere of fission products and therefore reduce reliance upon the downstream air cleanup systems. In order to assure additional defense-in-depth protection for the spectrum of accidents for which these systems provide mitigation, the TS action will include a verification of operability of at least one train of the CS system.

Based on the zero risk impact for severe accidents being well below the acceptance guidelines of RG 1.174, and the additional restriction on operability of at least one CS train in the TS, the NRC staff finds the proposed new action requirements and 24-hour CTs are acceptable.

3.5.7 LCO 3.7.13 – ECCS PREACS

The ECCS PREACS filters air from the area of the active ECCS components during the recirculation phase of a LOCA to remove volatized iodine resulting from leakage of post-accident recirculation fluid from the ECCS systems. As for the SBEACS, ICS, and PREACS discussed in Section 3.5.6, the design basis for the system is a postulated MHA involving a LOCA with a short duration uncovery of fuel, resulting from a temporary interruption, or significant degradation, of the ECCS flow. During the recirculation phase of ECCS operation following a LOCA, radioactive water from containment flows through the active portions of the ECCS. The ECCS PREACS assures that any nominal leakage from the ECCS (e.g., pump seals, valve flanges, etc.) is filtered before reaching the environment. Releases associated with the MHA are significantly below the release which would occur for a postulated large early release (at least two orders of magnitude lower). Therefore, this system does not provide any mitigation capability for preventing either core damage or large early releases.

The current standard TSs do not address the condition of two inoperable trains of the ECCS PREACS; therefore, a default LCO 3.0.3 entry is required, resulting in an immediate plant shutdown. The proposed change would provide a 24-hour CT to restore at least one train of the affected system to operable status, to permit continued operation under an existing action requirement. (Note that the current standard TS do provide a 24-hour CT when both trains are inoperable due specifically to ECCS pump room boundary inoperability.)

As noted above, the ECCS PREACS does not provide any core damage or large early release mitigation. Therefore, the risk metrics are zero for this system. However, it may be conservatively assumed that if both ECCS PREACS trains are unavailable following a postulated core damage event, then some radioactive release above design limits, but well below the large early release level, would occur. Since ECCS recirculation is only used for large and medium LOCAs, a bounding estimate for a core damage event during recirculation is made by assuming all large and medium LOCAs go to core damage; the total frequency of these LOCAs was identified as 4E-5/year, so that over a 24-hour period the probability of a significant core damage event which would require the ECCS PREACS would be:

(4E-5/year) x (24 hours) x (year/8760 hours) = 1.2E-7

Assuming a once per three year entry into the new TS would result in a frequency of a "less than LERF" release of about 4.0E-8/year. This frequency is within the acceptance guidance of RG 1.174 applicable to large early releases, and therefore, provides a context for consideration of the risk impact for smaller releases.

As noted in the TR, there are also higher frequency design basis accidents (e.g., rod ejection and reactor coolant pump locked rotor) which are assumed to result in fuel damage and therefore rely upon these systems to filter any containment leakage. These accidents are associated with releases from the fuel into containment two or more orders of magnitude below those associated with the MHA described above, and four or more orders of magnitude below large early releases.

Unlike for the other radiological filtration systems, the operation of the CS system does not have any mitigating impact on the unavailability of the ECCS PREACS, because the sprays transport the fission products from the containment atmosphere into the sumps, which is the source of recirculation water which is then assumed to leak into the ECCS pump room. Therefore, there is no requirement for operability of the CS system when the ECCS PREACS is inoperable.

Some designs of the ECCS PREACS rely upon the system to provide temperature control of the ECCS pump rooms, which may, therefore, impact the operability of the ECCS pumps and equipment. The NRC staff has not evaluated the unavailability of the ECCS pump room cooling function, and therefore, this TS change is prohibited if the ECCS PREACS is required by the plant design basis for ECCS pump room cooling.

Based on the zero risk impact for severe accidents being well below the acceptance guidelines of RG 1.174, and the prohibition for applicability of this TS change for plant designs which credit the room cooling function of the ECCS PREACS, the NRC staff finds the proposed new action requirement and 24-hour CT is acceptable.

3.5.8 LCO 3.7.11 – Control Room Emergency Air Cleanup System

The CREACS provides for filtration of outside air delivered to the control room by the ventilation system in the event of radioactive releases of particulates or iodine from containment following an accident involving fuel failures. This is to assure that control room personnel are protected from potential radiation exposures in excess of regulatory limits. The system may also provide protection of control room personnel from chemical or toxic gas releases by isolating the control room air intakes.

The current standard TSs do not address the condition of two inoperable trains of these systems; therefore, a default LCO 3.0.3 entry is required, resulting in an immediate plant shutdown. The proposed change would provide a 24-hour CT to restore at least one train of the CREACS to operable status, to permit continued operation under an existing action requirement. (Note that the current standard TS do provide a 24-hour CT when both trains are inoperable due specifically to control room pressure boundary inoperability.)

In the event of an accident involving radioactive releases without the availability of the CREACS, there would be no direct impact on the capability of the control room staff to perform any actions required to mitigate severe core damage or large early releases, because alternative protective measures would be implemented to reduce the dose impacts. If the accident did not involve severe core damage, control room doses even without the CREACS would be minimal, and therefore the CREACS has no direct role in preventing core damage (i.e., $\Delta CDF = 0$). If a core damage accident did occur with CREACS unavailable, then the bounding impact would be to simply assume the event proceeded to a large early release based on the unavailability of the control room personnel to perform any mitigating actions. This assumption would be very conservative, since large releases occur primarily due to containment bypass accidents, and control room actions following core damage do not prevent the release from occurring.

A bounding estimate for CDF of CE plants was identified as 1E-4/year, so that over a 24-hour period the probability of a significant core damage event, which with the CREACS unavailable is assumed to proceed to a large early release, would be:

(1E-4/year) x (24 hours) x (year/8760 hours) = 2.7E-7

Assuming a once per three year entry into the new TS, and assuming the entire 24-hour duration of the CT is used, the conservatively calculated $\Delta LERF$ is about 9.0E-8/year. This $\Delta LERF$, and the zero ΔCDF, are below the acceptance guidelines of RG 1.174.

A significant contributor to control room radiological hazards was identified in the TR from the release of radioactive RCS fluid from a SGTR event. A required TS action to verify LCO 3.4.16, "RCS Specific Activity" is met will be included in the new proposed action to provide additional defense-in-depth.

The TR also addressed a TS action to require initiation of mitigating actions to lessen the effects of potential hazards of smoke, chemical, radiological or toxic gas releases. The NRC staff considers the specific hazards and compensatory measures to be plant-specific, and did not

find sufficient information to conclude that the proposed changes are acceptable for these events without a plant-specific evaluation. In response to a NRC staff RAI, Reference 3 identifies that these mitigating actions were previously reviewed and approved by the NRC staff for TSTF traveler TSTF-448 (Reference 11). TSTF-448 authorizes a generic TS change to permit a 24-hour CT when the control room boundary is inoperable, and includes the same mitigating actions to assure protection of the control room staff from non-radiological hazards. Therefore, plants which have previously adopted TSTF-448 would not require any further plant-specific reviews in order to adopt the proposed change to TS 3.7.11.

Based on the risk impact being below the acceptance guidelines of RG 1.174, the additional restriction on meeting RCS specific activity limits in the TSs and the plant-specific information regarding smoke, chemical, radiological and toxic gas release mitigation, for plants that have not adopted TSTF-448, the NRC staff finds the proposed new action requirement and 24-hour CT acceptable.

3.5.9 LCO 3.7.12 – Control Room Emergency Air Temperature Control System

The CREATCS provides for temperature control of the control room when it is isolated during accident conditions. (For some CE plants, the cooling function is integrated with the system used for air cleanup in LCO 3.7.11.) This assures control room temperature will not exceed equipment operability requirements.

The standard TSs explicitly require entry into LCO 3.0.3 when both CREATCS trains are inoperable. The proposed change would provide a 24-hour CT to restore at least one CREATCS train to operable status, to permit continued operation under an existing action requirement.

The TR stated that the unavailability of the CREATCS has a negligible impact on severe accident risk, based on long room heatup times, availability of alternate cooling strategies, and alternate means to control emergency systems locally. The NRC staff reviewed the basis for this conclusion and considered the potential plant impacts if an accident occurred which isolated the control room while the CREATCS was inoperable.

If an accident occurred which isolated the control room without cooling, and core cooling was being maintained by the ECCS, then there would be negligible radiological consequences and the operators could simply unisolate and realign the normal control room ventilation system to provide continued cooling of the control room. Therefore, there would be no impact on CDF (i.e., $\Delta CDF = 0$).

If core damage occurred after the accident and the control room needed to remain isolated without cooling, the bounding impact would be to simply assume the event proceeded to a large early release based on the unavailability of the control room personnel to perform any mitigating actions. This assumption would be very conservative, since large releases occur primarily due to containment bypass accidents, and control room actions following core damage do not prevent the release from occurring.

A bounding estimate for CDF of CE plants was identified as 1E-4/year, so that over a 24-hour period the probability of a significant core damage event, which with the CREATCS unavailable is assumed to proceed to a large early release, would be:

(1E-4/year) x (24 hours) x (year/8760 hours) = 2.7E-7

Assuming a once per three year entry into the new TS, and assuming the entire 24-hour duration of the CT is used, the conservatively calculated ΔLERF is about 9.0E-8/year. This ΔLERF, and the zero ΔCDF, are below the acceptance guidelines of RG 1.174.

Defense-in-depth is provided by alternative control room cooling actions and by the capability for local operation of equipment, if necessary. These actions are typically found in plant procedures, and are not required to be implemented by TS controls.

Based on the risk impact being below the acceptance guidelines of RG 1.174, the NRC staff finds the proposed new action requirement and 24-hour CT acceptable.

3.5.10 LCO 3.6.6A – Containment Spray and Cooling Systems (credit taken for iodine removal)
LCO 3.6.6B – Containment Spray/Coolers (credit not taken for iodine removal)

The CS system and the containment coolers provide containment heat removal following accidents which release high energy steam to the containment. In addition to the heat removal function, the CS system enhances post accident fission product removal. Each CE plant has implemented either LCO 3.6.6A or 3.6.6B, based on whether the plant design credits the fission product removal function of the CS system. Each train of the CS system provides a nominal 50% of the cooling function, and similarly each train of the containment coolers provides 50% of the cooling function; thus the combined capacity of both systems is 200%.

The standard TSs for LCO 3.6.6A and 3.6.6B provide for an explicit LCO 3.0.3 entry when less than 100% containment cooling capacity is available (i.e., any combination of three or more trains inoperable). The proposed change would provide a 12-hour CT to restore at least any two trains to operable status, to permit continued operation under an existing action requirement.

For LCO 3.6.6A, when both CS trains are inoperable, and therefore the fission product removal function is not available, an explicit LCO 3.0.3 entry is required. The proposed change would provide a 72-hour CT to restore at least one CS train to operable status, to permit continued operation under an existing action requirement.

For the first proposed change regarding the loss of 100% capacity in the containment cooling system, the condition as stated in the TR represents total loss of any heat removal capability for the containment atmosphere (i.e., no CS or cooler train operable). No compensatory measures are identified to provide defense-in-depth for the heat removal function. The TR also states that unavailability of containment heat removal will result in failure of ECCS sump recirculation mode of operation, which could lead to core damage when the refueling water storage tank is depleted. Therefore, the NRC staff finds that this proposed change is not acceptable, because it results in a loss of safety functions for containment heat removal and core cooling by ECCS recirculation, with no mitigation available from other systems. In response to a NRC staff RAI,

Reference 3 proposed to withdraw these proposed changes to TS 3.6.6A and TS 3.6.6B for less than 100% of containment cooling capacity available.

For the second proposed change regarding the loss of all CS trains when the system is credited for fission product removal, the proposed CT of 72 hours is inconsistent with the 24-hour CT applied for all other systems related to fission product removal, and therefore not consistent with the intent of the TR to provide a short time period to restore one train to operable status. In addition, no compensatory measures were identified as was done for the other fission product removal systems. Therefore, the NRC staff finds that this change, as proposed, is not acceptable, because it results in a loss of safety function (fission product removal) for an unacceptable period of time without compensatory measures.

In response to a NRC staff RAI, Reference 3 proposed to revise the 72-hour CT for TS 3.6.6A to a 24-hour CT, consistent with the other iodine removal TS changes. Reference 3 also identified that the TS-required operability of the containment coolers would provide a similar iodine removal function such that additional TS requirements for operability of other iodine removal systems would not be required. A TS action for operability of the CREACS was proposed to assure additional defense-in-depth for control room functionality when both CS trains are inoperable during the 24-hour CT.

Based on the information in the TR, the challenge frequency of the CS system for fission product removal is identical to the challenge frequency described for the SBEACS, ICS, and PREACS, described in Section 3.5.6. Similar to those analyses, it may be conservatively assumed that if both CS trains are unavailable following a postulated core damage event, then some radioactive release above design limits, but well below the large early release level, would occur. A bounding estimate for CDF of CE plants was identified as 1E-4/year, so that over a 24-hour period the probability of a significant core damage event which would require the unavailable system would be:

(1E-4/year) x (24 hours) x (year/8760 hours) = 2.7E-7

Assuming a once per three year entry into the new TS would result in a frequency of a "less than LERF" release of about 9.0E-8/year. This frequency is within the acceptance guidance of RG 1.174 applicable to large early releases, and therefore, provides a context for consideration of the risk impact for smaller releases.

When the function of the CS for fission product removal is unavailable, then the operability of the CREACS, which provides for filtration to protect control room habitability, will be verified as a defense-in-depth measure.

Based on the zero risk impact for severe accidents being well below the acceptance guidelines of RG 1.174, and the verification of operability of the CREACS, the NRC staff finds a new action requirement with a 24-hour CT would be acceptable for the case of both CS trains inoperable.

4.0 CONCLUSION

The NRC staff has reviewed TR WCAP-16125, Rev. 2, using the five key principles of risk-informed decision making, and concludes that the proposed TS changes, with two exceptions,

are acceptable, within the limits and conditions identified in this safety evaluation and provided below:

LCO 3.1.9 – Boration Systems – Operating: Provide a new condition for both boration flowpaths inoperable and an associated action requirement to restore at least one path to operable with a 24-hour CT.

LCO 3.4.9 – Pressurizer Heaters: Provide a new condition for both class 1E heaters inoperable and an associated action requirement to restore at least one class 1E heater to operable with a 24-hour CT.

LCO 3.4.11 – Pressurizer PORVs and Associated Block Valves: Provide a new condition for both PORVs or both block valves inoperable and an associated action requirement to restore at least one PORV or block valve to operable with an 8-hour CT. The proposed change must include a requirement controlled in the TS action to confirm that the LCO for auxiliary feedwater is met, which requires both trains to be operable. In addition, the new 8-hour CT does not apply in the standard TS to PORVs which are leaking and unisolable. Plants implementing the proposed change must verify that their TS changes are consistent with the standard TS with regards to leaking and unisolable PORVs.

LCO 3.5.1 – Safety Injection Tanks: The staff does not find the proposed changes acceptable.

LCO 3.5.2 – ECCS – Operating (Low Pressure Safety Injection System): The staff does not find the proposed changes acceptable.

LCO 3.6.8 – Shield Building Exhaust Air Cleanup System: Provide a new condition for both SBEACS trains inoperable and an associated action requirement to restore at least one SBEACS train to operable with a 24-hour CT. The proposed change must include verification of operability of at least one train of the CS system when both SBEACS trains are inoperable.

LCO 3.6.10 – Iodine Cleanup System: Provide a new condition for both ICS trains inoperable and an associated action requirement to restore at least one ICS train to operable with a 24-hour CT. The proposed change must include verification of operability of at least one train of the CS system when both ICS trains are inoperable.

LCO 3.7.15 – Penetration Room Exhaust Air Cleanup System: Provide a new condition for both PREACS trains inoperable and an associated action requirement to restore at least one PREACS train to operable with a 24-hour CT. The proposed change must include verification of operability of at least one train of the CS system when both PREACS trains are inoperable.

LCO 3.7.13 – ECCS PREACS: Provide a new condition for both ECCS PREACS trains inoperable and an associated action requirement to restore at least one ECCS PREACS train to operable with a 24-hour CT. The proposed change is not applicable if the ECCS PREACS is required by the plant design basis for ECCS pump room cooling.

LCO 3.7.11 – Control Room Emergency Air Cleanup System: Provide a new condition for both CREACS trains inoperable and an associated action requirement to restore at least one CREACS train to operable with a 24-hour CT. The proposed change must include verification

that LCO 3.4.16, "RCS Specific Activity," is met, and must provide plant-specific information regarding smoke, chemical, radiological and toxic gas release mitigation unless TSTF-448 has previously been implemented.

LCO 3.7.12 – Control Room Emergency Air Temperature Control System: Provide a new condition for both CREATCS trains inoperable and an associated action requirement to restore at least one CREATCS train to operable with a 24-hour CT.

LCO 3.6.6A – Containment Spray and Cooling Systems (credit taken for iodine removal): Provide a new condition for both CS trains inoperable and an associated action requirement to restore at least one CS train to operable with a 24-hour CT. The proposed change must include verification that the LCO for CREACS is met. These proposed changes were identified in Reference 3 and supersede the proposed changes in the TR (Reference 2).

Additional changes proposed in the TR to address loss of containment cooling capability for LCO 3.6.6A and LCO 3.6.6B are not acceptable, and were withdrawn by Reference 3.

All the changes are acceptable only assuming that appropriate TS notes are provided which assure that the loss of safety function action requirements are not applicable for operational convenience and that voluntary entry into these action requirements in lieu of other alternatives that would not result in redundant systems or components being inoperable are prohibited.

The staff further notes that the proposed changes do not alter the regulations for notifications and reports required by 10 CFR 50 involving the loss of safety function, and that any plant-specific license amendment which provides a condition to address a loss of safety function would not obviate the requirement for a licensee to provide such notifications and reports.

5.0 REFERENCES

1. Letter from F. P. Schiffley II to U. S. NRC, Subject: Transmittal of WCAP-16125, Rev. 1, "Justification for Risk-Informed Modifications to Selected Technical Specifications for Conditions Leading to Exigent Plant Shutdown, (PA-LSC-0364)," dated January 7, 2008 (ADAMS Accession Number ML080390521).

2. Letter from D. Buschbaum to U. S. NRC, Subject: Responses to the NRC Request for Additional Information (RAI) on Topical Report (TR) WCAP-16125-NP, Revision 1, "Justification for Risk-Informed Modifications to Selected Technical Specifications for Conditions Leading to Exigent Plant Shutdown, (PA-LSC-0364 Revision 2)," dated August 10, 2009 (ADAMS Accession Number ML092260399).

3. Letter from D. Buschbaum to U. S. NRC, Subject: Responses to the NRC Request #2 for Additional Information (RAI) on Topical Report (TR) WCAP-16125-NP, Revision 1, "Justification for Risk-Informed Modifications to Selected Technical Specifications for Conditions Leading to Exigent Plant Shutdown, (PA-LSC-0364 Revision 2)," dated July 8, 2009 (ADAMS Accession Number ML091940063).

4. Regulatory Guide 1.174 Rev. 1, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," USNRC, dated November 2002.

5. Regulatory Guide 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications," USNRC, dated August 1998.

6. NUREG-0800, Standard Review Plan 19.2, "Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance," June 2007.
7. NUREG-0800, Standard Review Plan 16.1, "Risk-Informed Decisionmaking: Technical Specifications," Revision 1, dated March 2007.

8. NUREG/CR-5750, "Rates of Initiating Events at U. S. Nuclear Power Plants: 1987 – 1995," dated February 1999.

9. NUREG/CR-1570, "Risk Assessment of Severe Accident Induced Steam Generator Tube Rupture," dated March 1998.

10. NUREG/CR-6338, "Resolution of Direct Containment Heating Issue for all Westinghouse Plants with Large Dry Containments or Subatmospheric Containments," dated January 1996.

11. TSTF-448-A, "Control Room Habitability," Rev. 3, dated August 8, 2006 (ADAMS Accession Number ML062210095).

12. USNRC Final Policy Statement: Use of Probabilistic Risk Assessment in Nuclear Regulator Activities" *Federal Register* (60 FR 42622, August 16, 1995).

13. 10 CFR 50.46 "Acceptance Criteria for emergency core cooling systems for light-water nuclear power reactors."

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Date: May 24, 2010

RESOLUTION OF COMMENTS ON DRAFT SAFETY EVALUATION FOR

TOPICAL REPORT WCAP-16125-NP, REVISION 2

"JUSTIFICATION FOR RISK-INFORMED MODIFICATIONS TO

SELECTED TECHNICAL SPECIFICATIONS FOR CONDITIONS

LEADING TO EXIGENT PLANT SHUTDOWN"

PRESSURIZED WATER REACTOR OWNERS GROUP

PROJECT NO. 694

This Attachment provides the U.S. Nuclear Regulatory Commission (NRC) staff's review and disposition of the comments made by the Pressurized Water Reactor Owners Group (PWROG) on the draft safety evaluation (SE) for WCAP-16125-NP, Revision 2. The PWROG provided its comments in a letter dated November 17, 2009, "Comments to the NRC Draft Safety Evaluation for Pressurized Water Reactor Owners Group (PWROG) on Topical Report (TR) WCAP-16125-NP, Revision 2, "Justification for Risk-Informed Modifications to Selected Technical Specifications for Conditions Leading to Exigent Plant Shutdown," (PA-LSC-0364, Revision 2)."

Editorial changes suggested by the PWROG are noted in bold.

1. PWROG Comment (draft SE Page 4, Line 35):

Recommend modifying sentence as follows: **However, since these proposed changes to CTs are not intended for routine voluntary removal of redundant systems or components from service in lieu of other alternatives that would not result in redundant systems or components being inoperable**, ...

NRC Resolution for Comment 1 on the draft SE:

The NRC staff agrees with this comment, but has included the phrase "operational convenience" consistent with the standard technical specifications LCO 3.0.3.

2. PWROG Comment (draft SE Page 4, Lines 29 – 30):

Regulatory Guides 1.174 and 1.177 are only applicable to risk-informed completion times.

The following change is suggested: Revise "application of a particular TS action requirement" to "**risk-informed completion time.**"

NRC Resolution for Comment 2 on the draft SE:

The NRC staff has changed "TS action requirement" to "risk-informed completion time," retaining the qualifying clause "application of a particular."

3. PWROG Comment (draft SE Page 5, Line 23):

The following change is suggested: Revise "result in" to "**require**."

NRC Resolution for Comment 3 on the draft SE:

The NRC staff agrees with the proposed wording.

4. PWROG Comment (draft SE Page 5, Line 29):

The following change is suggested: Revise "unavailable" to "**inoperable**."

NRC Resolution for Comment 4 on the draft SE:

The NRC staff agrees with the proposed wording.

5. PWROG Comment (draft SE Page 8, Line 42):

The following change is suggested: Revise "administrative guidance" to "**administrative controls**."

NRC Resolution for Comment 5 on the draft SE:

The NRC staff agrees with the proposed wording.

6. PWROG Comment (draft SE Page 17, Lines 9 – 10):

The following change is suggested: "plant specific information...gas release mitigation," should be revised to "**plant specific information...gas release mitigation, unless TSTF-448 has been implemented**,"

NRC Resolution for Comment 6 on the draft SE:

The NRC staff modified the sentence to read "for plants that have not adopted TSTF-448" instead of "unless TSTF-448 has been implemented."

7. PWROG Comment (draft SE Page 18, Line 17):

The following change is suggested: Revise "uses" to "**has implemented either**."

NRC Resolution for Comment 7 on the draft SE:

The NRC staff agrees with the proposed wording.

8. PWROG Comment (draft SE Page 20, Line 43):

The following change is suggested: Revise "prohibited" to "**not applicable to**."

NRC Resolution for Comment 8 on the draft SE:

The NRC staff agrees with the proposed wording.

9. PWROG Comment (draft SE Page 21, Lines 17 - 19):

PWROG suggest replacing the text with: **The changes are not intended for routine voluntary removal of redundant systems or components from service in lieu of other alternatives that would not result in redundant systems or components being inoperable.**

NRC Resolution for Comment 9 on the draft SE:

The NRC staff disagrees with proposed change. Voluntarily entering a 24-hour action with no operable train is not acceptable. The text has been revised to be consistent with the existing TS Bases for LCO 3.0.3. However, the proposed TS changes to implement this initiative must include prohibitions to address voluntary entry into these actions when other alternatives exist, consistent with the existing LCO 3.0.3.

PWR Owners Group
Member Participation* for Project/Task PA-LSC-0364

Utility Member	Plant Site(s)	Participant	
		Yes	No
AmerenUE	Callaway (W)		X
American Electric Power	D.C. Cook 1&2 (W)		X
Arizona Public Service	Palo Verde Unit 1, 2, & 3 (CE)	X	
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
Dominion Kewaunee	Kewaunee (W)		X
Dominion VA	North Anna 1 & 2, Surry 1 & 2 (W)		X
Duke Energy	Catawba 1 & 2, McGuire 1 & 2 (W), Oconee 1, 2, 3 (B&W)		X
Entergy	Palisades (CE)	X	
Entergy Nuclear Northeast	Indian Point 2 & 3 (W)		X
Entergy Operations South	Arkansas 2, Waterford 3 (CE), Arkansas 1 (B&W)	X	
Exelon Generation Co. LLC	Braidwood 1 & 2, Byron 1 & 2 (W), TMI 1 (B&W)		X
FirstEnergy Nuclear Operating Co	Beaver Valley 1 & 2 (W), Davis-Besse (B&W)		X
Florida Power & Light Group	St. Lucie 1 & 2 (CE)	X	
Florida Power & Light Group	Turkey Point 3 & 4, Seabrook (W)		X
Luminant Power	Comanche Peak 1 & 2 (W)		X
Nuclear Management Company	Prairie Island 1&2, Pt. Beach 1&2 (W)		X
Omaha Public Power District	Fort Calhoun (CE)	X	
Pacific Gas & Electric	Diablo Canyon 1 & 2 (W)		X
Progress Energy	Robinson 2, Shearon Harris (W), Crystal River 3 (B&W)		X
PSEG - Nuclear	Salem 1 & 2 (W)		X
Southern California Edison	SONGS 2 & 3 (CE)	X	
South Carolina Electric & Gas	V.C. Summer (W)		X
So. Texas Project Nuclear Operating Co.	South Texas Project 1 & 2 (W)		X
Southern Nuclear Operating Co.	Farley 1 & 2, Vogtle 1 & 2 (W)		X
Tennessee Valley Authority	Sequoyah 1 & 2, Watts Bar (W)		X
Wolf Creek Nuclear Operating Co.	Wolf Creek (W)		X

* **Project participants as of the date the final deliverable was completed. On occasion, additional members will join a project. Please contact the PWR Owners Group Program Management Office to verify participation before sending this document to participants not listed above.**

PWR Owners Group
International Member Participation* for Project/Task PA-LSC-0364

Utility Member	Plant Site(s)	Participant	
		Yes	No
British Energy	Sizewell B		X
Electrabel (Belgian Utilities)	Doel 1, 2 & 4, Tihange 1 & 3		X
Kansai Electric Co., LTD	Mihama 1, Ohi 1 & 2, Takahama 1 (W)		X
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	
Nuklearna Electrarna KRSKO	Krsko (W)		X
Nordostschweizerische Kraftwerke AG (NOK)	Beznau 1 & 2 (W)		X
Ringhals AB	Ringhals 2, 3 & 4 (W)		X
Spanish Utilities	Asco 1 & 2, Vandellos 2, Almaraz 1 & 2 (W)		X
Taiwan Power Co.	Maanshan 1 & 2 (W)		X
Electricite de France	54 Units		X

* **This is a list of participants in this project as of the date the final deliverable was completed. On occasion, additional members will join a project. Please contact the PWR Owners Group Program Management Office to verify participation before sending documents to participants not listed above.**

ACKNOWLEDGEMENT

The author wishes to acknowledge James Metcalf of Polestar Applied Technologies for support in the development of the radiological release evaluations contained in this report, and Brian Mann of Excel Services Corporation and James Andracheck of Westinghouse for help in coordinating the topical report with the technical specifications markups.

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) (also referred to as 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 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 LCO 3.0.3 entry) or conditions exist where the specific Action includes a 1 hour shutdown requirement (similar to an 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 defense-in-depth measures within the TS Action Statement framework.

When using risk information to modify existing plant Technical Specifications (TSs), Regulatory Guide 1.177 requires an assessment is to be performed evaluating the impact of the TS modification to Defense in Depth and Safety Margins and a three tiered risk informed process be followed. Tier 1 of the process involves a risk impact assessment of the proposed TS change. Tier 2 involves "an identification of potentially high-risk configurations 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." The objective of the Tier 2 evaluation is to ensure that appropriate restrictions on dominant risk-significant configurations associated with the change are in place. Tier 3 considerations involve the application of a configuration risk management program to assess the risk associated with emergent equipment unavailabilities. Specific Tier 3 considerations

presented in this report are intended to be included within the plant configuration risk management program. Actions identified as Tier 3 are intended to reduce the impact of the degraded plant configuration, expedite component repair, and increase the awareness of the unavailability. Changes proposed in this document have also been considered in the development of TSTF 426.

An evaluation of the TS changes proposed in this report are provided in Section 5. That section specifically identifies the risk impact of the proposed change (Tier 1), addresses defense in depth considerations and discusses any Tier 2 or Tier 3 impacts. In several instances, inoperability of TS systems evaluated in this effort did not result in situations where concurrent inoperability of other components, or concurrent plant actions would result in high risk. Hence, in the absence of a high risk condition the Tier 2 action was often identified as "None." Where appropriate, specific actions that should be considered in the plant Configuration Risk Management Program (CRMP) are specifically identified as Tier 3 actions.

In many instances the defense in depth assessment concluded that specific actions should be taken to mitigate the potential consequences of the system inoperability. In several instances, it is recommended that specific actions be placed either directly in the Technical Specifications or in the plant administrative controls. These actions will limit the consequences of the associated plant inoperability. As the proposed changes are both infrequent and of very short duration, there is negligible impact on plant safety margins.

Revision 2 of this document has been developed to incorporate the responses to an NRC RAI on Revision 1. The more significant changes included in this revision are:

1. Re-calculation of the ICCDP and ICLERP risk metrics to reflect the full AOT rather than AOT-1 hour.
2. Addition of Tables 4.2.6 and 4.2.7 comparing the risk metrics associated with the proposed TS change to regulatory guide goals identified in RG 1.174 and RG 1.177.
3. The requested AOT extension for the PORV and the CSS TS has been reduced to meet RG 1.177 limits.
4. The "Basis for Proposed Change" sections included in Section 5 has been upgraded to include a discussion of risk metrics, defense in depth, and compensatory measures.
5. Where appropriate, Tier 2 sections have been expanded to explicitly identify those actions considered to be included in this category.
6. The defense in depth actions required to be included in either the Technical Specification or administrative guidance has been clearly identified.
7. Requests for any TS end state modifications have been removed.
8. To clarify the implementation actions, Table 2-2 has been added. This table summarizes the commitments associated with the adoption of the various Technical Specifications changes requested in the Topical Report.

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 hour AOT 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. Table 2-2 summarizes the associated conditions for implementation of the proposed AOT extensions. These conditions are discussed in Section 5 and represent the results of the defense in depth and Tier 2 and 3 assessments for the various proposed TS modifications.

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 Risk Impacts Resulting from Proposed Modifications to Technical Specifications

NUREG-1432 LCO	SYSTEM/ COMPONENT	CONDITION	CURRENT COMPLETION TIME[8]	PROPOSED COMPLETION TIME TO RESTORE [ONE] TRAIN	ICCDP (See Notes 1 & 2)	ICLERP (See Note 1)
3.1.9 (NA-ISTS)	Boration System	System Inoperable	No Condition defined. Default LCO 3.0.3 entry.	24 hrs	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	3.0E-7 (See Note 9)	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(Note 5)	4 hrs. Confirm LCO 3.7.5 met	4.6E-7	3.4E-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	< 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).	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. 8 hours if two Containment Spray Trains and two Containment Cooling Trains inoperable.	5E-7 (when CARC not available) Insignificant impact for PWRs with diverse containment cooling systems (See Note 6)	Note 10
3.6.6.B	CSS (See Note 4)	Two Containment Spray Trains and two Containment Cooling Trains inoperable	Explicit LCO 3.0.3 entry	8 hrs	5E-7 (when CARC not available) (See Note 6)	Note 10
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	Note 7	Note 7

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

NUREG-1432 LCO	SYSTEM/ COMPONENT	CONDITION	CURRENT COMPLETION TIME[8]	PROPOSED COMPLETION TIME TO RESTORE [ONE] TRAIN	ICCDP (See Notes 1 & 2)	ICLERP (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.	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.	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.	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.	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	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 full AOT (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. TSTF-422 provides guidance for implementing risk informed end states.
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 No CDF or LERF contribution to component inoperability (See Section 4.4).
8 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). A licensee must implement TSTF-422-A to justify the risk informed end states.
9 Assumes probability of manual RCS pressure control is high.
10 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

Table 2-2
Summary of Conditions of Implementation

NUREG-1432	SYSTEM/ COMPONENT	CONDITION	PROPOSED COMPLETION TIME TO RESTORE [ONE] TRAIN	REQUIREMENTS/CONDITIONS FOR ADOPTION OF AOT EXTENSION
3.1.9 (NA-ISTS)	Boration System	System Inoperable	24 hrs	When using the extended AOT, administrative procedures will verify the availability of at least one HPSI subsystem.
3.4.9	Pressurizer Heaters	Two Groups of Class 1E Heaters Inoperable	24 hrs	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 is included in TSTF-426.
3.4.11	PORVs	Inability of two PORVs to Open, or Inability of both PORVs to close and block valves to be closed	4 hrs. Confirm LCO 3.7.5 is met .	A Required Action requires verification that LCO 3.7.5, Auxiliary Feedwater System, is met when the extended completion time is used. This requires both trains of AFW to be Operable. The AFW LCO is applicable in Modes 1, 2 and 3. The AOT 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.
3.5.1	SITs	Two or More SITs Inoperable	24 hrs to return all but one SIT to Operable status. All ECCS trains are Operable.	A Required Action requires verification that LCO 3.5.2, ECCS - Operating, is met when the extended completion time is used. This requires both trains of ECCS to be Operable.
3.5.2	LPSI	Two LPSI subsystems Inoperable	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).	A Required Action requires verification that LCO 3.5.1, SITs, is met when the extended completion time is used. This requires all SITs to be Operable. Both HPSI subsystems must also be Operable. This is true for plants that have requirements consistent with NUREG-1432. 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. Plants with TS not consistent with NUREG-1432 should verify that these restrictions are implemented in their TS. CRMP should include guidance on availability of leak before break detection system. A note should be added to CRMP to note that during inoperability 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.

Table 2-2
Summary of Conditions of Implementation

NUREG-1432	SYSTEM/ COMPONENT	CONDITION	PROPOSED COMPLETION TIME TO RESTORE [ONE] TRAIN	REQUIREMENTS/CONDITIONS FOR ADOPTION OF AOT EXTENSION
3.6.6.A	CSS	Two Containment Spray Trains Inoperable	72 hours if two Containment Spray Trains inoperable.	None
		Two Containment Spray Trains and two Containment Cooling Trains inoperable.	8 hours if two Containment Spray Trains and two Containment Cooling Trains inoperable.	Plant guidance, procedures and equipment should be available for refill of the RWST To minimize the potential for a containment challenges and ensure defense in depth this later condition (both trains of CS and CARCs inoperable) should not be entered with concurrent inoperabilities of the AFW or ECCS system.
3.6.6.B	CSS	Two Containment Spray Trains and two Containment Cooling Trains inoperable	8 hrs	Plant guidance, procedures and equipment should be available for refill of the RWST To minimize the potential for containment challenges and ensure defense in depth, this later condition (both trains of CS and CARCs inoperable) should not be entered with concurrent inoperabilities of the AFW or ECCS system.
3.6.8	SBEACS	Two Trains Inoperable	24 hrs. Confirm availability of one train of CSS.	A Required Action requires verification that one train of containment spray is Operable. It should be noted in the in appropriate administrative guidance that availability of other plant air cleanup systems will marginally reduce impact of SBEACS unavailability.
3.6.10	ICS	Two Trains Inoperable	24 hrs. Confirm operability of one train of CSS.	A Required Action requires verification that one train of containment spray is Operable. It should be noted in the in appropriate administrative guidance that availability of other plant air cleanup systems will marginally reduce impact of system unavailability.
3.7.11	CREACS	Two Trains Inoperable	24 hrs. Confirm LCO 3.4.16 is met. Upon entry, initiate action to implement mitigating actions.	Required Actions require: 1) verification that LCO 3.4.16, "RCS Specific Activity," and 2) initiation of actions to implement mitigating actions consistent with the actions already required under Condition A when the Control Room barrier is inoperable. It should be noted in the appropriate administrative guidance that operability of other plant air cleanup systems, the CSS and containment air cooling units (if available) will reduce impact of system unavailability. Plant administrative controls which implement 10CFR50.65 (a)(4) should consider the status of the following technical specifications in assessing and managing plant risk while operating: TS 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 that the availability of alternate shutdown panels and local shutdown stations should remote actions become necessary.
3.7.12	CREATCS	Two Trains Inoperable in MODE 1, 2, 3 or 4	24 hrs.	Administrative guidance/procedures should be in place for alternate means of establishing temporary CR cooling.

Table 2-2
Summary of Conditions of Implementation

NUREG-1432	SYSTEM/ COMPONENT	CONDITION	PROPOSED COMPLETION TIME TO RESTORE [ONE] TRAIN	REQUIREMENTS/CONDITIONS FOR ADOPTION OF AOT EXTENSION
3.7.13	ECCS PREACS	Two Trains Inoperable	24 hrs. Confirm operability of one train of CREACS.	A Required Action requires verification that at least one train of Control Room Emergency Air Cleanup System be Operable when the extended completion time is used. 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 Phosphate" when in this condition.
3.7.15	PREACS	Two Trains Inoperable	24 hrs Confirm operability of one train of CSS.	A Required Action requires verification that at least one train of Control Room Emergency Air Cleanup System be Operable when the extended completion time is used. Administrative CRMP guidance should include a recommendation to maintain at least one train of CREACS available upon entry into this condition It should be noted in the appropriate administrative guidance that availability of other plant air cleanup systems will marginally reduce impact of system unavailability

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.

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 requested time intervals 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.

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 necessary, using compensatory measures to limit concurrent unavailabilities and evaluating repair activities and alternatives. In some instances the required defense-in-depth 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. It is further assumed that the unavailability of the affected system will lead to all associated events to progress towards core damage. Potential mitigating strategies not credited in this analysis and other associated conservatisms are summarized in response to request for additional information question 4 (Reference 21).

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

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

where:

$ICCDP_{goal}$ = 1.0E-6

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

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

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

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

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

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

where:

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

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

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

	Table 4.1-1: Mapping of Mitigating Components and Frequency of Events Mitigated (a)								
System/Component Unavailable	**Event Frequency (per year)**								**Component Challenge Frequency (per year) (g)**
	LBLOCA	**MBLOCA**	**SBLOCA**	**SGTR**	**Stuck Open PORV**	**Stuck Open PSV**	**Events Leading to F&B**	**ATWS**	
SIT	5.0E-06	(b)	(b)	(b)	(b)	(b)	(b)	(b)	5.0E-06
LPSI	5.0E-06	4.0E-05	(d)	(b)	(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 x 1.2E-5 = 1.68E-05 per year (value rounded up to 1.7E-05 per year).

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Table 4.1-2 Time (hrs) for an Unavailable System to Accumulate an Incremental CDP of 1.0 E-06				
System/Component Unavailable	**Mean Challenge Frequency (1/yr)**	**Time (hours) to reach CDP=1.0 E-06**	**Proposed AOT (hours)**	**ICCDP**
SIT	5.00E-06	1752	24	1.37E-08
LPSI	4.50E-05	195	24	1.23E-07
CS (no CARC available)	5.50E-04	16	8	5.02E-07
PORV	1.01E-03	9	4	4.60E-07
Boration	1.70E-05	515	24	4.66E-08

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 1.01E-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 failing to meet 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 of Proposed Change in Core Damage Frequencies

System/Component Unavailable	ICCDP	Change in CDF (per year)	
SIT	1.37E-08	4.57E-09	2.74E-09
LPSI	1.23E-07	4.11E-08	2.47E-08
CS (no CARC available)	5.02E-07	1.67E-07	1.00E-07
PORV	4.60E-07	1.53E-07	9.21E-08
Boration	4.66E-08	1.55E-08	9.32E-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:

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

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

In this case, the unavailability of the Class 1E pressurizer heaters is assumed to increase the plant trip potential 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. The associated changes in core damage frequency for losing all pressurizer heaters once-in-5 years and once-in-3 years are 6 E-08 per year and 1 E-07 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 vary 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.

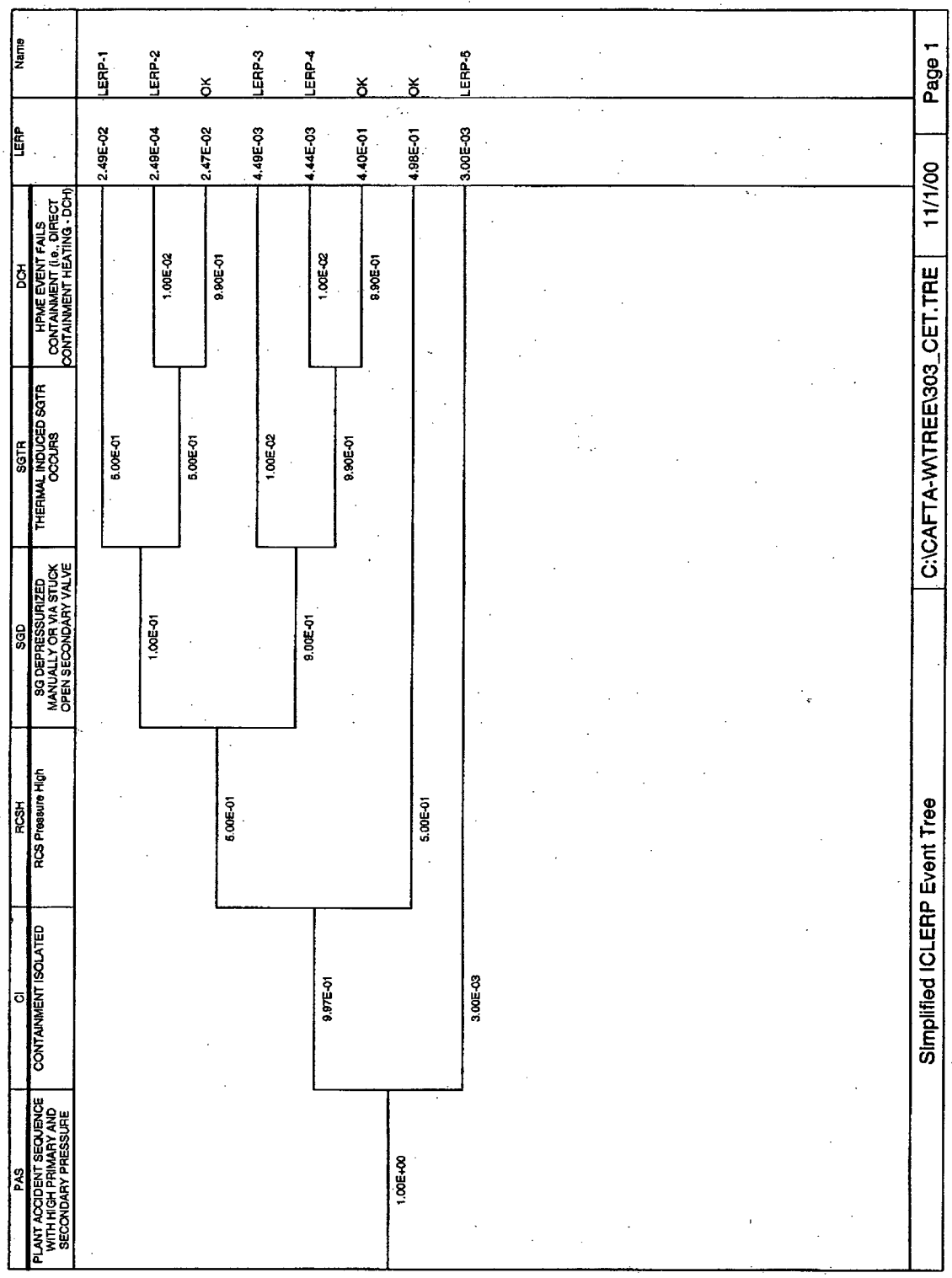

Figure 4.2-1: Simplified Incremental Large Early Release Event Tree

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)	8	**5.0E-7**	1.0E-2	1.1E-4	1.8E-3	2.0E-3	3.0E-3	1.7E-2	**9.8E-9**
PORV	4	**4.6E-7**	5.0E-2	5.5E-4	9.0E-3	9.8E-3	3.0E-3	7.2E-2	**3.4E-8**
Boration Systems	24	**4.66E-8**	5.0E-2	5.5E-4	9.0E-3	9.8E-3	3.0E-3	7.2E-2	**3.4E-9**
Pressurizer Heaters	24	**3.00E-7** (Note 1)	2.5E-2	2.7E-4	4.5E-3	4.9E-3	3.0E-3	3.8E-2	**1.1E-8**

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

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

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

where:

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

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

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

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
Potential Risk of Proposed Change in Large Early Release Frequencies

System/Component Unavailable	ICCDP	LERP	ICLERP	Change in LERF (per year)	
				1-in-3 yr. Entry	1-in-5 yr. Entry
SIT	1.37E-08	3.00E-03	4.11E-11	1.37E-11	8.22E-12
LPSI	1.23E-07	3.00E-03	3.70E-10	1.23E-10	7.40E-11
CS (no CARC available)	5E-07	1.68E-02	8.4E-09	2.81E-09	1.7E-09
PORV	4.6E-07	7.22E-02	3.3E-08	1.1E-08	6.6E-09
Boration	4.66E-08	7.22E-02	3.36E-09	1.12E-09	6.73E-10
Pressurizer Heaters	3.00E-07	3.76E-02	1.13E-08	3.76E-09	2.26E-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 indicates 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	Change in CDF (per year)		LERP	ICLERP	Change in LERF (per year)	
		1-in-3 yr. Entry	1-in-5 yr. Entry			1-in-3 yr. Entry	1-in-5 yr. Entry
SIT	1.37E-08	4.57E-09	2.74E-09	1.00E-02	1.37E-10	4.57E-11	2.74E-11
LPSI	1.23E-07	4.11E-08	2.47E-08	1.00E-02	1.23E-09	4.11E-10	2.47E-10
CS (no CARC available)	5.02E-07	1.67E-07	1.00E-07	7.23E-02	3.63E-08	1.21E-08	7.26E-09
PORV	4.60E-07	1.53E-07	9.21E-08	3.22E-01	1.48E-07	4.94E-08	2.97E-08
Boration	4.66E-08	1.55E-08	9.32E-09	3.22E-01	1.50E-08	5.00E-09	3.00E-09
Pressurizer Heaters	3.00E-07	1.00E-07	6.00E-08	1.66E-01	4.98E-08	1.66E-08	9.96E-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) Risk Impact Associated with Bounding (95th %) Initiating Event Frequencies

The initiating event frequencies contributing to the overall challenge frequency or CDF for each system/component were statistically combined. Each initiating event frequency was assumed to be log-normally distributed. The 95th % 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 95th % Upper Bound ICCDPs

System/Component Unavailable	Proposed AOT (hours)	95th Upper Bound Frequency (per year)	ICCDP
SIT	24	1.00E-05	2.74E-08
LPSI	24	1.10E-04	3.01E-07
CS (no CARC available)	8	1.10E-03	1.00E-06
PORV	4	2.50E-03	1.14E-06
Boration Systems	24	2.50E-05	6.85E-08

Table 4.2-5 Summary of the Risk Impact for 95th % Bounding Frequencies							
System/Component Unavailable	ICCDP	Change in CDF (per year)		LERP	ICLERP	Change in LERF (per year)	
		1-in-3 yr. Entry	1-in-5 yr. Entry			1-in-3 yr. Entry	1-in-5 yr. Entry
SIT	2.74E-08	9.13E-09	5.48E-09	3.00E-03	8.22E-11	2.74E-11	1.64E-11
LPSI	3.01E-07	1.00E-07	6.03E-08	3.00E-03	9.04E-10	3.01E-10	1.81E-10
CS (no CARC available)	1.00E-06	3.35E-07	2.01E-07	1.68E-02	1.69E-08	5.63E-09	3.38E-09
PORV	1.14E-06	3.81E-07	2.28E-07	7.22E-02	8.24E-08	2.75E-08	1.65E-08
Boration	6.85E-08	2.28E-08	1.37E-08	7.22E-02	4.95E-09	1.65E-09	9.89E-10
Pressurizer Heaters	3.00E-07	1.00E-07	6.00E-08	3.76E-02	1.13E-08	3.76E-09	2.26E-09

As indicated 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.

4.2.5 Final Comments and Discussion of Risk Metrics

Sections 4.1 and 4.2 provided estimates of the impact on the proposed changes to SIT, LPSI, CSS, PORV, Boration System and pressurizer heater Technical Specifications. The estimates include both the individual entry and incremental annual risks as measured by the increase in CDF, LERF, ICCDP and ICLERP. Tables 4.2-6 and 4.2-7 provide comparisons of these metrics to regulatory guidelines for small annual risk increases (RG.1.174) and small single entry incremental risk increases (RG 1.177). The proposed changes meet the RG 1.174 metrics for very small risk changes. It should also be noted that while not explicitly calculated there exists an offsetting risk of entering a plant shutdown to resolve the issue at another mode. Thus, the absolute risks of resolution of the inoperability "at power" are sufficiently low so that the potential incremental risk to the public (if any) is very small.

Table 4.2-6 Comparison of Estimated Incremental Risks to RG 1.174 Guidelines for "Permanent" Change				
System	**Change in CDF (1/yr)**	**RG 1.174 Guideline Metric for "Small Change" in Mean CDF (per year)**	**Change in LERF (per year)**	**RG 1.174 "Small Change" in Mean LERF (per year)**
SIT	4.57E-09	1.00E-06	1.37E-11	1.00E-07
LPSI	4.11E-08	1.00E-06	1.23E-10	1.00E-07
CS (no CARC available)	1.67E-07	1.00E-06	2.81E-09	1.00E-07
PORV	1.53E-07	1.00E-06	1.1E-08	1.00E-07
Boration Systems	1.55E-08	1.00E-06	1.12E-09	1.00E-07
Pressurizer Heaters (once in 3 years)	1.00E-07	1.00E-06	3.76E-09	1.00E-07

Table 4.2-7 Comparison of Estimated Incremental Risks to RG 1.177 Guidelines for TS Change				
System	**ICCDP**	**RG 1.177 Guidance for "Small Change" in Mean CDF (per year)**	**ICLERP**	**RG 1.177 Guidance for "Small Change" in Mean LERF (per year)**
SIT	1.37E-08	5.0E-07	4.11E-11	5.0E-08
LPSI	1.23E-07	5.0E-07	3.70E-10	5.0E-08
CS (no CARC available)	5E-07	5.0E-07	8.4E-09	5.0E-08
PORV	4.6E-07	5.0E-07	3.3E-08	5.0E-08
Boration Systems	4.66E-08	5.0E-07	3.36E-09	5.0E-08
Pressurizer Heaters	3.00E-07	5.0E-07	1.13E-08	5.0E-08

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 (i.e., $\Delta CDF = \Delta LERF = 0$). 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 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 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 4×10^{-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 1×10^{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)	2.5 to 30 x 10^{+6}	< 3.5 x 10^{+3}	< 20
(All other radiation mitigation systems inoperable)			
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 not met, the plant is required to verify that RCS specific activity is within the limits of Figure 3.4.16-1 by taking RCS 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 events, inoperability of both trains of control room ventilation system can potentially subject the plant staff to dose levels exceeding the licensing basis dose limit (typically 5 rem) and therefore actions to mitigate the potential challenge is important to supporting an AOT extension to 24 hours.

To support defense in depth, a review of the 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 including the following as Required Actions in the Technical Specifications as a condition of use of the extended AOT:

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

and

(2) Implementing mitigating actions such as:

- Increasing plant awareness to the inoperability, verifying operability of radiation monitors, and verifying availability of mitigating equipment, such as SCBAs.

Note that the 24 hour AOT requested herein is consistent with NUREG-1432 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 used to ensure the operators can function in the control room and to ensure control room equipment will function. The CRMP will include 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. Administrative guidance will be provided to monitor the control room temperature to ensure habitability of the control room and operability of TS equipment.

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, as discussed below) 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, consequently, radionuclide releases will be well below Part 100 limits.

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)[1]	Challenge Frequency (per year)	Probability of System Challenge Given Extended Entry into Proposed AOT
Iodine Cleanup System	24	1.0E-4[2]	2.7E-7
Shield Bldg. EACS	24	1.0E-4[2]	2.7E-7
CR EACS/EATCS	24	1.0E-4[2]	2.7E-7
PREACS	24	1.0E-4[2]	2.7E-7
ECCS – PREACS	24	4.5E-5[3]	1.2E-7
CS[4]	72	1.0E-4[2]	8.1E-7

(1) Both trains inoperable.
(2) Representative Bounding Estimate of Total Core Damage Frequency.
(3) Frequency of entry into recirculation.
(4) With Emergency Grade containment air cooler 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 remains 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 $\approx$ 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 risk due to these proposed TS modifications is negligible.

4.5 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 operation in accordance with of the proposed AOTs.

4.6 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 Recommendations
x) Tier 3 Recommendations

The TS changes being proposed generally are associated with the inoperability of an entire system (or unavailability of a given function). To address this loss of function, public safety is maintained by ensuring public risk is acceptably low, by taking compensatory measures to provide defense in depth using alternate systems, 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.

The requirement for an immediate (1 hour) shutdown is usually based on the loss of an important safety function. The 1 hour time to initiate a plant shutdown was chosen to provide time to prepare for the plant evolution. The goal is to place the plant in a condition where the health and safety of the public could be better assured, but in a controlled manner. However, no specific risk assessments were performed to determine the risk significance of 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 systems or components "at power" rather than to initiate a shutdown in one hour.

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 which are reflected in the Technical Specification Required Actions, avoidance of high risk conditions (Tier 2 requirements) and "good practices" (Tier 3 recommendations), which are included in licensee-controlled procedures and administrative guidance.

Tier 3 actions are generally associated with configuration risk management actions and are embedded within the plant maintenance rule program. 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 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 requires 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.

Since LCO 3.1.9 does not appear in NUREG-1432, it is not included in TSTF-426.

Plant Applicability

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

Limiting Condition for Operation (LCO)

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

Licensing Basis for LCO

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

Condition Requiring Entry into a Shutdown Action Statement

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

Proposed Modification to Actions

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

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 incremental conditional core damage probability (ICCDP) is about 4.66E-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.

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 3.36E-9. Even then, the resulting ICLERP is well below the RG 1.177 incremental risk (ICLERP) goal 5.0E-8 for a TS change.

An assessment of RG 1.174 risk metrics indicates that as a result of the low frequency of entry into this condition, the increase in core damage frequency and large early release frequency are 1.55E-08 per year and 1.12E-09 per year, respectively. These values are in the Region III (very small) risk increase region for the RG 1.174 risk metric for both change in CDF and LERF

Defense-in-Depth Consideration

In the event that both boration paths are inoperable, 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. This action will be considered in the plant administrative guidance.

Tier 2 Recommendations

None.

Tier 3 Recommendations

None.

5.1.2 (ISTS) LCO 3.4.9 – Pressurizer Heaters

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

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

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

Plant Applicability

All except St. Lucie 2

Limiting Condition for Operation (LCO)

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

Licensing Basis for LCO

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

Condition Requiring Entry into a Shutdown Action Statement

Default entry into LCO 3.0.3 is required when 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, and/or adjusting RCS make-up flow rate.

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 3E-7 (See Section 4.1.2). The resulting LERP increment is 1.15E-8 (See Section 4.2). Based on a bounding TS entry frequency of once per three years per plant, the incremental CDF and LERF associated with this inoperability are 1E-07 per year and 3.8 E-09 per year, respectively. Both results are in Region III of the RG 1.174 incremental risk guidelines (very small risk changes) and well below the derivative RG 1.177 guidance for ICCDP and ILERP (see Section 4, see also Tables 4.2.6 and 4.2.7).

Note that, 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, and RCS heat removal (feeding and steaming a SG to the atmosphere).

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 is included in TSTF-426.

Tier 2 Recommendations

None.

Tier 3 Recommendations

None.

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	Required Action End State when AOT 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) AOT to be consistent with other CE designed NSSSs (with PORVs) to allow 4 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 AOT to allow 4 hours to restore one block valve to Operable 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 Pressurizer Safety Valves, and the design basis analyses do not take credit for the PORV opening for accident mitigation. Section 4.1 indicates that the increased CDP associated with extending the AOT to 4 hours for inoperable PORVs (unable to open) is 4.6E-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. The ICLERP associated with this inoperability is 3.3E-08. Increases in plant Core Damage Frequency and Large Early Release Frequency based on a one in three year entry into this TS condition is estimated to be less than 1.5E-07 per year and 1.1E-08 per year, respectively. Both results are in Region III of the RG 1.174 incremental risk guidelines (very small risk changes) and well below the derivative RG 1.177 guidance for ICCDP and ILERP (see Section 4, see also Tables 4.2.6 and 4.2.7).

The 4 hr AOT 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.

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 in the event of a total loss of feedwater was identified post-TMI. This feature is risk significant and impacts core damage frequency. To minimize impact of the proposed change, 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. This requires both trains of the AFW System to be Operable and applies to Modes 1, 2 and 3.

Tier 2 Recommendations

None.

Tier 3 Recommendations

None.

5.1.4 (ISTS) LCO 3.5.1 – Safety Injection Tanks

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

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

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, LCO 3.0.3 must be entered immediately. The SIT 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 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

Insert a new Condition C to allow 24 hours to restore one SIT to Operable status when two or more SITs are inoperable, 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 AOT is 1.37E-8. Similarly for LERP, the conservative bounding calculation results in an ICLERP of 4.11E-11. Increases in plant Core Damage Frequency and Large Early Release Frequency based on a one in three year entry into this TS condition is estimated to be less than 4.57 E-09 per year and 1.37E-11 per year, respectively. Both results are within Region III (very small risk) of the RG 1.174 risk map and are well below the RG 1.174 incremental risk guideline. ICCDP and ICLERP values are well below the single entry small risk guideline defined in RG 1.177 TS. (See also Tables 4.2.6 and 4.2.7). These results confirm that the risk impact of the 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. A required action to verify that both trains of ECCS are Operable is included in the Technical Specifications.

The proposed AOT is consistent with the requirements of 10CFR50.46 which require that the license propose immediate steps to "bring plant design or operation" into compliance by ensuring the defined outage time is commensurate with the risk significance of the system. Availability of both trains of ECCS will limit the impact of SIT unavailability.

Tier 2 Recommendations

None.

Tier 3 Recommendations

None.

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.

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, an explicit entry into LCO 3.0:3 is required.

Proposed Modification to Actions

It is proposed that a 24-hour AOT for restoration of at least one inoperable LPSI subsystem be added. This condition will also require verification within 1 hour that all SITs are Operable. 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 short duration 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. 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 1.23E-7 and the ICLERP is 3.7E-10 for the proposed 24 hour AOT duration. These results are offset by the risk of transitioning the plant to a shutdown mode (See References 4 and 8). Increases in plant Core Damage Frequency and Large Early Release Frequency based on a one in three year entry into this TS condition are estimated to be less than 4.11E-08 per year and 1.23E-10 per year, respectively. Both results are within Region III (very small risk) of the RG 1.174 risk map and are well below the RG 1.174 incremental risk guideline. ICCDP and ICLERP values are well below the single entry small risk guideline defined in RG 1.177 TS. (See also Tables 4.2.6 and 4.2.7). These results confirm that the risk impact of the AOT extension is negligible.

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 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 assumed in large 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. A Technical Specification Required Action requires verification that the SITs are Operable in order to utilize the proposed AOT. The ISTS does not allow the use of the proposed AOT unless all HPSIs are Operable.

Tier 2 Recommendations

None.

Tier 3 Recommendations

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.

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 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).

Plant Applicability

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

Limiting Conditions For Operation (LCO)

Two SBEACS trains shall be Operable in Modes 1, 2, 3, and 4.

Licensing Basis for LCO

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).

Condition Requiring Entry into Shutdown Action Statement

Both trains inoperable.

Proposed Modification to Actions

Allow 24 hours to restore at least one SBEACS train when both SBEACS trains are inoperable

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, ICLERP and will not change the plant CDF or LERF. Since extending the AOT on these systems has no impact on the aforementioned risk metrics, the impact of the proposed modification to the TS for this system is below the RG 1.174 incremental risk guidelines and derivative RG 1.177 TS guidance. Note that the basis for not crediting this system as a means to reduce LERF or ICLERP is that the associated filter system is not designed to accommodate fission product or annulus pressure loads associated with containment failure and bypass scenarios. 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. The frequency of these smaller radiological releases is estimated to be between 1E-04 and 0.01 events per year, with the greater event frequency being associated with lesser releases (See Table 4.3-1). Availability of at least one train of CSS and alternate radiological cleanup systems, as discussed in defense in depth section, provides significant means of limiting radiological releases to the public from these lesser releases.

Defense-in-Depth Consideration

To minimize the impact of unavailability of the SBEACS, the Technical Specifications will require that at least one train of Containment Spray is Operable. Containment spray can effectively scrub the post accident containment atmosphere of fission products and reduces the reliance on downstream air cleanup systems.

Tier 2 Recommendations

None.

Tier 3 Recommendations

It should be noted in the in appropriate administrative guidance that availability of other plant air cleanup systems will marginally reduce impact of SBEACS unavailability.

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)

Two ICS trains shall be Operable in Modes 1, 2, 3, and 4.

Licensing Basis for LCO

For several PWRs, the ICS contributes to meeting 10 CFR 100 (Reference 9) siting requirement dose limits and supports compliance with 10 CFR 50 Appendix A, GDC 19 (Reference 10) for operator dose. The ICS also supports meeting GDC 41, "Containment Atmosphere Cleanup". These design basis calculations assume a high concentration of elemental iodine in the fission product release (See References 11 and 12). 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 Required Action to restore at least one ICS train within 24 hours when both ICS trains are inoperable.

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. It is also assumed that the released radioactive iodine will be reduced by the containment sprays.

There is no significant risk impact of extending the 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. 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.

Changes in the unavailability of the ICS have no direct impact on ICCDP, ICLERP and will not change the plant CDF or LERF. Since extending the AOT on these systems has no impact on the aforementioned risk metrics, the impact of the proposed modification to the TS for this system is below the RG 1.174 incremental risk guidelines and derivative RG 1.177 TS guidance.

The frequency of "less than LERF" radiological releases is estimated to be between 1E-04 and 0.01 events per year, with the greater event frequency being associated with lesser releases (See Table 4.3-1). Availability of at least one train of CSS, as discussed below, provides significant means of limiting radiological releases to the public from these lesser releases as well.

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, a Required Action is added to the TS which requires verification within 1 hour that at least one train of Containment Spray is Operable.

Tier 2 Recommendations

None.

Tier 3 Recommendations

It should be noted in the in appropriate administrative guidance that availability of other plant air cleanup systems will marginally reduce impact of system unavailability.

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, for some plants, during 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 for reasons other than an inoperable barrier in Modes 1, 2, 3 or 4.

Proposed Modification to Actions

It is recommended that a 24 hour period be provided to restore one train of CREACS for conditions other than boundary inoperabilities.

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.

Basis for Proposed Change

Changes in the unavailability of the CREACS has no direct impact on ICCDP, ICLERP and will not change the plant CDF or LERF. Since extending the AOT on these systems has no impact on the aforementioned risk metrics, the impact of the proposed modification to the TS for this system is below the RG 1.174 incremental risk guidelines and derivative RG 1.177 TS guidance.

The probability that an event would occur challenging this system during the extended outage period is very low. In the event an accident would occur in this time frame, defense in depth measures are described below to ensure control room personnel are protected.

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 less than the licensing basis limits (typically 5 REM or the equivalent to any part of the body). To maximize protection to the operators under the proposed AOT, mitigating actions will be taken to lessen the effect on the CRE occupants from potential hazards. 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. Such actions could include restricting transport of hazardous materials and verifying availability of personal radiological protection systems).

As SGTR events are a significant contributor to potential CRE radiological hazards due to the release of radioactive RCS fluid. The Technical Specifications include a Required Action to verify that LCO 3.4.16 "RCS Specific Activity" is met.

Tier 2 Recommendations

None.

Tier 3 Recommendations

Plant administrative controls should consider the status of the following technical specifications in assessing and managing plant risk while operating: TS 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.

It should be noted in the appropriate administrative guidance that operability of other plant air cleanup systems, the CSS and containment air cooling units (if available) will reduce impact of system unavailability.

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

Eliminate Condition E and add a new Condition B which allows 24 hours to restore at least one CREATCS train to Operable status before requiring a plant shutdown.

Basis for Proposed Change

As a result of the low (approx. 10^{-4} per year) frequency of challenges to this system, long room heatup times, availability of alternate cooling strategies and alternate means to control emergency systems locally, changes in the short term unavailability of the CREATCS have negligible impact on ICCDP, ICLERP , ΔCDF or ΔLERF. Consequently, the impact of the proposed modification to the TS for this system is below the RG 1.174 incremental risk guidelines and derivative RG 1.177 TS guideline. Furthermore, some plants have analyses that demonstrate that post-accident control room heatup levels will not challenge control room personnel or equipment for periods up to 24 hours.

Defense-in-Depth Consideration

In the event the CR would need to be evacuated, emergency equipment may be controlled locally.

Tier 2 Recommendations

None.

Tier 3 Recommendations

Administrative procedures should direct the staff to monitor CR temperature to ensure the CR remains habitable and that electrical cabinets are not exposed to excessive temperatures. Actions can be taken to control the temperature in the CR 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.

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

The ECCS PREACS 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 Auxiliary Building. The primary function of the ECCS PREACS is to filter volatized iodine resulting from leakage of post-accident recirculation fluid from the ECCS pumps. The ECCS PREACS consists of two independent, redundant trains of equipment that provide filtering of air in the ECCS pump rooms during post-LOCA recirculation cooling.

Plant Applicability

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

Limiting Condition for Operation (LCO)

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

Licensing Basis for LCO

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 10 CFR 100 and 10 CFR 50, 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 (the radionuclide distribution is dependent upon methodology selected). All released radionuclides are assumed instantaneously swept into the sump, recirculated, and leaked at an assumed rate, usually 1 gpm.

Condition Requiring Entry into a Shutdown Action Statement

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

Proposed Modification to Actions

Allow 24 hours to restore one train of ECCS PREACS to Operable status.

Basis for Proposed Change

While the ECCS Pump Room Emergency Air Clean-up System (PREACS) affects the magnitude of post-accident radioiodine releases, operation of ECCS PREACS has no direct impact on ICCDP and ICLERP. 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.

Changes in the unavailability of the ECCS PREACS have no direct impact on ICCDP, ICLERP and will not change the plant CDF or LERF. Since extending the AOT on these systems has no impact on the aforementioned risk metrics, the impact of the proposed modification to the TS for this system is below the RG 1.174 incremental risk guidelines and derivative RG 1.177 TS guidelines.

A 24 hour AOT for this inoperability is also reasonable based on the low probability of a DBA. The frequency of these smaller radiological releases is estimated to between 1E-04 and 0.01 events per year, with the greater event frequency being associated with lesser releases (See Table 4.3-1).

Defense-in-Depth Consideration

On-site impact of any higher than expected radioactive releases due to the unavailability of the ECCS PREACS can be mitigated by ensuring that the Control Room Emergency Air Cleanup System (CREACS) is Operable. The Technical Specifications will contain a Required Action to verify that at least one train of CREACS is Operable when using the proposed AOT.

Tier 2 Recommendations

None.

Tier 3 Recommendations

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 Phosphate" 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 at least one train of PREACS to Operable.

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.

Changes in the unavailability of the PREACS have no direct impact on ICCDP or ICLERP, and will not change the plant CDF or LERF. Since extending the AOT on these systems has no impact on the aforementioned risk metrics, the impact of the proposed modification to the TS for this system is below the RG 1.174 incremental risk guidelines and derivative RG 1.177 TS guidelines. Consequently, 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. Unavailability of the PREACS will not have a risk significant impact on radiological releases to the public or to the control room.

The 24 hour AOT extension is also supported by the low probability of a DBA during the period of inoperability. The frequency of these smaller radiological releases is estimated to be between 1E-04 and 0.01 events per year, with the greater event frequency being associated with lesser releases (See Table 4.3-1).

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 Technical Specifications will require verification that at least one train of Containment Spray is Operable. The Containment Spray System will mitigate leakage of airborne radionuclides from the containment.

Tier 2 Recommendations

None.

Tier 3 Recommendations

It should be noted in the in appropriate administrative guidance that availability of other plant air cleanup systems will marginally reduce impact of system unavailability. Administrative CRMP guidance should include a recommendation to maintain at least one train of CREACS available upon entry into this condition.

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 LCO 3.6.6.A Required Actions for restoring inoperable containment spray systems are less stringent than the requirements in LCO 3.6.6.B. 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, provide an 8 hour AOT to restore at least one CS train or CAC train when all CS trains and all CAC trains are inoperable. 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 or one CAC operates. Consequently, a minimum containment heat removal capability is required to ensure both long-term containment integrity and core damage prevention. Containment Spray and CAC are also considered in the 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 8 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, one train of CAC or one train of CS assures adequate heat removal capability.

For the case where both CACs and CSS are unavailable, Table 4.1-2 indicates a ICCDP impact of 5E-7 which results in a ICLERP of 8.4E-09 for the proposed 8 hour AOT duration. These are considered bounding estimates as the risk assessment did not consider the potential for restoration of either system. This is particularly significant since for this TS change, the time available for repair is likely to exceed the proposed AOT.

Increases in plant Core Damage Frequency and Large Early Release Frequency based on a one in three year entry into this TS condition is estimated to be less than 1.67E-07 per year and 2.81E-09 per year, respectively. As a result of the low challenge frequency and infrequent expected use, 8 hours provides a reasonable AOT. By allowing a short duration repair at power the plant can more efficiently apply resources thus reducing the overall component unavailability and avoiding any additional risks that may arise as a result of plant shutdown.

To minimize the potential for a containment challenges and ensure defense in depth this later condition (both trains of CS and CARCs inoperable) should not be entered with concurrent inoperabilities of the AFW or ECCS system..

A comparison of risk metrics for this condition is presented in Tables 4.2.6 and 4.2.7.

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, and redundancy afforded by the availability of the CARC. 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 and 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.

To minimize the potential for a containment challenges and ensure defense in depth this later condition (both trains of CS and CARCs inoperable) should not be entered with concurrent inoperabilities of the AFW or ECCS system..

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 by using the inventory available in 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 Recommendations

None.

Tier 3 Recommendations

Plants implementating this TS change should have proceduralized actions, equipment and inventory for refilling the RWST. These actions already exist under the Severe Accident Management Guidance.

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	8 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 is recommended to be 24 hours. In specific instances, recommendations for longer and shorter AOTs are made, as appropriate.

The proposed TS changes justified 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

1. RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant Specific Changes to the Licensing Basis," USNRC, July 1998.
2. RG 1.177, "An Approach for Plant Specific Risk-Informed Decision Making: Technical Specifications," USNRC, August 1998.
3. NUREG-1432, "Standard Technical Specifications, Combustion Engineering Plants," Revision 3.1, USNRC.
4. CE NPSD-1186, "Technical Justification for the Risk-Informed Modification to Selected Required Action End States for CEOG PWRs," CE Owner's Group, April 2000.
5. NUREG-1570, "Risk Assessment of Severe Accident Induced Steam Generator Tube Rupture," USNRC, March 1998.
6. NUREG/CR-6338, "Resolution of Direct Containment Heating Issue for all Westinghouse Plants with Large Dry Containments or Subatmospheric Containments," Pilch, M.M., et. al., Sandia National Laboratories, January 1996.
7. NUREG/CR-5750, "Rates of Initiating Events at U.S. Nuclear Power Plants: 1987 - 1985," USNRC, February 1999.
8. CE NPSD-1045-A, "Joint Application Report: Modifications to Containment Spray System, and the Low Pressure Safety Injection System Technical Specifications," CE Owner's Group, March 1998.
9. 10 CFR 100, "Reactor Site Criteria," USNRC, 1991.
10. 10 CFR 50, "Appendix A: General Design Criteria 79," USNRC, 1991.
11. TID 14844, "Calculation of Distance Factors for Power Reactor Sites," USAEC, 1962.
12. Regulatory Guide 1.4, Revision 2, "Assumptions used for Evaluating the Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors," USNRC, June 1974.
13. NUREG-1465, "Accident Source Terms for Light Water Reactors," February 1995.
14. Not used.
15. Not used.
16. Not used.
17. Not used.
18. CE NPSD-1168-A, "Joint Application Report: Modifications to CIV Technical Specification," Westinghouse Electric Company, September 2000.
19. NUREG/CR-5500, "Reliability Study: Auxiliary/Emergency Feedwater System, 1987-1995," USNRC, August 1998.
20. Arkansas Nuclear One – Unit 2 Letter, 2CAN030003, "Proposed License Change For Cycle 14 Risk-Informed Operation," from Craig Anderson (ANO-2) to U.S. Nuclear Regulatory Commission, dated March 9, 2000.

21. Westinghouse Letter, LTR-ESI-02-39, "Response to NRC RAIs Concerning Topical Report CE NPSD-1208, Justification of Risk-Informed Modifications to Selected Technical Specifications for Conditions Leading to Exigent Plant Shutdown" (TSTF-360), dated February 20, 2002.
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.

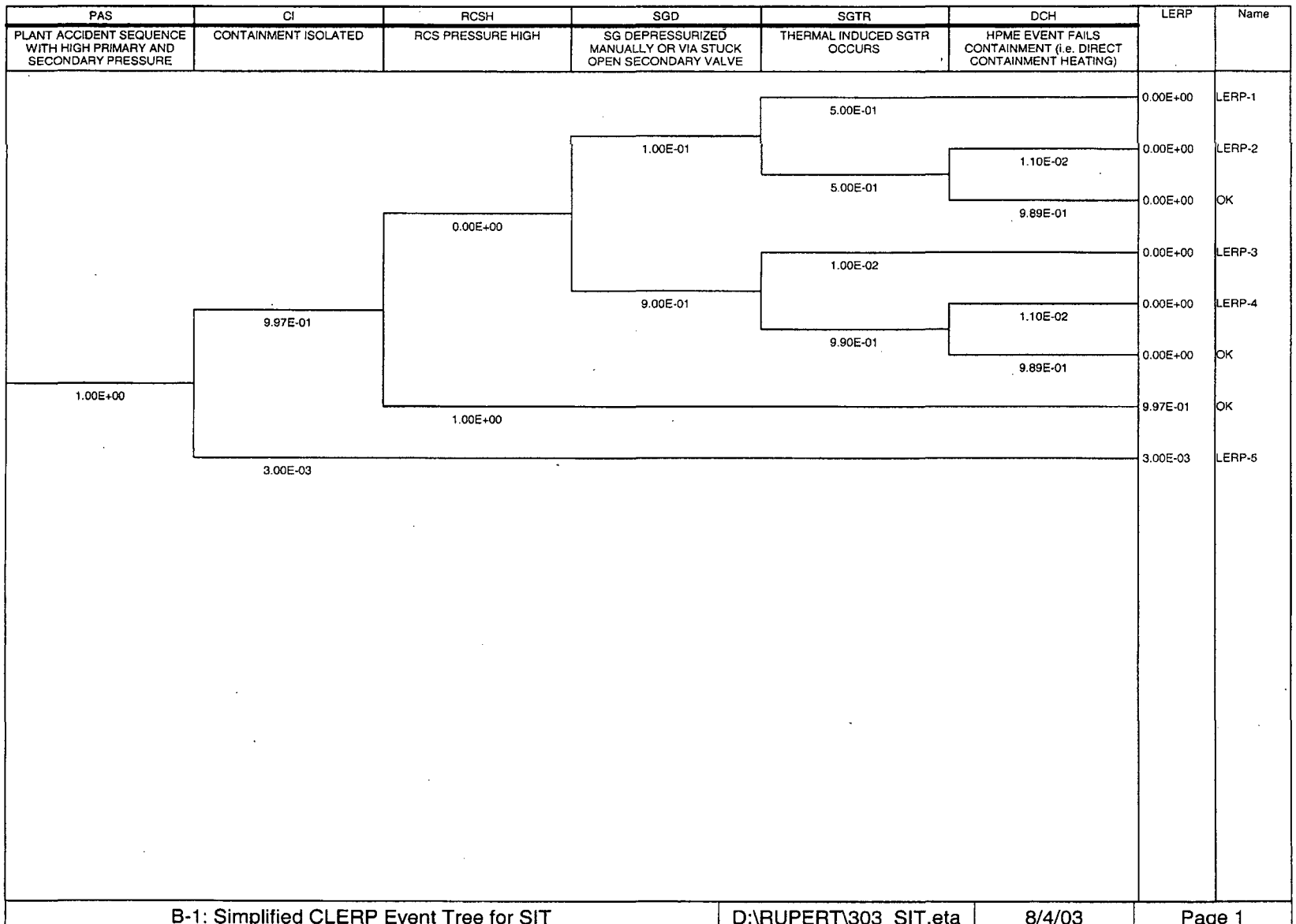
APPENDIX A

Technical Specification Cross-Reference

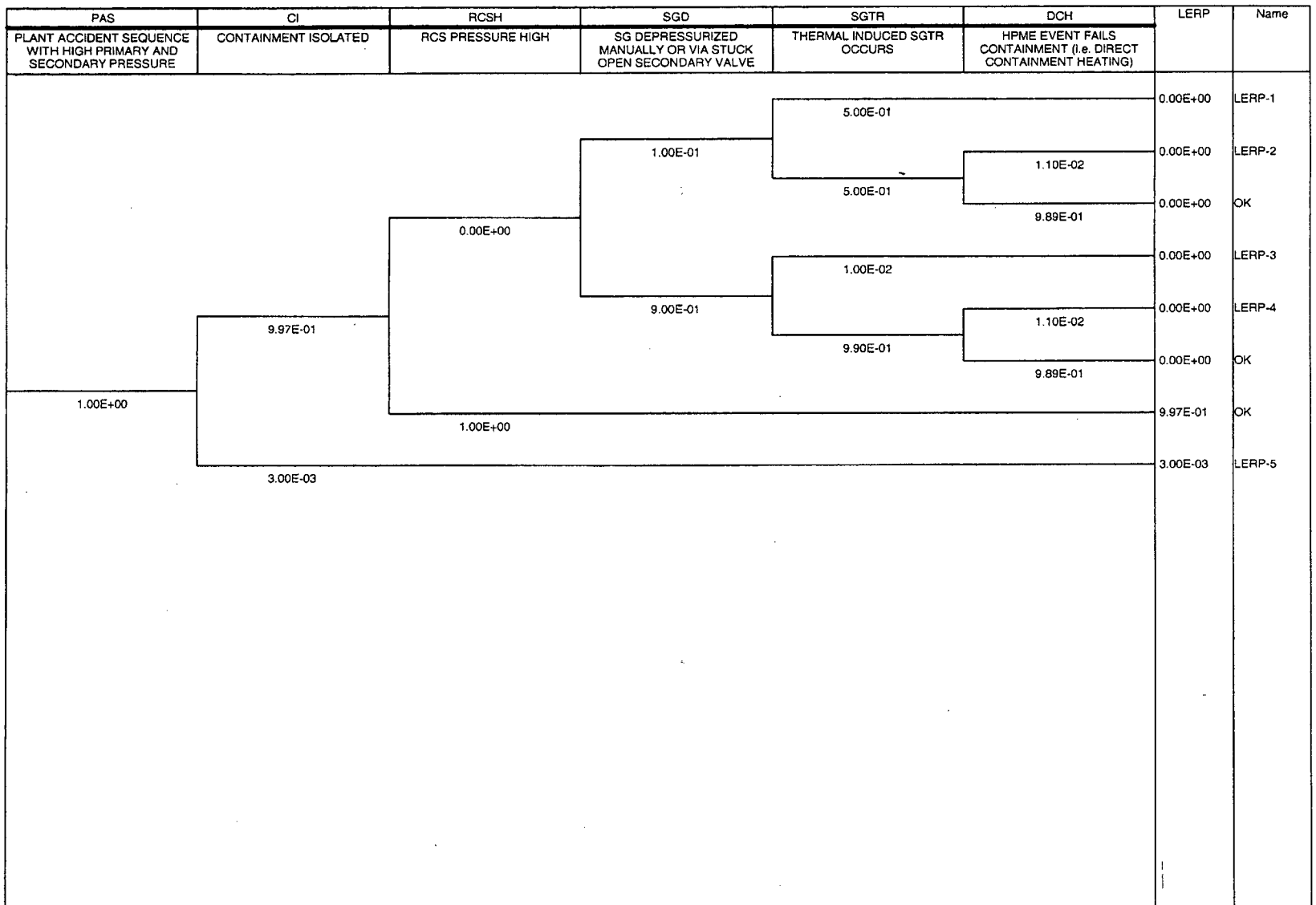
This Appendix contained archival information that has been deleted.

APPENDIX B

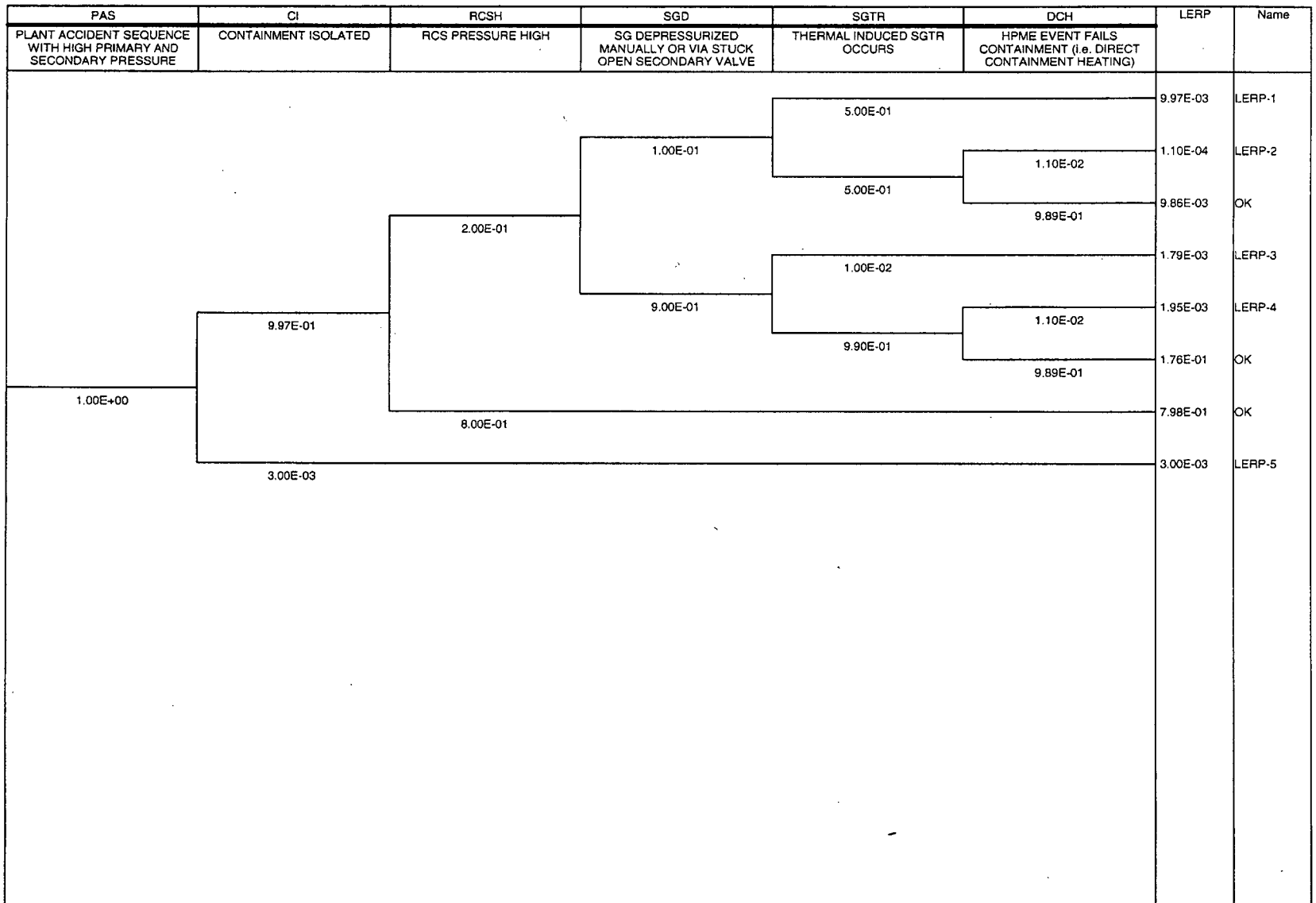
System Specific LER Event Trees



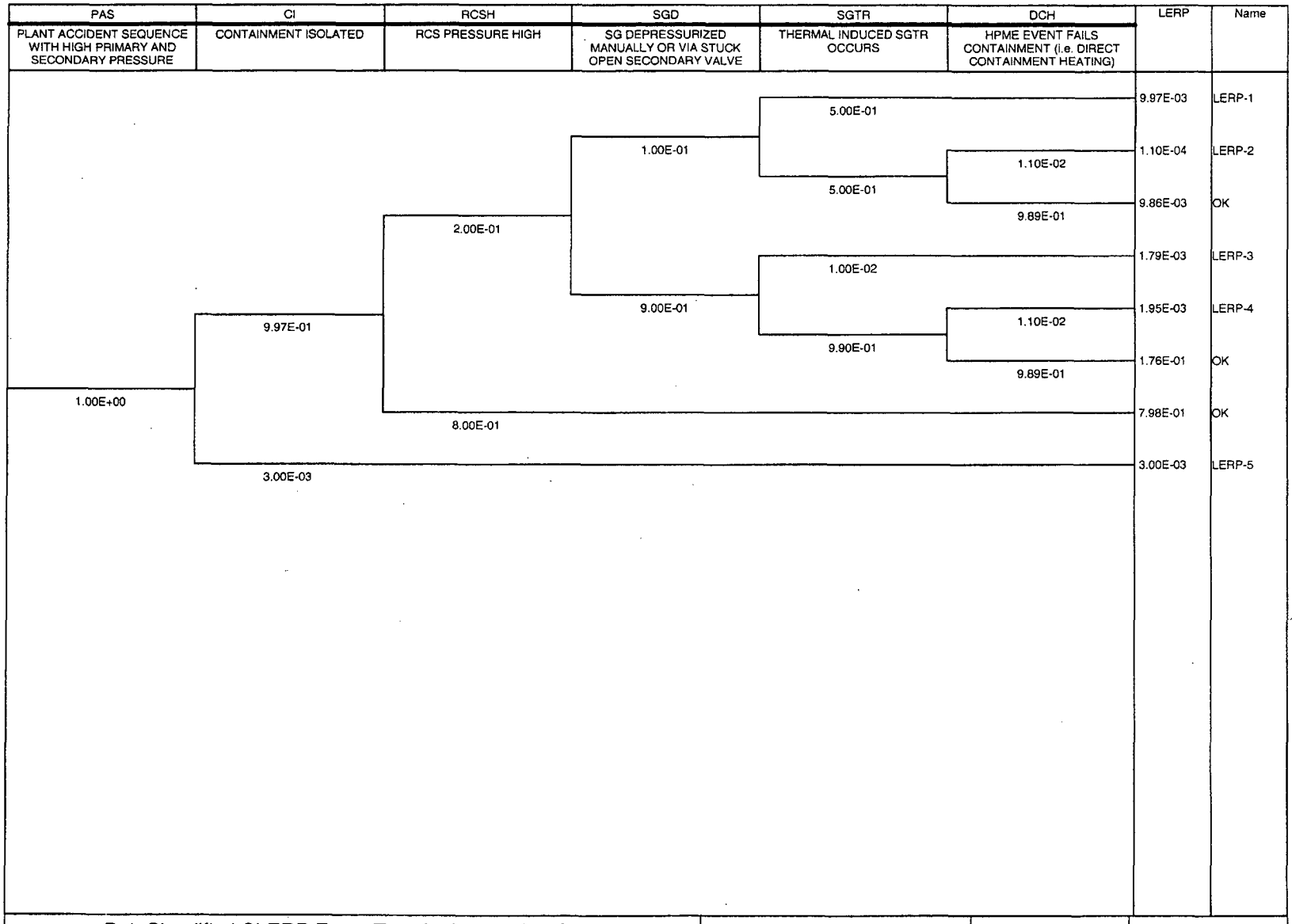
B-1: Simplified CLERP Event Tree for SIT | D:\RUPERT\303_SIT.eta | 8/4/03 | Page 1



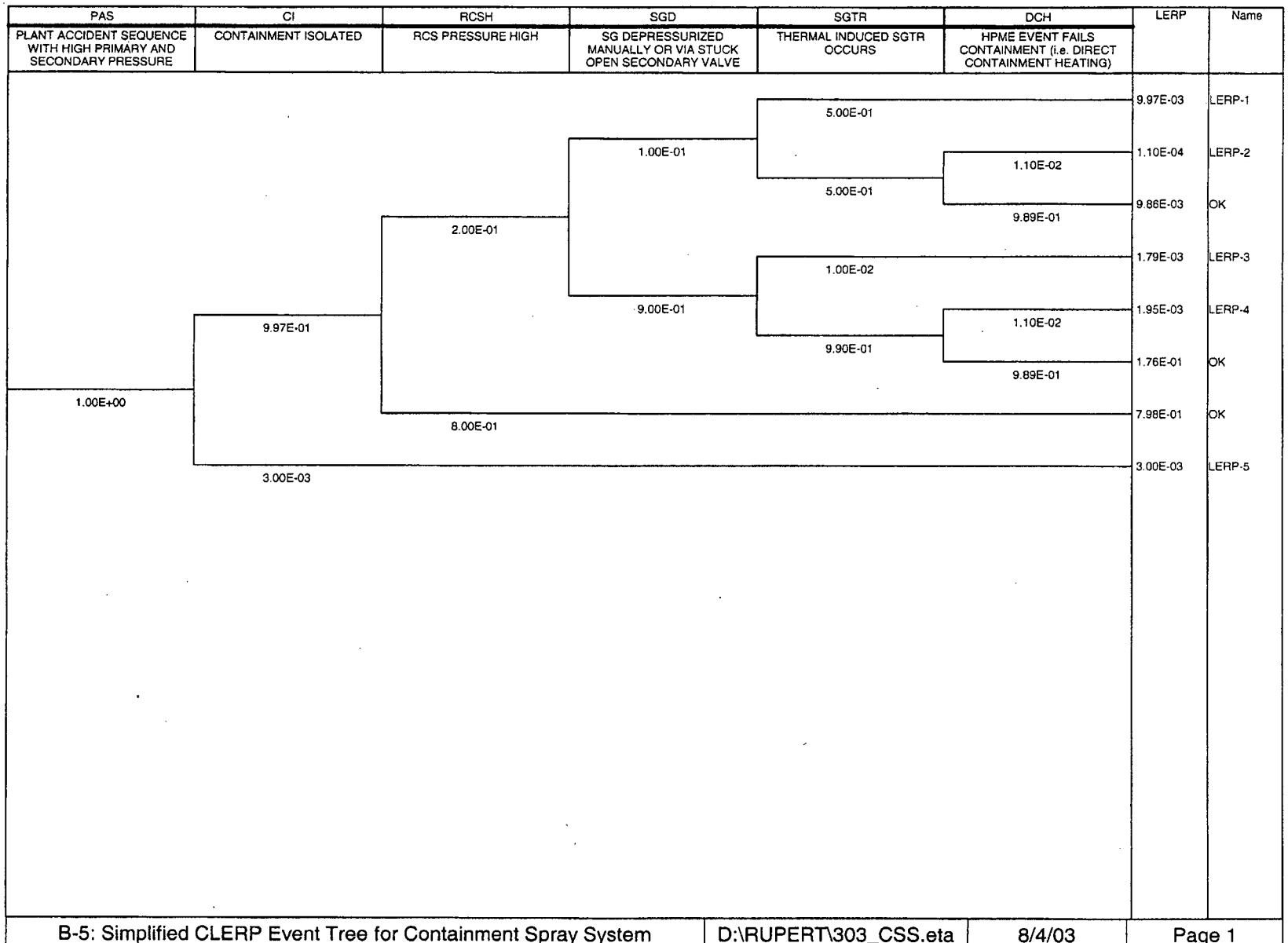
B-2: Simplified CLERP Event Tree for LPSI	D:\RUPERT\303_LPSI.eta	8/4/03	Page 1



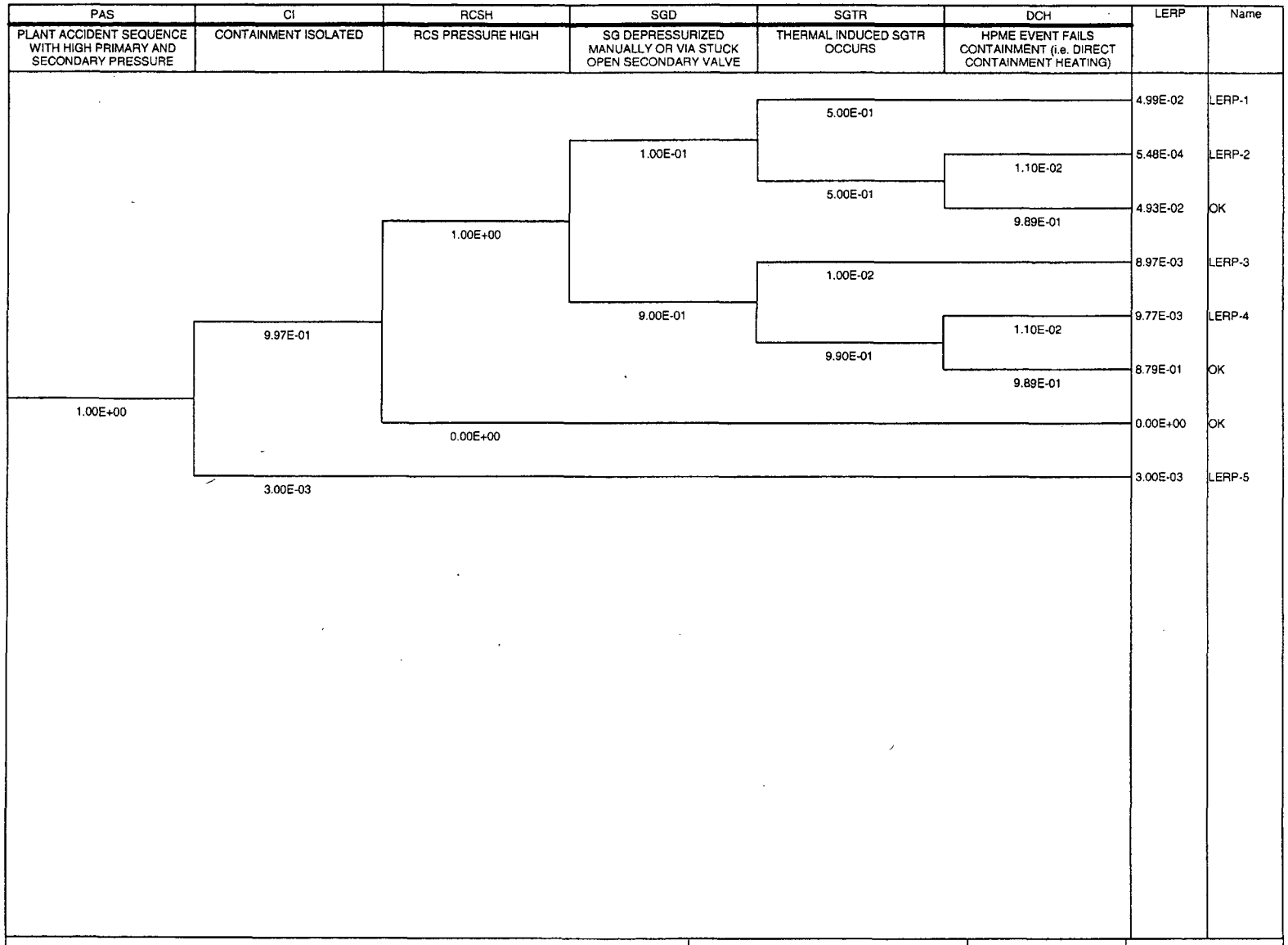
B-3: Simplified CLERP Event Tree for HPSI with PORV	D:\RUPERT\303_HPSP.eta	8/4/03	Page 1



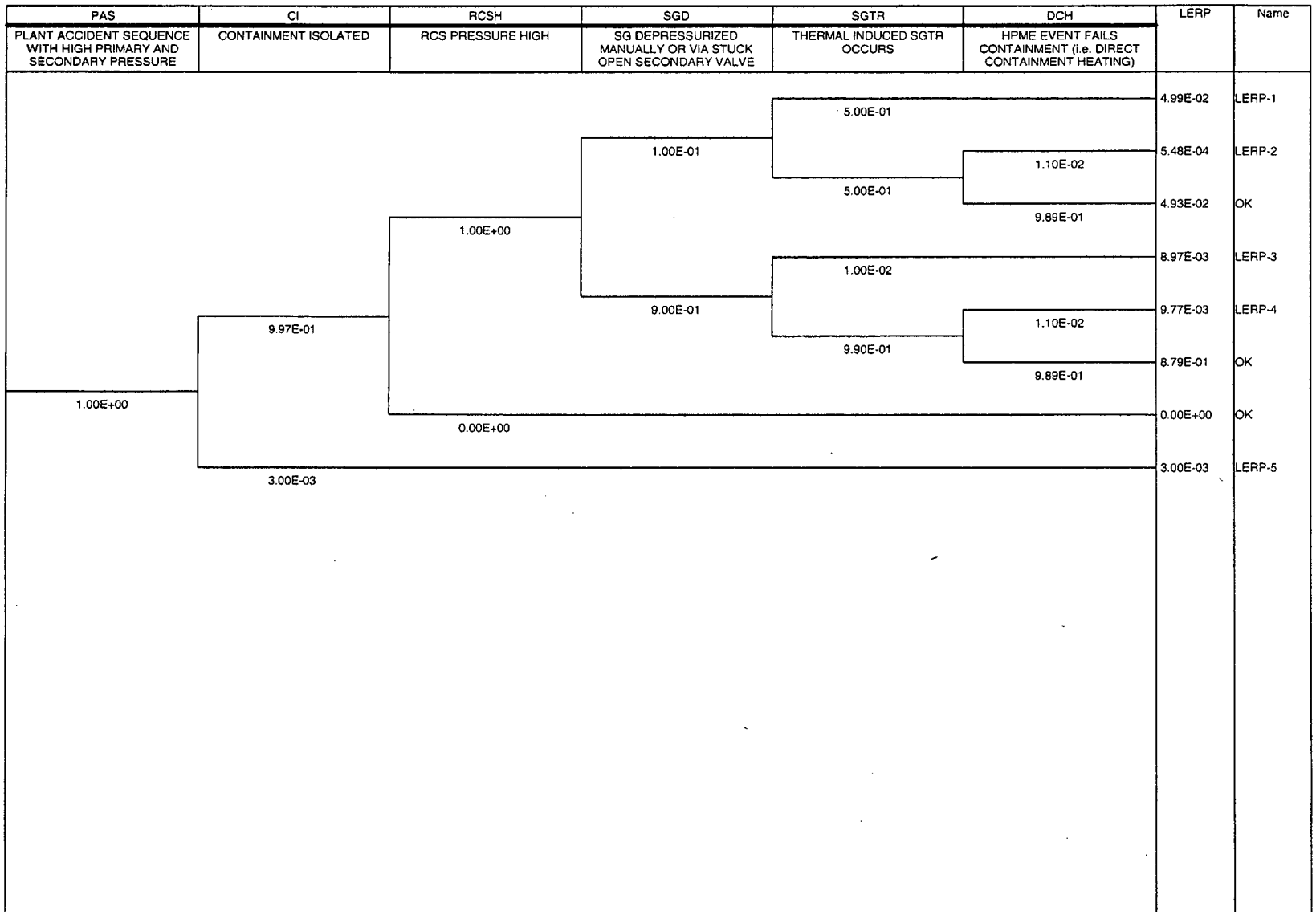
B-4: Simplified CLERP Event Tree for HPSI w/o PORV	D:\RUPERT\303_HPSI.eta	8/4/03	Page 1



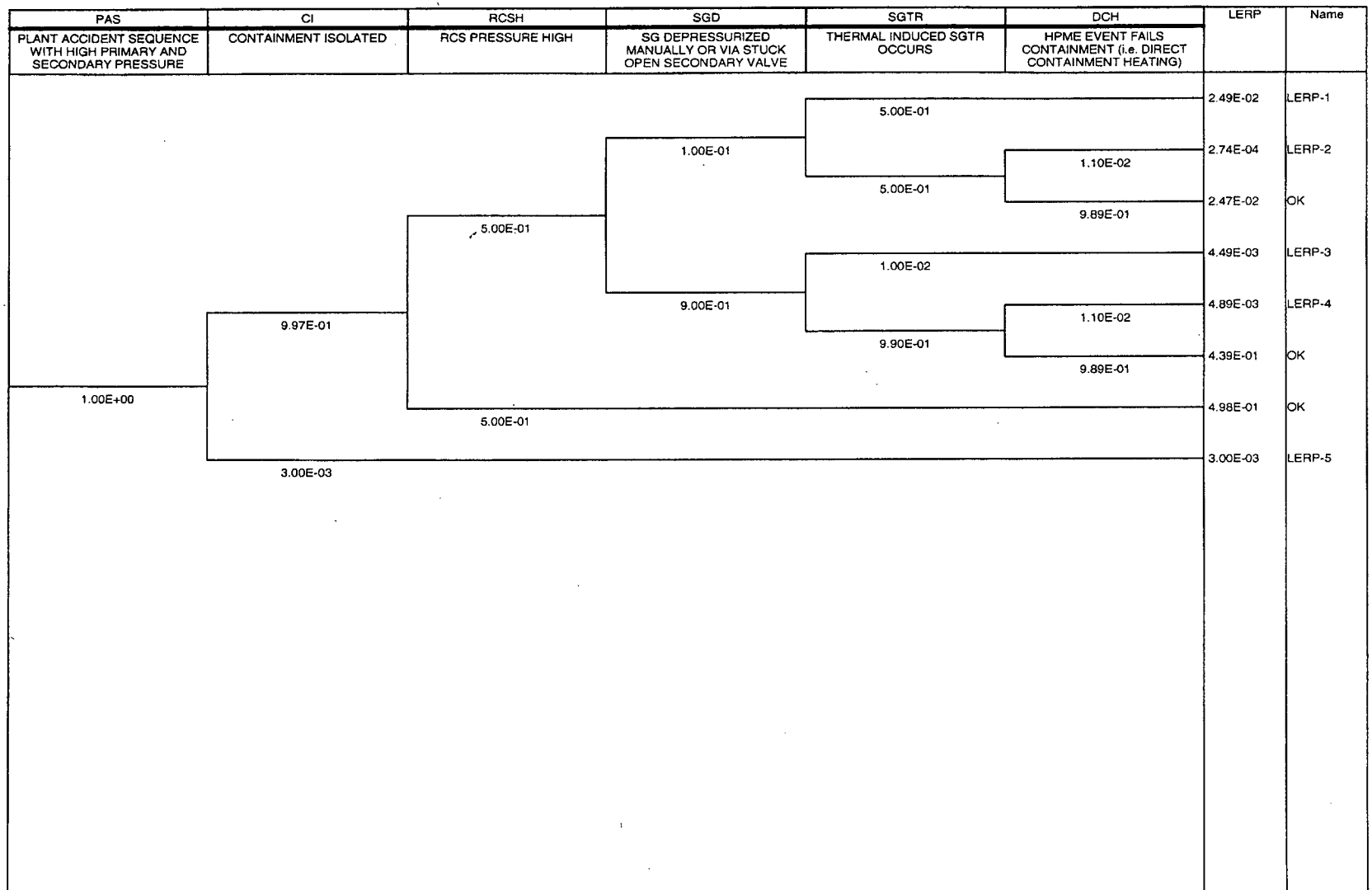
B-5: Simplified CLERP Event Tree for Containment Spray System | D:\RUPERT\303_CSS.eta | 8/4/03 | Page 1



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

Historical Review Information

PWROG
PWR Owners Group

Program Management Office
4350 Northern Pike
Monroeville, Pennsylvania 15146

Project No. 694

WCAP-16125-NP-A

February 16, 2007

OG-07-69

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

Subject: **PWR Owners Group Withdrawal of Report WCAP-16125-NP-A, Rev. 0 – "Justification for Risk-Informed Modifications to Selected Technical Specifications for Conditions Leading to Exigent Plant Shutdown" (Original TAC MB1257) – PA-LSC-0364**

The Pressurized Water Reactor Owners Group (PWROG) hereby withdraws WCAP-16125-NP-A, Rev 00," dated August 2004.

A conference call was held on February 7, 2007 to discuss emergent concerns identified at the NEI /NRC RITSTF meeting of January 18, 2007 associated with inconsistencies between the NRC Safety Evaluation (SE), topical report, the associated TSTF and implementation guidance. Withdrawal of the topical will allow the PWROG to add clarifications to the topical and improve its value as the industry pilot for RITSTF Initiative 6 b and c, "Revise or Add Actions to Preclude Entry into LCO 3.0.3." In addition, the withdrawal will allow the NRC to revise the SE to improve implementation and general exportability. At the conclusion of the call, the PWROG agreed to:

1) Withdraw WCAP-16125-NP-A,
2) Work with NRC to revise the topical report consistent with a revised version of the SE, and
3) Resubmit the revised WCAP to the NRC for review and approval. The PWROG expects to resubmit on or before April 30, 2007.

The NRC noted that the proposed changes approved in the current SE will likely be found to be applicable in the revised SE and that the needed changes to the WCAP are expected to be limited, but important. The NRC staff further agreed to provide an expedited review of the revised topical (*Approximately 6 months from date of issuance*) and to support the PWROG request for a continuation of the fee waiver associated with the existing TAC. It was agreed that a technical issues resolution meeting with the Staff will be held expeditiously. The PWROG further requests that a pre-submittal meeting be held four weeks prior to submittal. The purpose of the later meeting is to review the changes that have been made to the WCAP to ensure that all known issues are satisfactorily addressed.

Correspondence related to this transmittal should be addressed to:

Mr. Gordon Bischoff, Program Manager
Owners Group Program Management Office
Westinghouse Electric Company
P.O. Box 355
Pittsburgh, Pennsylvania 15230-0355

If you have any questions, please do not hesitate to contact me at (630) 287-8780 or Mr. Tom Laubham of the PWR Owners Group, Program Management Office at (412) 374-6788.

Sincerely,

Thomas J. Laubham for,

Frederick P. "Ted" Schiffley II, Chairman
PWR Owners Group

cc: Sean Peters, NRC
PWROG Management Committee
Licensing Subcommittee (CE Members Only)
Risk Management Subcommittee
Program Management Office
R. Schneider, W
J. Andrachek, W
J. Andre, W
C. Morgan, W
G. Ament, W
J. Gresham, W

Program Management Office
4350 Northern Pike
Monroeville, Pennsylvania 15146

April 11, 2007

OG-07-144

To: Licensing Subcommittee – CE Members

Subject Pressurized Water Reactor Owners Group
NRC Withdrawal of the Safety Evaluation for WCAP-16125-NP-A, LSC-0364

On February 16, 2007 (OG-07-69), the PWR Owners Group withdrew WCAP-16125 NP-A, Rev. 0 "Justification for Risk-Informed Modifications to Selected Technical Specifications for Conditions Leading to Exigent Plant Shutdown" (Original TAC MB1257). This withdrawal was at the request of the NRC and was done so in order to resolve NRC concerns with the approved topical report. Enclosed for your information is the NRC letter acknowledging the PWR Owners Group withdrawal of WCAP-16125-NP-A and the withdrawal of the NRC Safety Evaluation.

If you have questions, please feel free to contact me at (412) 374-6788.

Regards,

Electronically Approved

Thomas J. Laubham
Project Manager
PWR Owners Group
Program Management Office

TJL:ml

Enclosure

cc: PMO
Ray Schneider, W
Jerry Andre, W
Jim Gresham, W

Management Committee
Gary Ament, W
J. Andrachek, W
Bob Schomaker, AREVA-NP

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

April 3, 2007

RECEIVED
APR 09 2007
OG PROJECT OFFICE

Mr. Gordon Bischoff, Manager
Owners Group Program Management Office
Westinghouse Electric Company
P.O. Box 355
Pittsburgh, PA 15230-0355

SUBJECT: WITHDRAWAL OF PRESSURIZED WATER REACTOR OWNERS GROUP (PWROG) TOPICAL REPORT (TR) WCAP-16125-NP-A, REV. 0, "JUSTIFICATION FOR RISK-INFORMED MODIFICATIONS TO SELECTED TECHNICAL SPECIFICATIONS FOR CONDITIONS LEADING TO EXIGENT PLANT SHUTDOWN" (TAC NO. MD4936)

Dear Mr. Bischoff:

By letter dated February 16, 2007 (Agencywide Documents Access Management System (ADAMS) Accession No. ML070520189), the PWROG withdrew TR WCAP-16125-NP-A, Rev. 0, "Justification for Risk-Informed Modifications to Selected Technical Specifications for Conditions Leading to Exigent Plant Shutdown." Since the approved TR has been withdrawn, the U.S. Nuclear Regulatory Commission staff's safety evaluation (SE) dated July 9, 2004 (ADAMS Accession No. ML041940288), is no longer required to support the approved TR, and therefore, is hereby withdrawn.

If you have questions regarding this matter, please contact Sean E. Peters at (301) 415-1842.

Sincerely,

Stacey L. Rosenberg, Chief
Special Projects Branch
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

Project No. 694

Enclosure: Withdrawn SE

cc: Mr. James A. Gresham, Manager
Regulatory Compliance and Plant Licensing
Westinghouse Electric Company
P.O. Box 355
Pittsburgh, PA 15230-0355

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

SAFETY EVALUATION OF WCAP-16125-NP, REVISION 0 (CE NPSD-1208, REVISION 1)

"JUSTIFICATION FOR RISK-INFORMED MODIFICATIONS TO SELECTED TECHNICAL SPECIFICATIONS FOR CONDITIONS LEADING TO EXIGENT PLANT SHUTDOWN"

1.0 INTRODUCTION

By letter dated January 24, 2001 (Reference 1), the Combustion Engineering Owners Group submitted Topical Report CE NPSD-1208, Rev. 0, "Justification for Risk-Informed Modifications to Selected Technical Specifications for Conditions Leading to Exigent Plant Shutdown" (Reference 2), for review by the NRC staff. By letter dated October 3, 2003 (Reference 3), the Westinghouse Owners Group (WOG) provided WCAP-16125, Revision 0, which updated and superceded CD NPSD-1208 in its entirety. Also, WCAP-16125 incorporated responses to NRC staff request for additional information on CE NPSD-1208. WCAP-16125 provides the technical justification for proposed risk-informed modifications to Technical Specifications (TSs) such that unnecessary exigent plant shutdowns resulting from entry into Limiting Condition for Operation (LCO) 3.0.3 (or equivalent ACTION STATEMENTS) may be avoided. The required action for conditions that imply a loss of function, related to a system or component included within the scope of the plant TS, is entry into LCO 3.0.3. Currently, upon entering LCO 3.0.3, one hour is allowed to prepare for an orderly shutdown before initiating a change in plant operation. This includes time to permit the operator to coordinate the reduction in electrical generation with the load dispatcher to ensure the stability and availability of the electrical grid. The WOG is proposing to define and/or modify various TS Conditions to accommodate extension of the currently required time of one hour to initiate plant shutdown for members of the WOG with Combustion Engineering (CE) Nuclear Steam Supply Systems (NSSS) designs. The proposed extension, related to specific systems or components, is based on the system's risk significance and varies from 4 hours to 72 hours.

WITHDRAWN

The proposed changes are typically associated with plant conditions where both trains of a two-train redundant system are declared inoperable and at the same time there is either no specified action in the TS for the condition (requiring a default LCO 3.0.3 entry) or conditions exist where the defined action includes an explicit LCO 3.0.3 entry. The intent of the proposed TS changes is to provide a risk-informed alternative to the current LCO 3.0.3 requirements such that the plant staff has adequate time to fully evaluate the situation or restore loss of function while the plant remains operating at power, thus avoiding unnecessary unscheduled plant shutdowns and minimizing transition and realignment risks.

WCAP-16125-NP also provides system-specific integrated justifications (i.e., risk and defense-in-depth arguments) for several proposed TS Required Action statement changes to allow a MODE 4 (hot shutdown) end state, for repair purposes of two-train redundant systems that do not have explicit LCO 3.0.3 entry requirements, when the proposed extended time cannot be met. Justifications for the proposed end state changes are based on generic risk assessments documented in topical report CE-NPSD-1186 (Reference 4), entitled "Technical Justification for the Risk-Informed Modification to Selected Required Action End States for CEOG Member PWRs" which has been reviewed by the staff.

2.0 BACKGROUND

The intent of the proposed TS changes is to provide needed flexibility in the performance of corrective maintenance during power operation and at the same time enhance overall plant safety by:

- avoiding unnecessary unscheduled plant shutdowns,
- minimizing plant transitions and associated transition and realignment risks,
- providing increased flexibility in scheduling and performing maintenance and surveillance activities, and
- providing explicit guidance in areas that currently does not exist.

It should be noted that many of the proposed TS changes affect the existing plant shutdown requirements for plant conditions where the plant operation is not in explicit compliance with the plant design basis. The proposed actions provide a risk-informed process for establishing shutdown priorities aiming at reducing overall plant risk and increasing public health and safety protection.

WITHDRAWN

The proposed TS changes, including end state changes, are summarized in Table 1 of this safety evaluation report (SER). Such changes cover a diverse range of systems and components with essentially four separate impacts on plant risk. They are:

- TS changes related to systems or components contributing to accident prevention. The removal of these systems/components has the potential to increase the plant risk through the increased potential for plant upsets (i.e., potential for increased initiated event frequencies). A typical example in this category are the pressurizer heaters whose unavailability could complicate the complexity of plant pressure control and lead to a plant trip.
- TS changes related to systems or components contributing to accident mitigation. These systems are in standby during normal plant operation and are intended to function during accidents to prevent core damage. Typical examples in this category are the Emergency Core Cooling System (ECCS) and the pressurizer Power Operated Relief Valves (PORVs).
- TS changes related to systems or components contributing to large early release prevention. The primary role of these systems is to function during a core damage accident to prevent large releases of radioactive materials. A typical example in this category is the containment (the only component in this category for which a TS change is proposed).
- TS changes related to systems/components contributing to control of delayed radiation releases to the environment. The primary role of these systems is to prevent radiation

releases above TS limits and meet design basis requirements. Thus, the unavailability of these systems has no impact on the surrogate risk metrics associated with core damage and large early releases. Typical examples in this category are the ECCS room ventilation system and the containment iodine cleanup system.

Although the improved standard technical specification (ISTS) numbering system (NUREG-1432, Reference 5) is used for convenience in Table 1, the analyses provided in WCAP-16125-NP support these changes for all CE designed NSSS plants.

Table 1 Summary of Proposed Modifications to Technical Specifications.

ISTS #	SYSTEM	INOPERABILITY CONDITION	CURRENT ACTION AND ASSOCIATED COMPLETION TIME (CT)	PROPOSED CHANGES: COMPLETION TIME(CT) AND END STATE
LCO 3.1.9 (Non-ISTS)	Boration System - Operating	Both boration paths inoperable	No condition defined. Default LCO 3.0.3 entry.	24 hrs CT for restoring one path. Allow MODE 3 end state when boric acid source tank volume, temperature, or concentration are out of limits.
LCO 3.4.9	Pressurizer Heaters	Both groups of class 1E heaters inoperable	No condition defined. Default LCO 3.0.3 entry.	24 hrs CT for restoring one group.
LCO 3.4.11	Pressurizer Power Operated Relief Valves (PORVs) and Associated Block Valves (BVs)	ISTS LCO 3.4.11 CONDITION E (or equivalent): Two PORVs inoperable and not capable of being manually cycled. ISTS LCO 3.4.11 CONDITION F (or equivalent): Two BVs inoperable.	Varies with plant. ISTS LCO 3.4.11 CONDITION E (or equivalent): Close associated block valve in 1 hour AND remove power from associated block valve in one hour, AND be in MODE 3 in 6 hours AND MODE 4 in [12] hours. ISTS LCO 3.4.11 CONDITION F (or equivalent): Restore one block valve to operable in 2 hours. ISTS Condition G requires MODE 3 in 6 hours and MODE 4 in [12] hours if Condition F not met.	ISTS LCO 3.4.11 CONDITION E (or equivalent): Allow 8 hours CT to restore one PORV, for conditions where a PORV is unable to reclose once challenged but may be isolated. ISTS LCO 3.4.11 CONDITION F (or equivalent): Allow 8 hours to restore one BV.
LCO 3.5.1	Safety Injection Tanks (SITs)	Two or more SITs inoperable (ISTS CONDITION D)	Explicit 3.0.3 entry	Revise ISTS Condition D to allow 24 hours CT for restoring one SIT.

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ISTS #	SYSTEM	INOPERABILITY CONDITION	CURRENT ACTION AND ASSOCIATED COMPLETION TIME (CT)	PROPOSED CHANGES: COMPLETION TIME(CT) AND END STATE
LCO 3.5.2	Low Pressure Safety Injection (LPSI)	Two LPSI subsystems inoperable	Default 3.0.3 entry	24 hours CT for restoring one LPSI subsystem (ISTS Condition D would be deleted).
LCO 3.5.2	High Pressure Safety Injection (HPSI)	Two HPSI subsystems inoperable (ISTS Condition D)	Explicit 3.0.3 entry	4 hours CT for restoring one HPSI subsystem.
LCO 3.6.6A	Containment Spray System (CS)	Two CS trains inoperable OR any combination of three or more trains inoperable (i.e., containment air coolers (CAC*)) (ISTS Condition F)	Explicit 3.0.3 entry	12 hrs CT for restoring one CS train if CAC is not available. 72 hours CT for restoring one CS if one train of CAC is available.
LCO 3.6.10	Iodine Cleanup System (ICS)	Two ICS trains inoperable	No condition defined. Default 3.0.3 entry	24 hours CT for restoring one train. Allow MODE 4 end state.
LCO 3.6.1	Containment (CTMT)	Inoperable	Defined 1 hour shutdown (MODE 5 in 36 hours)	8 hours CT for restoring containment operability. Allow MODE 4 end state if containment leakage is excessive due to reasons other than the inoperability of two or more Containment Isolation Valves in the same flow path.
LCO 3.6.13	Shield Building Exhaust Air Cleanup System (SBEACS)	Two trains inoperable	No condition defined. Default 3.0.3 entry	24 hours CT for restoring one train. Allow MODE 4 end state.

WITHDRAWN

ISTS #	SYSTEM	INOPERABILITY CONDITION	CURRENT ACTION AND ASSOCIATED COMPLETION TIME (CT)	PROPOSED CHANGES: COMPLETION TIME(CT) AND END STATE
LCO 3.7.11	Control Room Emergency Air Cleanup System (CREACS)	Two trains inoperable	No condition defined. Default 3.0.3 entry	24 hours CT for restoring one train (or the time to reach 5 REM, which may be less than 24 hours). Proposed change applies to radiation protection function only. Allow MODE 4 end state.
LCO 3.7.12	Control Room Emergency Air Temperature Control System (CREATCS)	Two trains inoperable (ISTS Condition E)	Explicit 3.0.3	24 hours CT for restoring one train. Allow MODE 4 end state.
LCO 3.7.13	Emergency Core Cooling System (ECCS) Pump Room Exhaust Air Cleanup System (ECCS PREACS)	Two trains inoperable	No condition defined. Default 3.0.3 entry	24 hours CT for restoring one train. Allow MODE 4 end state.
LCO 3.7.15	Penetration Room Exhaust Air Cleanup System PREACS)	Two trains inoperable	No condition defined. Default 3.0.3 entry	24 hours CT for restoring one train. Allow MODE 4 end state.

WITHDRAWN

* Also know as containment air recirculation coolers (CARC)

WCAP-16125-NP documents a risk-informed analysis of the proposed TS changes. Probabilistic Risk Assessment (PRA) results and insights are used, in combination with results of deterministic assessments, to identify and justify the proposed TS changes for all CE NSSS design plants. This is in accordance with guidance provided in Regulatory Guides (RGs) 1.174 and 1.177 (References 6 and 7, respectively).

The approach used to assess the risk impact of the proposed changes is discussed and evaluated in Section 3. Section 4 evaluates the results of the risk assessment. Section 5 provides integrated justifications (i.e., both probabilistic and deterministic arguments) for each of the proposed system-specific TS changes. Finally, Section 6 summarizes the staff's conclusions from the review of the proposed TS changes.

3.0 RISK ASSESSMENT APPROACH

The objective of the WOG's risk assessment was to show that the implementation of the proposed TS changes are not expected to lead to any significant risk increases. In performing the risk-informed assessments and interpreting the results, the following two assumptions are tacitly made:

- 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 LCO 3.0.3 conditions are rare. Furthermore, when such a condition does arise, the actual cause of the inoperability is often due to an incomplete "paper trail" or a partial system failure rather than a deleterious common-cause failure of critical components leading to a functional failure of an entire system.

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

The risk impact of the proposed TS changes was assessed following the three-tiered approach recommended in RG 1.177 for evaluating proposed extensions in currently allowed Completion Times (CTs):

- The first tier involves the assessment of the change in plant risk due to the proposed TS change. Such risk change is expressed (1) by the change in the average yearly core damage frequency (ΔCDF) and the average yearly large early release frequency (ΔLERF) and (2) by the incremental conditional core damage probability (ICCDP) and the incremental conditional large early release probability (ICLERP). The assessed ΔCDF and ΔLERF values are compared to acceptance guidelines, consistent with the Commission's Safety Goal Policy Statement as documented in RG 1.174, so that the plant's average baseline risk is maintained within a minimal range. The assessed ICCDP and ICLERP values are compared to acceptance guidelines provided in RG 1.177 which aim at ensuring that the plant risk does not increase unacceptably during the period the equipment is taken out of service.

- The second tier involves the identification of potentially high-risk configurations 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 equipment testing were also involved. The objective is to ensure that appropriate restrictions are in place to avoid any potential high-risk configurations.

- The third tier involves the establishment of an overall configuration risk management program (CRMP) to ensure that potentially risk-significant configurations resulting from maintenance and other operational activities are identified. The objective of the CRMP is to manage configuration-specific risk by appropriate scheduling of plant activities and/or appropriate compensatory measures.

The approach used in implementing the three-tiered approach of RG 1.177 to support the proposed TS changes is summarized and evaluated in the following Sections 3.1 to 3.3.

3.1 Risk Impact

This section summarizes and evaluates the approach used to assess the risk impact of the proposed extensions of the time interval for initiating plant shutdown when a safety system or function are unavailable. These conditions are associated with LCO 3.0.3 which currently

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requires initiating plant shutdown within one hour from the time such conditions are discovered. The risk impact measures, and associated success criteria, which are used in the risk-informed decision making process are defined and evaluated in Section 3.1.1. The methodology which was used to assess the various risk impacts of changes, covering a diverse range of systems and components with different impacts on plant risk, are discussed and evaluated in Sections 3.1.2 to 3.1.4.

3.1.1 Risk Impact Measures and Acceptance Criteria

The guidance provided in RG 1.177 addresses only systems/components contributing to core damage frequency (CDF) and/or large early release frequency (LERF). However, in the risk-informed analysis provided by the WOG, the philosophy of the three-tiered approach was extended to encompass also TS changes involving systems/components contributing to radiation release prevention other than large early release. For this purpose, appropriate risk measures (similar to ΔLERF and ICLERP) and acceptance criteria were introduced for systems whose function is to prevent radiation releases other than large early release.

The ICCDP associated with the proposed extension of the time interval for initiating plant shutdown when a system is declared inoperable, a condition which currently requires an immediate plant shutdown (within one hour), is expressed by the following equation:

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$$ICCDP = \Delta R_{CDF} \cdot d = (R_{1,CDF} - R_{o,CDF}) \cdot d \qquad (1)$$

where:

ΔR_{CDF} = the conditional risk increase, in terms of CDF, caused by the specified system's unavailability,

d = the proposed extension of the time interval during which the plant is allowed to keep operating at power given the condition,

$R_{1,CDF}$ = the plant CDF with the system (or component) permanently unavailable,

$R_{o,CDF}$ = the plant CDF without the proposed time extension.

A similar expression is used for ICLERP by substituting the appropriate measure of risk, i.e., LERF instead of CDF.

$$ICLERP = \Delta R_{LERF} \cdot d = (R_{1,LERF} - R_{o,LERF}) \cdot d \qquad (2)$$

where:

ΔR_{LERF} = the conditional risk increase, in terms of LERF, caused by the specified system's unavailability,

d = the proposed extension of the time interval during which the plant is allowed to keep operating at power given the condition,

$R_{1,LERF}$ = the plant LERF with the system (or component) permanently unavailable,

$R_{o,LERF}$ = the plant LERF without the proposed time extension.

The changes in CDF and LERF (i.e., ΔCDF and ΔLERF, respectively) for each system or component are obtained by multiplying the respective ICCDP and ICLERP values by the yearly frequency, f, the system or component is expected to be declared inoperable:

$$\Delta CDF = ICCDP \cdot f \quad (3)$$

$$\Delta LERF = ICLERP \cdot f \quad (4)$$

For TS changes related to systems or components contributing to accident initiation and/or core damage prevention, the risk impact is measured in terms of both CDF and LERF. For these systems values for ICCDP and ΔCDF as well as for ICLERP and ΔLERF were assessed. For TS changes related to systems or components contributing to large early release prevention, the risk impact is measured only in terms of LERF. Therefore, for these systems only values for ICLERP and ΔLERF were assessed.

Risk impact measures similar to ICLERP and ΔLERF were introduced for systems and components whose function is to prevent radiation releases other than large early release. These risk impact measures are (1) the incremental conditional radiation release (above TS limits) probability, ICRRP, and (2) the change in the radiation release (above TS limits) frequency, ΔRRF. A similar expression as the ones used for ICCDP and ICLERP (equations 1 and 2, respectively) can be used for ICRRP by substituting the appropriate measure of risk, i.e., radiation release (above TS limits) frequency (RRF) instead of CDF:

WITHDRAWN

$$ICRRP = \Delta R_{RRF} \cdot d = (R_{1,RRF} - R_{o,RRF}) \cdot d \quad (5)$$

where:

ΔR_{RRF} = the conditional risk increase, in terms of RRF, caused by the specified system's unavailability,

d = the proposed extension of the time interval during which the plant is allowed to keep operating at power given the condition,

$R_{1,RRF}$ = the plant RRF with the system permanently unavailable,

$R_{o,RRF}$ = the plant RRF without the proposed time extension.

The change in RRF (i.e., ΔRRF) for each system or component is obtained by multiplying the respective ICRRP value by the yearly frequency, f, the system or component is expected to be declared inoperable:

$$\Delta RRF = ICRRP \cdot f \quad (6)$$

The assessed ICRRP and ΔRRF values are compared to acceptance criteria similar to the ones reported in RGs 1.177 and 1.174 for core damage and large early release risks, respectively. The results of the risk assessments, in terms of the various risk measures, and their comparison to acceptance criteria are discussed in Section 4.

The acceptance criteria for core damage and large early release risks are selected based on guidance provided in RGs 1.174 and 1.177. Regulatory Guide 1.174 indicates that a ΔCDF smaller than 1E-6/year and a ΔLERF smaller than 1E-7/year are considered very small. Therefore, the length of the extended time intervals which are proposed as an alternate to the immediate entry into the current LCO 3.0.3 requirements were selected to satisfy these criteria.

Acceptance guidelines provided in Regulatory Guide 1.177 for evaluating the core damage and large early release risks associated with a "single AOT entry" are also considered. These guidelines, which are based on traditionally acceptable levels of risk increases during equipment outages for maintenance activities, indicate that an ICCDP smaller than 5E-7/year and an ICLERP smaller than 5E-8/year are considered very small. Therefore, the length of the extended time intervals are selected to also satisfy these criteria. However, small deviations are allowed in some cases in agreement with the following statements included in Regulatory Guide 1.177:

- The acceptance guidelines are intended to provide an indication in numerical terms of what is considered acceptable and, therefore, should not be interpreted as being overly prescriptive.

- The intent in comparing the PRA results with the acceptance guidelines of RG 1.177 is to demonstrate, with reasonable assurance, consistency with the Commission's Safety Goal Policy Statement so that the plant's average baseline risk is maintained within a minimal range (i.e., meet the acceptance criteria for ΔCDF and ΔLERF). In this case, since the entry into LCO 3.0.3 is a significantly less frequent event than LCO entries on which the numerical values mentioned in RG 1.177 are based, the acceptance criteria for ΔCDF and ΔLERF can be met with reasonable assurance even when the numerical values of the acceptance guidelines provided in RG 1.177 are not strictly met. The frequency of events leading to LCO 3.0.3 is not expected to increase significantly following the proposed change because such events may be reportable and may require a licensee event report, per NUREG-1022 Revision 2 (Reference 8). In addition, events leading to LCO 3.0.3 are used in performance indicators and the reactor oversight program.

- The numerical guidelines should be taken into account along with other considerations, including operating experience and practical considerations associated with test and maintenance practices. An important consideration, related to LCO 3.0.3 entry, is the need for adequate time for plant staff to diagnose a condition resulting in the inoperability of a system and restore its function and, thus, avoid transition to shutdown risk.

WITHDRAWN

As shown by the results discussed below in Section 4, the above mentioned RG 1.177 criteria are strictly met for most of the proposed TS-specific changes. For a few cases, small deviations are used in accordance with the above listed statements. As discussed in Section 4, even in such cases the RG 1.177 criteria would most likely be strictly met if the avoided shutdown risk and the effect of the various conservative assumptions had been taken into account.

Acceptance criteria for radiation release risks, other than large early release risks, are also defined. It is conservatively assumed that a ΔRRF value smaller than 1E-7 per year (i.e., the same as for a large release) is considered very small and, therefore, acceptable. In addition, in order to ensure that the acceptance criterion for ΔRRF will be met, the ICRRP value for each entry is required to be smaller than 5E-7 (i.e., the same value used in the criterion for ICCDP). It should be noted that the conservative acceptance criteria for radiation release risks, other than large early release risks, are introduced for the purposes of this evaluation and should not be generalized or interpreted as NRC guidance for other risk-informed applications.

3.1.2 Approach for Assessing Core Damage Risk Impact

The proposed TS changes involve two categories of systems and components which require different approaches for assessing the risk impact, in terms of core damage, associated with the unavailability of such systems and components. The first category includes standby safety systems and components which contribute to accident mitigation and core damage prevention. The second category refers to systems and components whose unavailability contributes to core damage through the initiation of accidents.

<u>Systems Contributing to Accident Mitigation</u>

WITHDRAWN

A bounding generic approach for evaluating the ICCDP and ΔCDF values, for each of the proposed TS changes associated with systems and components in this category, is used. The conditional core damage risk increase, ΔR_{CDF}, which is used to assess the value of ICCDP according to equation (1) and the value of ΔCDF according to equation (3) of Section 3.1.1, is conservatively assumed to be the same as the system's (or component's) yearly challenge frequency. This assumption is conservative because in many cases plants have the capability to mitigate a range of events using alternative equipment and procedures or using available equipment which are less than the optimal set. An example of an alternative success path, for which no credit is taken in the assessment when two HPSI trains are unavailable, is the ability to use an aggressive reactor coolant system (RCS) cooldown to minimize inventory loss and allow alternative makeup capability using the charging pumps or the LPSI pumps. An example of using available equipment which are less than the optimal set, for which no credit is taken in the assessment when two or more SITs are unavailable, is the ability to use the available SITs together with the LPSI pumps to mitigate a significant range of large LOCAs.

A system's or component's challenge frequency was assessed quantitatively by considering all accident sequences by accident initiating event, modeled in the probabilistic risk assessment (PRA) of representative WOG member plants with CE NSSS designs, that require operation of the specific system or component to mitigate the accident. The assessed challenge frequencies, for all systems and components contributing to accident mitigation that are associated with the proposed TS changes, are reported in Table 2.

Table 2 Challenge Frequency of Mitigating System or Component by Accident Initiating Event.

System/ Component Unavailable	Event Frequency (per year)						Component Challenge Frequency (per year)
	LOCA	SGTR	Stuck Open PORV	Stuck Open PSV	Transients with F&B	ATWS	
SIT	5.0E-6	0.0	0.0	0.0	0.0	0.0	5.0E-6
LPSI	4.5E-5	~0.0	0.0	0.0	0.0	0.0	4.5E-5
HPSI (plants with PORVs)	5.5E-4	7.0E-5	1.0E-3	2.5E-3	1.0E-3	0.0	5.1E-3
HPSI (plants w/o PORVs)	5.5E-4	7.0E-5	0.0	2.5E-3	0.0	0.0	3.1E-3
CS (plants w/o CARCs)	5.5E-4	~0.0	~0.0	~0.0	~0.0	~0.0	5.5E-4
PORV	0.0	0.0	0.0	0.0	1.0E-3	8.4E-6	1.0E-3
Boration System	0.0	0.0	0.0	0.0	0.0	1.7E-5	1.7E-5

WITHDRAWN

It should be noted that only internal initiating events are considered in estimating a system's or component's challenge frequency. However, the risk assessment results would not be significantly different to impact any conclusions had external initiating events been considered, for two reasons. First, the system's (or component's) challenge frequency used in the risk assessments was estimated conservatively. The staff's review indicated that, for the most limiting cases, the challenge frequency estimates used in the assessments are as much as two times higher than estimates found in most PRA applications using internal events only. Second, the proposed TS changes will be implemented with certain requirements which will be described in the WOG implementation guidance document. The purpose of these requirements is to limit the challenge frequency of an inoperable system or component and ensure that appropriate compensatory measures are in place. Such requirements are expected to reduce the challenge frequency of an inoperable system or component from all initiating events, including external events.

Table 2 also reports contributions to a system's (or component's) total challenge frequency that are associated with important accident categories analyzed in PRAs, such as LOCAs and transients with stuck open PORVs. The total challenge frequency of a system or component is the sum of all contributions associated with the various accident categories.

In addition to the conservatism introduced by assuming that a system's or component's challenge frequency equals the conditional core damage risk increase, ΔR, when the system or component is unavailable, the challenge frequency values themselves are assessed conservatively. For example, the value of the challenge frequency of the feed-and-bleed (F&B) function, which is calculated as the product of the frequency of loss of main feedwater (MFW) and the unavailability of auxiliary feedwater (AFW), is based on bounding values for both the frequency of initiating events leading to loss of MFW (0.5/yr) and the unavailability of the AFW (2E-3).

The change in CDF, ΔCDF, for each system or component is obtained by multiplying the respective ICCDP value by the yearly frequency, f, the system or component is expected to be declared inoperable (i.e., $\Delta CDF = ICCDP \cdot f$). For the purpose of this study, the value of the yearly frequency, f, is conservatively assumed to be equal to 0.2 (i.e., once every five years) and the same for all systems and components associated with the proposed TS changes. The assumed frequency value is based on an industry review of licensee event reports which concluded that entry into LCO 3.0.3 due to any inoperability of multiple trains or components has occurred with a frequency of less than once per five years. This average value is conservative for two reasons. First, the actual LCO 3.0.3 entry frequencies associated with the inoperability of risk important systems contributing to core damage and/or large early release frequency, such as the HPSI system, are significantly smaller than the 0.2 per year value which is an average for all systems and components. Second, experience indicates that when an entry into an LCO 3.0.3 is required, the system is often partially functional (e.g., an ECCS inoperability that occurred on February 5,1998 at SONGS-2 affected the recirculation mode of operation only) and the condition can be corrected in a short time (e.g., less than one hour).

WITHDRAWN

Based on the above discussion, the staff concludes that the approach used to assess the core damage risk impact, for systems and components contributing to accident mitigation, is acceptable.

Systems Contributing to Accident Initiation

The proposed TS changes include only one system in this category, the Class 1E pressurizer heaters. ISTS LCO 3.4.9 includes requirements for both groups of safety-related pressurizer heaters to have minimum heating power and emergency power supply capability. The safety-related pressurizer heaters have two primary functions. One function is to keep the reactor coolant in a subcooled condition with natural circulation following a loss of offsite power (LOOP) event during which the normally available station powered non-safety related heaters become unavailable. Although no credit is taken in design basis accident analyses for the pressurizer heaters, they have been included in the TS because they are needed to maintain long term subcooling during a LOOP event. However, pressurizer heaters are not required to achieve a post-trip plant cooldown since successful cooldown can be achieved, with minimal impact on plant risk, due to the availability of reactor vessel and pressurizer vents. Consequently, the pressurizer heaters do not have a significant role in the mitigation of core damage events.

A second function of the safety-related pressurizer heaters is to back up the station powered non-safety related heaters which are normally available to control reactor coolant pressure during steady state operation. The unavailability of these heaters would reduce the plant's ability to control the normal operating parameters and consequently will increase the potential of plant trip. The approach used to assess the risk associated with this increased plant trip potential is described below.

The conditional core damage risk increase, ΔR_{CDF}, which is used to assess the value of ICCDP according to equation (1) and the value of ΔCDF according to equation (3) of Section 3.1.1, is calculated as the product of the increase in the initiating event (reactor trip) frequency, ΔIE, associated with the unavailability of the safety-related heaters and the probability of core damage given a normal plant high or low pressure trip, CDP_{trip} (i.e., $\Delta R_{CDF} = \Delta IE \cdot CDP_{trip}$). Bounding values of ΔIE (0.05 plant trip events/day or, equivalently, 18.3 plant trip events/year) and CDP_{trip} (6E-6) are used in the assessment. The value of ΔIE was calculated by assuming that the plant trip frequency increases by over an order of magnitude (from 0.004 to 0.05 events/day) when the safety-related pressurizer heaters are unavailable. This assumption is conservative because the normal station powered (non-class 1E) heaters are available for reactor coolant pressure control. It should be noted that the availability of the non-class 1E heaters will be required for extending plant shutdown initiation, as proposed. The assumed value of CDP_{trip} (6E-6) is based on previous PRA results and bounds the assessed values for WOG member plants with CE NSSS designs.

WITHDRAWN

The change in CDF, ΔCDF, is obtained by multiplying the respective ICCDP value by the yearly frequency, f, the class 1E pressurizer heaters are expected to be declared inoperable (i.e., ΔCDF = ICCDP · f). The value of the yearly frequency, f, is conservatively assumed to be equal to 0.2. The rationale is the same as for the case of systems and components contributing to accident mitigation discussed in this section.

The staff concludes that the approach used to assess the core damage risk impact, for systems and components contributing to accident initiation, makes use of conservative assumptions and is acceptable.

3.1.3 Approach for Assessing Large Early Release Risk Impact

The proposed TS changes involve two categories of systems and components which require different approaches for assessing the risk impact, in terms of large early release frequency, associated with the unavailability of such systems and components. The first category includes safety systems and components whose unavailability leads to a CDF increase and, therefore, also to a LERF increase. The second category refers to systems and components whose function is to mitigate the effects of a core damage and prevent large early releases (i.e., their unavailability leads to a direct LERF increase).

<u>Systems Contributing to Accident Initiation and/or Core Damage Prevention</u>

A bounding generic approach is used to assess the ICLERP and ΔLERF values for each of the proposed TS changes associated with systems and components in this category. The conditional large early release risk increase, ΔR_{LERF}, which is used to assess the value of ICLERP according to equation (2) of Section 3.1.1, is calculated as the product of two terms:

the system's or component's conditional core damage risk increase, ΔR_{CDF} (which is conservatively assumed to be equal to the system's or component's yearly challenge frequency as discussed in Section 3.1.2) and the conditional large early release probability given core damage, CLERP (i.e., $\Delta R_{LERF} = \Delta R_{CDF} \cdot CLERP$). The staff noted that there could be an additional contribution to ΔR_{LERF} which has been ignored in the assessment. Such contribution is associated with core damage sequences where the containment's ability to retain the damaged core is impacted by the system's or component's unavailability without a corresponding CDF increase. However, the staff's review has determined that, for the systems and components impacted by the proposed changes, the impact of this omission on the results of the risk assessment (i.e., the assessed value of ΔR_{LERF}) is too small to change any conclusions.

A large early release is defined as the significant and unmitigated release of radioactivity from the containment in a time frame prior to effective evacuation of population such that there is a potential for early health effects. Such a release results from events which lead to containment failure at or shortly after vessel breach, containment bypass events and loss of containment isolation. A WOG review of large early release scenarios for the CE designed pressurized water reactors (PWRs) indicated that early releases arise as a result of one of three classes of scenarios: (1) containment bypass events; (2) severe accidents accompanied by loss of containment isolation; and (3) containment failure associated with energetic events in the containment. The first class of scenarios includes interfacing system loss of coolant accidents (ISLOCAs) and steam generator tube ruptures (SGTRs) with a concomitant loss of steam generator (SG) isolation (e.g., due to stuck open main steam isolation valves). The second class of scenarios includes any severe accident in conjunction with failure to isolate the containment. The third class of scenarios includes high-pressure melt ejection (HPME) phenomena, including direct containment heating (DCH), and hydrogen conflagrations and detonations. A simplified large early release (LER) event tree was developed and used to assess conservatively the contributions to CLERP from each of these three classes of scenarios which are likely to result in a large early release. The sum of such contributions is the total plant CLERP.

WITHDRAWN

The CLERP value is the fraction of the conditional core damage risk increase, ΔR_{CDF}, which propagates into a large early release event. Bounding CLERP values were assessed, for each system and component associated with the proposed TS changes, using a simplified LER event tree with the following top events:

- Containment isolation following core damage;
- High reactor coolant system (RCS) pressure at core damage;
- Secondary side depressurization of the SGs;
- Thermally-induced SGTR; and
- Containment failure due to reactor pressure vessel (RPV) lower head failure.

The assessed CLERP values are based on conservative assumptions and data, such as the following:

- The probability of containment isolation failure was assumed to be 3E-3. This value is based on containment isolation failure data used in PRAs performed by WOG member utilities with CE NSSS designs which range between 1E-4 and 3E-3.

- All incremental CDF arising from PORV or boration system unavailability results in a high RCS pressure plant damage state with no possibility of operator actions to depressurize the RCS prior to failure of the reactor vessel lower head.

- 20% of incremental CDF arising from HPSI system unavailability for plants with PORVs results in a high RCS pressure plant damage state. This is a conservative assumption since the HPSI system is primarily used to mitigate moderate and low pressure events.

- 50% of incremental CDF arising from a reactor trip induced by the unavailability of the safety-related pressurizer heaters results in a high RCS pressure plant damage state.

- The probability of SG tube failure (prior to failure of the RCS) when exposed to high pressure core damage states is assumed to be 50% (assuming secondary side depressurization). This probability is associated with severely degraded SG tubes and therefore is conservative. In addition, it is assumed that all thermally-induced SGTR events are large early releases.

- The probability of SG depressurization, via operator action or failure of a main steam safety valve (MSSV) to close, is assumed to be 10% (successful SG isolation is needed to maintain SG tube integrity for high RCS pressure plant damage states). It should be noted that sensitivity studies were performed by the NRC staff with the probability of SG depressurization increased to 30% which is very conservative.

- The conditional containment failure probability due to high pressure melt injection (HPMI) is assumed to be 1%.

WITHDRAWN

The quantification of the simplified LER event tree, with the above mentioned conservative assumptions, resulted in the following five large early release scenarios (labeled CLERP-1 to CLERP-5) assuming a core damage event:

CLERP-1: High RCS pressure plant damage state, the containment is isolated, the SG is depressurized due to a stuck open MSSV and thermally-induced SG tube rupture occurs resulting in a large early release.

CLERP-2: High RCS pressure plant damage state, the containment is isolated and the SG is depressurized due to a stuck open MSSV with the SG tubes intact but an HPME failure occurs resulting in a large early release.

CLERP-3: High RCS pressure plant damage state, the containment is isolated and the SG pressurized but thermally-induced SG tube rupture occurs resulting in a large early release.

CLERP-4: High RCS pressure plant damage state, the containment is isolated and the SG pressurized with tubes intact but an HPME failure occurs resulting in a large early release.

CLERP-5: Failure to isolate the containment which results in a large early release.

The assessed conditional large early release probabilities, given core damage, for these scenarios are summarized in Table 3 for each applicable system or component associated with the proposed TS changes. The sum of such contributions is the total plant CLERP.

Table 3 Conditional Large Early Release Probability (CLERP) Estimates, Given Core Damage, Due to Unavailability of Mitigating Systems and Components.

	CLERP-1	CLERP-2	CLERP-3	CLERP-4	CLERP-5	CLERP
SIT	0.0	0.0	0.0	0.0	3.0E-3	3.0E-3
LPSI	0.0	0.0	0.0	0.0	3.0E-3	3.0E-3
HPSI (plants with PORVs)	1.0E-2	1.1E-4	1.8E-3	2.0E-3	3.0E-3	1.7E-2
HPSI (plants w/o PORVs)	1.0E-2	1.1E-4	1.8E-3	2.0E-3	3.0E-3	1.7E-2
CS (plants w/o CARCs)	1.0E-2	1.1E-4	1.8E-3	2.0E-3	3.0E-3	1.7E-2
PORV	5.0E-2	5.5E-4	9.0E-3	9.9E-3	3.0E-3	7.2E-2
Boration System	5.0E-2	5.5E-4	9.0E-3	9.9E-3	3.0E-3	7.2E-2
Pressurizer Heaters	2.5E-2	2.7E-4	4.5E-3	4.9E-3	3.0E-3	3.8E-2

WITHDRAWN

Even though the assessed CLERP values are based on conservative assumptions and data, a sensitivity study was performed on two key parameters which are dominant CLERP contributors. These parameters are: (a) the probability that a thermally-induced SGTR will occur in advance of another RCS structure failure, and (b) the probability that an MSSV will fail open depressurizing one SG. The objective of the sensitivity study was to provide reasonable assurance that there are no areas of uncertainty, in modeling assumptions and data, which could have a significant impact on results and conclusions. The results of the sensitivity study are discussed and evaluated in Section 4.

The staff finds that conservative assumptions and data were used to assess the values of the quantities (e.g., ΔR_{CDF}, LERP and f) needed to estimate the change in LERF (i.e., ΔLERF). The bounding nature of the approach used to assess the large early release risk impact, in combination with the performance of a sensitivity study on two key parameters which are dominant CLERP contributors, provide adequate assurance about the robustness of the results used to support the proposed TS changes.

Systems Contributing to Large Early Release Prevention Given Core Damage

This category refers to systems and components whose function is to mitigate the effects of a core damage and prevent large early releases (i.e., their unavailability leads to a direct LERF increase). The only component in this category which is associated with the proposed TS changes is the containment. Since large early releases result from a gross opening in the containment, such as a stuck open purge valve, it is assumed that all core damage events proceed to a large early release. Thus, the conditional large early release risk increase, ΔR_{LERF}, which is used to assess the values of ICLERP and ΔLERF according to equations (2) and (4) of Section 3.1.1, respectively, is equal to the plant's CDF (assumed to be 1E-4/year). The staff review finds that this approach is bounding because it is conservatively assumed that (1) any containment inoperability will completely degrade the containment and (2) a plant's CDF is 1E-4/year (for most plants the CDF at power operation from internal and external events is less than1E-4/year).

3.1.4 Approach for Assessing Non-Large Early Release (Non-LER) Risk Impact

This category refers to systems and components whose function is to prevent radiation releases other than large early release. Availability of such equipment is typically required to meet design basis dose limits. The conditional non-LER risk increase, ΔR_{RRF}, which is used in assessing the value of ICRRP according to equation (5) as well as the value of and ΔRRF according to equation (6) of Section 3.1.1 is equal to the plant's CDF (assumed to be 1E-4/year) except for the emergency core cooling system (ECCS) penetration room emergency air cleanup system (PREACS). The value of ΔR_{RRF} for the ECCS PREACS was assessed to be 4.5E-5/year since the challenge of this system is limited to core damage events associated with large and medium LOCAs only. The staff review finds that this approach is bounding because it is conservatively assumed that (1) equipment in this category, except for the ECCS PREACS, are challenged (i.e., with probability of 1) during a core damage event and (2) a plant's CDF is 1E-4/year (for most plants the CDF at power operation from internal and external events is less than1E-4/year). The staff also finds that the challenge frequency of 4.5E-5/year assumed for the ECCS PREACS is bounding because this system will not be challenged unless the core damage event is associated with large and medium LOCAs only.

WITHDRAWN

3.1.5 Uncertainty/Sensitivity Evaluation

As stated in Sections 3.1.2 to 3.1.4, the performed risk assessments in support of the proposed TS changes are based on many conservative assumptions. Nevertheless, areas of potential uncertainty, in both data and modeling assumptions, were identified which could have an impact on results and conclusions. Some of these areas of uncertainty are associated with design and operational variability among the various WOG member plants with CE NSSS designs. The following areas of potential uncertainty were identified (all associated with systems or components whose loss of function contributes to CDF or LERF):

- The system's or component's challenge frequency,
- The system's or component's average loss of function frequency,
- The probabilities of two key parameters which are dominant CLERP contributors:
  (a) the probability of occurrence of a thermally-induced SGTR in advance of another

RCS structure failure (event TI-SGTR); and (b) the probability that an MSSV will fail open depressurizing one SG (event SGD).

These areas of uncertainty were evaluated, by performing the following two sensitivity studies (for each system or component), to determine how they impact the results and conclusions of the risk assessments:

- In sensitivity case #1, the risk impact was re-assessed with simultaneous changes of the values of three parameters as follows:
  - The probability of occurrence of a thermally-induced SGTR in advance of another RCS structure failure (event TI-SGTR) was increased from 50% (base case) to 100%;
  - the probability that an MSSV will fail open depressurizing one SG (event SGD) was increased from 10% to 30%;
  - the frequency of loss of function was increased from once every five years to once every three years.
- In sensitivity case #2, the risk impact was re-assessed using the upper 95th percentile value of the system's or component's challenge frequency.

WITHDRAWN

The staff concludes that the performed sensitivity studies, in conjunction with the many conservative and bounding assumptions and data used in the base case analyses (discussed in Sections 3.1.2 to 3.1.4), provide adequate information regarding the robustness of the risk assessment results.

3.2 Identification of Potentially High Risk Configurations

The second tier of the three-tiered approach recommended in RG 1.177 involves the identification of potentially high-risk configurations that could exist if equipment, in addition to that associated with the TS change, were to be taken out of service simultaneously. Insights from the risk assessments, in conjunction with important assumptions made in the analyses and defense-in-depth considerations, were used to identify such configurations. To avoid these potentially high-risk configurations, specific restrictions to the implementation of the proposed TS changes were considered. For example, when the safety-related pressurizer heaters are unavailable, the availability of the non-class 1E heaters will be required for extending plant shutdown initiation. These restrictions should be part of the industry's implementation guidance and will be evaluated as part of the NRC staff review of the ISTS changes.

3.3 Configuration Risk Management

The third tier of the three-tiered approach recommended in RG 1.177 involves the establishment of an overall configuration risk management program (CRMP) to ensure that potentially risk-significant configurations resulting from maintenance and other operational activities are identified. The objective of the CRMP is to manage configuration-specific risk by appropriate scheduling of plant activities and/or appropriate compensatory measures. This objective is met by licensee programs to comply with the Maintenance Rule 10 CFR 50.65 (a)(4) requirement to assess and manage risk resulting from maintenance and other operational

activities. These programs can support licensee decision making regarding the appropriate actions to control risk whenever a risk-informed TS is entered.

4.0 RISK ASSESSMENT RESULTS AND INSIGHTS

The risk assessment approach, documented in Section 3, was implemented to the system-specific technical specifications listed in Table 1 (Section 2). In addition to the base case risk assessments, two sensitivity cases (discussed below) were performed to draw insights regarding the robustness of the results. The core damage and large early release risk impacts (base case) are summarized in Tables 4 and 5, respectively. In addition to the base case results, the results of two sensitivity cases are reported in Tables 6 and 7, for proposed TS changes related to systems and components contributing to core damage and/or large early release. Finally, the assessed risk impacts for systems and components which contribute to non-LER are summarized in Table 8. As discussed below, no sensitivity analyses were necessary for systems and components contributing to non-LER due to the consistently conservative assumptions used in the analysis in conjunction with the conservative interpretation of the risks associated with such systems and components.

Table 4 summarizes the risk impact, in terms of CDF change, ΔCDF, for each of the proposed TS changes related to systems and components contributing to core damage and/or large early release. The first column lists the systems and components for which a TS change is proposed. The second column lists the new proposed completion time (CT) to restore the system's or component's function before initiation of plant shutdown is required according to LCO 3.0.3. The third column is the conditional risk increase, in terms of CDF, ΔR_{CDF}, caused by the system's or component's loss of function. As discussed in Section 3, ΔR_{CDF} was conservatively assumed to be the same as the system's or component's challenge frequency. The fourth column lists the ICCDP values for continued plant operation at power for the entire proposed CT given loss of the system's or component's function. The ICCDP values are obtained by multiplying the proposed CT value (column 2) by ΔR_{CDF} (column 3). The fifth column lists the assessed average expected CDF changes, ΔCDF, associated with the proposed CT extensions. The ΔCDF values are obtained by multiplying the corresponding ICCDP value by the average frequency of loss of function (which in the base case risk assessment was assumed to occur once every five years for all the systems and components listed in the first column). The assessed ΔCDF values (fifth column of Table 4) indicate that all proposed TS changes for the systems and components listed in Table 4 would have an insignificant risk impact with respect to core damage (significantly less than the acceptance criterion of 1E-6/year for core damage frequency based on guidance provided in RGs 1.174 and 1.177). This conclusion is true without any credit for the averted risk associated with avoiding plant shutdown or any consideration for the many conservative assumptions used in the risk assessment.

The assessed ICCDP values (fourth column of Table 4) are also within RG 1.177 acceptance guidelines. With the exception of HPSI and PORVs, the ICCDP is smaller than 5E-7 which is considered very small and acceptable. Based on the arguments presented in Section 3, where the "acceptance criteria" are discussed, the small ICCDP deviations from the numerical guideline of 5E-7 shown for HPSI and PORVs in Table 1 are in agreement with guidance provided in RG 1.177 and, therefore, acceptable. It should be noted that the intent of the numerical guideline for ICCDP is to ensure that the acceptance criterion for ΔCDF is met. In

WITHDRAWN

this case, since the entry into LCO 3.0.3 is a significantly less frequent event than LCO entries on which the numerical values mentioned in RG 1.177 are based, the acceptance criterion for ΔCDF can be met with reasonable assurance even when the numerical value of the acceptance guideline for ICCDP provided in RG 1.177 is not strictly met. Furthermore, the RG 1.177 numerical guideline of 5E-7 would most likely be strictly met also for HPSI and the PORVs if the avoided shutdown risk and the effect of the various conservative assumptions had been taken into account.

Table 4 Core Damage Risk Impact Results.

Component/System	Proposed CT (hrs)	ΔR_{CDF}/yr or Challenge Frequency/yr	ICCDP	ΔCDF
SIT	24	5.0E-6	1.3E-8	2.6E-9
LPSI	24	4.5E-5	1.2E-7	2.4E-8
HPSI (Plants w/ PORVs)	4	5.1E-3	1.7E-6	3.5E-7
HPSI (Plants w/o PORVs)	4	3.1E-3	1.1E-6	2.1E-7
CS (no CARC available)	12	5.5E-4	6.9E-7	1.4E-7
PORV	8	1.0E-3	8.1E-7	1.6E-7
Boration System	24	1.7E-5	4.5E-8	9.0E-9
Pressurizer Heaters	24	1.0E-4	2.9E-7	5.8E-8
Containment	8	1.0E-4	N/A	N/A

WITHDRAWN

An observation of the values of the conditional risk increase (in terms of CDF, ΔR_{CDF}), listed in the third column of Table 4, indicates that the loss of function for any of the systems/components listed in column 1 would be found by the significance determination process (SDP) to be a risk significance event. Therefore, licensees will have no incentive to allow the current low frequency of these events to increase after the proposed extensions are granted.

Table 5 summarizes the risk impact, in terms of LERF change, ΔLERF, for each of the proposed TS changes related to systems and components contributing to large early release. The first three columns are identical with those in Table 4. The fourth column lists the total plant CLERP values (from Table 3) given the unavailability of the corresponding system or component listed in the first column. The fifth column is the conditional risk increase, in terms of LERF, ΔR_{LERF}, caused by the corresponding system's or component's loss of function. The ΔR_{LERF} value for each system and component list in the first column is obtained by multiplying the ΔR_{CDF} value (column 3) by the corresponding CLERP value (column 4). The sixth column lists the ICLERP values for continued plant operation at power for the entire proposed CT given loss of the system's or component's function. The ICLERP value for each system and component listed in the first column is obtained by multiplying the proposed CT value (column

2) by the corresponding ΔR_{LERF} value (column 3). Finally, the seventh column lists the assessed average expected LERF changes, ΔLERF, associated with the proposed CT extensions. The ΔCDF values are obtained by multiplying the corresponding ICLERP value by the average frequency of loss of function (which in the base case risk assessment was assumed to occur once every five years for all the systems and components listed in the first column).

Table 5 Large Early Release Risk Impact Results.

Component/System	Proposed CT (hrs)	ΔR_{CDF}/yr	CLERP	ΔR_{LERF}/yr	ICLERP	ΔLERF/yr
SIT	24	5.0E-6	3.0E-3	1.5E-08	4.1E-11	9.0E-12
LPSI	24	4.5E-5	3.0E-3	1.4E-07	3.7E-10	7.4E-11
HPSI (Plants w/ PORVs)	4	5.1E-3	1.7E-2	8.6E-05	3.9E-08	7.8E-09
HPSI (Plants w/o PORVs)	4	3.1E-3	1.7E-2	5.2E-05	2.4E-08	4.8E-09
CS (Plants w/o CARC)	12	5.5E-4	1.7E-2	9.3E-06	1.3E-08	2.5E-09
PORV	8	1.0E-3	7.2E-2	7.3E-05	6.7E-08	1.3E-08
Boration System	24	1.7E-5	7.2E-2	1.2E-06	3.4E-09	6.7E-10
Pressurizer Heaters	24	1.0E-4	3.8E-2	3.8E-06	1.1E-08	2.3E-09
Containment	8	1.0E-4	1.0	1.0E-04	9.0E-08	1.8E-08

WITHDRAWN

The assessed ΔLERF values indicate that all proposed TS changes, for the systems and components listed in the first column, would have an insignificant risk impact with respect to large early release (significantly less than the acceptance criterion of 1E-7/year based on guidance provided in RGs 1.174 and 1.177). This conclusion is true without any credit for the averted risk associated with avoiding plant shutdown or any consideration for the many conservative assumptions used in the risk assessment.

The assessed ICLERP values are also within RG 1.177 acceptance guidelines. With the exception of the PORVs, the ICLERP is smaller than 5E-8 which is considered very small and acceptable. Based on the arguments presented in Section 3, where the "acceptance criteria" are discussed, the small ICLERP deviation from the numerical guideline of 5E-8 shown for the PORVs in Table 5 are in agreement with guidance provided in RG 1.177 and, therefore, acceptable. What is stated above with respect to ICCDP is also valid for ICLERP (i.e., the intent of the numerical guideline for ICLERP is to ensure that the acceptance criterion for ΔLERF is met). In this case, since the entry into LCO 3.0.3 is a significantly less frequent event than LCO entries on which the numerical values mentioned in RG 1.177 are based, the acceptance criterion for ΔLERF can be met with reasonable assurance even when the

numerical value of the acceptance guideline for ICLERP provided in RG 1.177 is not met. Furthermore, the RG 1.177 numerical guideline of 5E-8 would most likely be met also for PORVs if the avoided shutdown risk and the effect of the various conservative assumptions had been taken into account.

Table 6 summarizes the results of sensitivity case #1. The risk impact was re-assessed with more conservative assumptions and data which were determined to be associated with areas of potential uncertainty that could have a significant impact on results and conclusions.

Table 6 Sensitivity Case #1 Risk Impact Results (SGD=0.3 and TI-SGTR =1 and f= 1/3).

Component/System	Proposed CT (hrs)	ICCDP	ΔCDF/yr	ICLERP	ΔLERF/yr
SIT	24	1.3E-8	4.4E-9	1.3E-10	4.4E-11
LPSI	24	1.2E-7	3.9E-8	1.2E-09	4.0E-10
HPSI (Plants w/ PORVs)	4	1.7E-6	5.8E-7	1.3E-07	4.2E-08
HPSI (Plants w/o PORVs)	4	1.1E-6	3.5E-7	7.7E-08	2.6E-08
CS (Plants w/o CARC)	12	6.9E-7	2.3E-7	5.0E-08	1.7E-08
PORV	8	8.1E-7	2.7E-7	2.6E-07	8.7E-08
Boration System	24	4.5E-8	1.5E-8	1.4E-08	4.8E-09
Pressurizer Heaters	24	2.9E-7	9.6E-8	4.8E-08	1.6E-08
Containment	8	N/A	N/A	9.0E-08	3.0E-08

WITHDRAWN

Even though it is believed that the CLERP values for the base case were assessed conservatively, the probabilities of two key parameters which are dominant CLERP contributors have been increased to or close to their upper bounds. The probability of occurrence of a thermally-induced SGTR in advance of another RCS structure failure (event TI-SGTR) was increased from 50% (base case) to 100% and the probability that an MSSV will fail open depressurizing one SG (event SGD) was increased from 10% to 30%. In addition, the frequency of loss of function was increased from once every five years to once every three years. The objective of this sensitivity study was to provide additional assurance that there are no areas of uncertainty, in modeling assumptions and data, which could have a significant impact on results and conclusions.

The results of sensitivity case #1 indicate an insignificant risk impact, with respect to both core damage and large early release (the ΔCDF and the ΔLERF values for all proposed TS changes are smaller than 1E-6/yr and 1E-7/yr, respectively), even for such "bounding" assumptions. Furthermore, this conclusion is true without any credit for the averted risk associated with avoiding plant shutdown.

Table 7 summarizes the results of sensitivity case #2. Even though the base case risk impact is based on several conservative assumptions, as discussed in Section 3.1.2, the risk impact

was re-assessed using the upper bound 95th percentile value of the challenge frequency for each system or component listed in the first column of Table 7. The objective of this sensitivity study was to provide additional assurance that there are no areas of uncertainty, associated with the challenge frequency estimates, which could have a significant impact on results and conclusions. The 95th percentile values of the challenge frequency are listed in column 3 (as discussed in Section 3, the conditional core damage risk increase was conservatively assumed to be the same as the system's or component's challenge frequency).

Table 7 Sensitivity Case #2 Risk Impact Results (95% Challenge Frequency Distribution Limit).

Component/System	Proposed CT (hrs)	ΔR_{CDF}/yr	ICCDP	ΔCDF/yr	ICLERP	ΔLERF/yr
SIT	24	1.0E-5	2.6E-8	5.2E-9	8.2E-11	1.6E-11
LPSI	24	1.1E-4	2.9E-7	5.8E-8	9.0E-10	1.8E-10
HPSI (Plants w/ PORVs)	4	1.4E-2	4.8E-6	9.2E-7	1.1E-07	2.2E-08
HPSI (Plants w/o PORVs)	4	9.6E-3	3.3E-6	6.6E-7	7.4E-08	1.5E-08
CS (Plants w/o CARC)	12	1.1E-3	1.4E-6	2.8E-7	2.5E-08	5.1E-09
PORV	8	2.5E-3	2.0E-6	4.0E-7	1.6E-07	3.3E-08
Boration System	24	2.5E-5	6.8E-8	1.4E-8	5.0E-09	1.0E-09

WITHDRAWN

The results of sensitivity case #2 indicate an insignificant risk impact, with respect to both core damage and large early release (the ΔCDF and the ΔLERF values for all proposed TS changes are smaller than 1E-6/yr and 1E-7/yr, respectively), even for such "bounding" assumptions. Furthermore, this conclusion is true without any credit for the averted risk associated with avoiding plant shutdown.

Table 8 summarizes the risk impact, in terms of radiation release frequency (RRF) change, ΔRRF, for each of the proposed TS changes related to systems and components contributing to radiation release (non-LER). Availability of such equipment is typically required to meet design basis dose limits.

The first column lists the systems and components for which a TS change is proposed. The second column lists the new proposed completion time to restore the system's or component's function. The third column is the conditional radiation release (non-LER) risk increase, ΔR_{RRF}, caused by the system's or component's loss of function (conservatively assumed, except for the ECCS PREACS, to be challenged for sure during a core damage event). The fourth column lists the ICRRP values for continued plant operation at power for the entire proposed CT given loss of the system's or component's function. The ICRRP values are obtained by multiplying the proposed CT value (column 2) by ΔR_{RRF} (column 3). The last two columns list the assessed average expected RRF changes, ΔRRF, associated with the proposed CT extensions, for two different loss of function frequencies (i.e., the base case frequency of once very five years and

the more conservative case of once every three years). The ΔRRF values are obtained by multiplying the corresponding ICRRP value by the average frequency of loss of function (i.e., by 0.2 and 0.33, respectively).

Table 8 Radiation Release (Non-LER) Risk Impact.

Component/System	Proposed CT (hrs)	ΔR_{RRF}/yr, or Challenge Frequency/yr	ICRRP	ΔRRF/yr (f = 1/5)	ΔRRF/yr (f = 1/3)
Iodine Cleanup System (ICS)	24	1.0E-4	2.6E-7	5.0E-8	8.3E-8
Shield Building Exhaust Air Cleanup System (SBEACS)	24	1.0E-4	2.6E-7	5.0E-8	8.3E-8
Control Room Emergency Air Cleanup System (CREACS)	24	1.0E-4	2.6E-7	5.0E-8	8.3E-8
Control Room Emergency Air Temperature Control System (CREATCS)	24	1.0E-4	2.6E-7	5.0E-8	8.3E-8
Penetration Room Exhaust Air Cleanup System (PREACS)	24	1.0E-4	2.6E-7	5.0E-8	8.3E-8
Emergency Core Cooling System Pump Room Exhaust Air Cleanup System (ECCS PREACS)	24	4.5E-5	1.1E-7	2.0E-8	3.3E-8
Containment Spray (for plants w/ CARC)	72	1.0E-4	8.0E-7	1.6E-7	2.7E-7

WITHDRAWN

The assessed ICRRP values (fourth column) as well as the ΔRRF values for both the base and the sensitivity case (last two columns of Table 8) are within the acceptance guidelines for radiation release (non-LER) risks. Such acceptance guidelines are discussed in Section 3.1.1 where conservative acceptance criteria for radiation release risks, other than large early release risks, are defined in analogy to the criteria documented in RGs 1.174 and 1.177 for CDF and LERF (i.e.,1E-7 per year for ΔRRF and 5E-7 for ICRRP). With the exception of Containment Spray (for plants with CARC), all proposed changes meet the acceptance criteria. The small deviations from the numerical guidelines for Containment Spray (for plants with CARC) also support the proposed change, especially when credit for the averted core damage and large early release risk associated with avoiding plant shutdown is taken into consideration. No sensitivity analyses were necessary for systems and components contributing to radiation releases (non-LER) due to the consistently conservative assumptions used in the analysis in conjunction with the conservative interpretation of the risks associated with such systems and components.

The staff finds that the risk assessment results support the proposed changes. The risk increases associated with the proposed TS changes, if any, will be insignificant based on guidance provided in RGs 1.174 and 1.177. Furthermore, the sensitivity studies and the many

conservative assumptions used in the analyses provide adequate assurance about the robustness of the results used to support the proposed TS changes.

5.0 EVALUATION OF SYSTEM-SPECIFIC INTEGRATED JUSTIFICATIONS

There are two categories of proposed system-specific TS changes. The first category includes changes associated with plant conditions requiring entry into LCO 3.0.3 to extend the time for restoring the system's or component's loss of function, thus avoiding unnecessary unscheduled plant shutdowns and minimizing transition and realignment risks. The second category includes changes to TS Required Action statements to allow a MODE 4 (hot shutdown) end state, for repair purposes of two-train redundant systems that do not have implicit LCO 3.0.3 entry requirements, when the proposed extended time cannot be met. The generic risk assessment for the proposed end state changes is documented in topical report CE-NPSD-1186 (Reference 4) which has been reviewed and approved by the staff. While all proposed system-specific TS changes include changes to extend the time for restoring the system's or component's loss of function (first category changes), some proposed system-specific TS changes include changes to modify the end state (second category changes). Therefore, the integrated justifications, discussed in this section, include insights from the generic risk assessments documented in both topical reports WCAP-16125-NP (Reference 3) and CE-NPSD-1186 (Reference 4).

WITHDRAWN

Due to the nature of the plant conditions associated with the proposed TS changes (i.e., loss of a system's or component's function), the redundancy and diversity typically associated with ensuring the deterministic aspect of defense-in-depth position is not always strictly possible. In these cases defense-in-depth is considered by (1) controlling the outage time for related equipment, (2) restricting activities which may challenge the unavailable systems or functions, (3) allowing only small time intervals for plant operation at power with a system or function unavailable, (4) using, whenever possible, contingency actions to limit concurrent unavailabilities appropriately, and (5) evaluating repair activities and alternatives. Defense-in-depth is evaluated in conjunction with the generic risk assessment results which conclude that the proposed system-specific TS changes would lead to insignificant risk increases and in most cases to net risk reductions. This conclusion is a consequence of the low expected challenge frequency of the systems or functions associated with the proposed TS changes, the very short proposed exposure times to the specified plant conditions and the offsetting benefits of avoiding plant transitions.

The proposed change in shutdown mode end states will result in plants remaining within the applicability of the specific LCOs for the length of time it takes to restore the LCO conditions. Since corrective maintenance will be necessary, the 10 CFR 50.65(a)(4) requirement to assess and manage risk will apply, and should confirm that remaining in the shutdown mode that is within the applicability of the LCO is acceptable for the plant specific configuration. NRC Regulatory Guide 1.182 (Reference 9) endorses NUMARC 93-01 Section 11 guidance for implementation of 10 CFR 50.65(a)(4), and shall be followed; including the conduct of an (a)(4) reevaluation for emergent conditions.

5.1 Boration System (LCO 3.1.9)

The boration system is required to control reactivity and to ensure sufficient shutdown margin to bring the plant to cold shutdown with the most reactive control element assembly (CEA) stuck

out and without credit for xenon. This system is also intended to mitigate possible return to power scenarios following a main steam line break (MSLB) or reactor coolant pump (RCP) restart. The boration system is also needed to ensure power reduction and mitigate ATWS events.

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

Limiting Condition for Operation (LCO): Two reactor coolant system (RCS) boron injection flow paths shall be operable in MODES 1, 2, 3, and 4, with the contents of the boric acid makeup (BAMU) tanks in accordance with the LCO. Two boration paths that are to remain available are: (1) the refueling water storage tank (RWST) and its feed to the charging pumps, and (2) one or both BAMU tanks with the respective feed paths to the charging pumps. Default entry into LCO 3.0.3 when both boration paths are unavailable in MODES 1, 2, 3 and 4.

Condition Requiring Entry into Shutdown Required Action: Both boration paths inoperable. That is (1) the RWST and its feed to the charging pumps and (2) both BAMU tanks with the respective feed paths are inoperable (default entry into LCO 3.0.3 is required).

Proposed Modification to Shutdown Required Actions: (1) Increase the time available to take action to restore one boration flow path to 24 hours for the cases in which both boration paths are inoperable; (2) allow MODE 3 as the final end state for the cases in which both boration paths are inoperable to repair conditions where the boric acid source tank volume, temperature or concentration are out of limits.

WITHDRAWN

Assessment: The staff evaluated the justification of the proposed system-specific TS changes by reviewing the risk assessment results and defense-in-depth arguments for each of the two proposed changes. The risk impact and defense-in-depth arguments associated with the proposed extension of the time to initiate plant shutdown is discussed first, followed by a similar discussion of the arguments in support of allowing MODE 3 as the end state for repairing an inoperable boration system.

The risk assessment results, documented in Section 4, indicate that the proposed 24-hour completion time for restoring one boration path will not lead to a significant increase in risk and may actually decrease risk. The risk impact of the proposed completion time was assessed to be well within the acceptance criteria reported in Regulatory Guides 1.174 and 1.177. Specifically, the proposed completion time extension would lead to the following risk increases: (1) the probability of core damage when the boration system is inoperable will increase by less than 5E-8 (the acceptance guideline for ICCDP is 5E-7); (2) the CDF will increase by less than 1E-8/year (the acceptance guideline for ΔCDF is 1E-6/year); (3) the large early release probability when the boration system is inoperable will increase by about 3.4E-9 (the acceptance guideline for ICLERP is 5E-8); and (4) the LERF will increase by less than 1E-9/year (the acceptance guideline for ΔLERF is 1E-7/year). It should be noted that the assessed risk impact does not take credit for avoiding the transition to shutdown risk by allowing adequate time to restore at least one boration path. Based on operational experience, there is a good likelihood that such restoration will be successful within the proposed 24 hour period, thus avoiding the transition risk. Although the transition to shutdown risk was not specifically quantified, insights from various risk assessment studies indicate that the avoided transition risk could be larger than the assessed risk increase resulting from allowing the plant to operate at

power for 24 hours with the boration system inoperable. Thus, there are good indications that the proposed completion time extension may actually be risk neutral or result in a decrease in risk. The risk impact argument is consistent with the fact that the contribution of the boration system in mitigating accidents other than ATWS events (e.g., LOCA and SGTR accidents) is very small because the HPSI system can perform the same function. Thus, the availability of HPSI during the time the plant is allowed to operate at power with the boration system inoperable ensures that defense-in-depth is maintained for many events. Although for ATWS events the HPSI system is not an effective backup to the boration system, the challenge probability of the boration system during the proposed completion time extension is extremely small (less than 1E-7).

References 3 and 4 state that the risk importance of the boration system is low during shutdown operation and, from a shutdown margin perspective assuming a specific boron concentration, MODE 3 will usually have a greater shutdown margin than MODE 4, and either mode would have greater shutdown margin than MODE 5. These statements are based on the fact that most of the time the moderator temperature coefficient (MTC) is negative.

However, the staff does not conclude that MODE 3 is the appropriate end state when two boration paths are inoperable. For the case of a steam line break (SLB), the released steam causes a decrease in the RCS temperature. With a negative MTC, the decrease in RCS temperature results in a positive reactivity addition. If the resulting positive reactivity is greater than the negative reactivity from the inserted control rod and the borated water from the RWST, the core may return to criticality for a post-trip core. In addition, for a typical PWR, an SLB in MODE 2 results in a more severe post-trip transient than that initiated from MODE 1 because the decay heat level in MODE 1 is higher and reduces the effect of the cooldown. An SLB initiated from MODE 3 with both boration paths operable is less severe than that initiated from MODE 2 because the pressure, temperature, and steam flow through the broken steam line for MODE 3 is less limiting. However, with both boration paths inoperable, the borated water for the charging pumps during an SLB is not available. Furthermore, the consequences of the SLB in MODE 4 are not significant because of the lower initial RCS temperature and an effective decoupling of the RCS secondary system from the primary system as the reactor coolant pumps are removed from service and the decay heat removal system is started. Therefore, licensees will have to demonstrate by analysis that the SLB from MODE 3 with both boration paths inoperable is bounded by the facility's Final Safety Analysis Report (MODE 2 cases) in terms of the post-trip criticality and low departure from nucleate boiling ratio (DNBR) or the case with an SLB initiated from MODE 4.

WITHDRAWN

Finding: The requested change to increase the time available to take action to restore one boration flow path to 24 hours for the cases in which both boration paths are inoperable is acceptable. The requested change to allow MODE 3 as the end state for cases where both boration paths are inoperable and the tank contents are out of limits is not acceptable without further plant specific justification.

Tier 2 Restrictions: None.

5.2 Pressurizer Heaters (LCO 3.4.9)

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 permits RCS pressure control by using the sprays and heaters during normal operation and in response to anticipated design basis accidents. The unavailability of Class 1E pressurizer heaters covered by the TS may complicate steady state plant pressure control and, thus, increase the potential for an unplanned reactor trip.

Another function of the Class 1E pressurizer heaters is to maintain plant subcooling during post accident cooldown by natural circulation. Although the unavailability of pressurizer heaters during natural circulation cooldown will extend the time to reach the shutdown cooling system entry conditions, heat removal will be adequately established via steam generator cooling.

Plant Applicability: All WOG member plants with CE NSSS designs except St Lucie-2.

Limiting Condition for Operation (LCO): Two groups of pressurizer heaters, [capable of being powered from an emergency power supply], must be operable in MODES 1, 2 and 3.

Condition Requiring Entry into Shutdown Required Action: Two safety-related pressurizer heater groups inoperable (default entry into LCO 3.0.3 is required).

WITHDRAWN

Proposed Modification to Shutdown Required Actions: Increase the time available to take action to restore one group of safety-related heaters before entry into ISTS LCO 3.4.9 Condition C to 24 hours.

Assessment: The risk assessment results, documented in Section 4, indicate that the proposed 24-hour completion time for restoring one group of safety-related pressurizer heaters before entering ISTS LCO 3.4.9 Condition C will not lead to a significant increase in risk and may actually decrease risk. The risk impact of the proposed completion time extension was assessed to be well within the acceptance criteria reported in Regulatory Guides 1.174 and 1.177. Specifically, the proposed completion time extension would lead to the following risk increases: (1) the probability of core damage when the safety-related pressurizer heaters are inoperable will increase by about 3E-7 (the acceptance guideline for ICCDP is 5E-7); (2) the CDF will increase by about 6E-8/year (the acceptance guideline for ΔCDF is 1E-6/year); (3) the large early release probability when the safety-related pressurizer heaters are inoperable will increase by less than 1E-8 (the acceptance guideline for ICLERP is 5E-8); and (4) the LERF will increase by about 2E-9/year (the acceptance guideline for ΔLERF is 1E-7/year). Furthermore, the proposed time extension may actually be risk neutral or result in a decrease in risk if credit for avoiding the transition to shutdown risk is taken.

The risk impact argument is consistent with the following observations. TS include requirements for both groups of safety-related pressurizer heaters to have minimum heating power [and emergency power supply capability]. The safety-related pressurizer heaters have two primary functions. One function is to keep the reactor coolant in a subcooled condition with natural circulation following a loss of offsite power (LOOP) event during which the normally available station powered non-safety related heaters become unavailable. Although no credit is taken in design basis accident analyses for the pressurizer heaters, they have been included in the TS because they are needed to maintain long term subcooling during a LOOP event. However, pressurizer heaters are not required to achieve a post-trip plant cooldown since successful

cooldown can be achieved, with minimal impact on plant risk, due to the availability of reactor vessel and pressurizer vents. Consequently, the pressurizer heaters do not have a significant role in the mitigation of core damage events. A second function of the safety-related pressurizer heaters is to back up the station powered non-safety related heaters which are normally available to control reactor coolant pressure during steady state operation. The unavailability of these heaters would reduce the plant's ability to control the normal operating parameters and consequently will increase the potential of plant trip.

The presence of both safety-related and non-safety-related heaters provides considerable defense-in-depth for many transient events, except following a LOOP event. For LOOP events and without the safety-related pressurizer heaters, a natural circulation cooldown may be required. Such cooldowns can be conducted via use of reactor vessel and pressurizer vents or SG venting via the atmospheric dump valves (ADVs).

The intent of the proposed completion time extension is to extend plant operation at power when the ability to control normal plant operation is not significantly degraded. Therefore, the proposed completion time extension should not be utilized when there is reason to believe that plant pressure and level cannot be controlled within operating bounds, as is the case when both the safety and non-safety pressurizer heaters are unavailable. This restriction should be reflected in the TS bases.

WITHDRAWN

Finding: The requested change to increase the time available to take action to restore one pressurizer heater group to 24 hours for cases when both groups are inoperable is acceptable.

Tier 2 Restrictions: None.

5.3 Pressurizer PORVs and Associated Block Valves (LCO 3.4.11)

PORVs are automatically opened at a specific set pressure when the pressurizer pressure increases and automatically closed on decreasing pressure. The PORVs may be manually operated using controls installed in the control room. An electric, normally open, block valve (BV) is installed between the pressurizer and the PORV. The function of the BV is to ensure RCS integrity by isolating a leaking or stuck-open PORV to permit continued power operation. Most importantly, the BV is used to isolate a stuck open PORV and terminate the RCS depressurization and coolant inventory loss.

Plant Applicability: Calvert Cliffs 1 & 2, St Lucie 1 & 2 (block valves), Millstone 2, Palisades, and Fort Calhoun Station.

Limiting Condition for Operation (LCO): Each PORV and associated block valve shall be operable in MODES 1, 2 and 3.

Condition Requiring Entry into Shutdown Required Action: Two PORVs inoperable and not capable of being manually cycled (ISTS LCO 3.4.11 Condition E or equivalent) or two BVs inoperable (ISTS LCO 3.4.11 Condition F or equivalent). There is a variability in LCO entry requirements among WOG member plants with CE NSSS designs for conditions with both PORVs inoperable or both BVs inoperable. Typically, a plant shutdown is required if the PORVs are not isolated and one PORV is not restored within one hour (ISTS LCO 3.4.11 Condition E or

equivalent) or when the PORVs are not placed in manual control within one hour and one BV is not recovered within two hours (ISTS LCO 3.4.11 Condition F or equivalent).

Proposed Modification to Shutdown Required Actions: Revise ISTS LCO 3.4.11 Condition E (or equivalent to allow an 8-hour completion time (CT) to restore one PORV for conditions where a PORV is unable to re-close once challenged, but may be isolated. This extension would not apply to PORVs that are leaking, and that cannot be isolated by block valves, or to PORVs that are not expected to be isolable following a demand.

Revise ISTS LCO 3.4.11 Required Action F.2 to allow 8 hours to restore one BV, for conditions where the associated PORV is unable to reclose.

Assessment: The risk assessment results, documented in Section 4, indicate that the proposed 8-hour completion time for the actions required by TS (i.e., actions associated with ISTS LCO 3.4.11 Conditions E and F or equivalent) will not lead to a significant increase in risk and, actually, may decrease risk by avoiding the risk associated with the transition to shutdown. The risk impact of the proposed completion time extension, without credit for avoiding the transition to shutdown risk, was assessed to be within the acceptance criteria reported in Regulatory Guides 1.174 and 1.177. Specifically, the proposed time extension would lead to the following risk increases: (1) the probability of core damage will increase by about 8E-7, which is close to the numerical guideline of 5E-7 for ICCDP used in RG 1.177; (2) the CDF will increase by about 2E-7/year, which is significantly less than the acceptance guideline of 1E-6/year for ΔCDF; (3) the large early release probability will increase by less than 7E-8, which is close to the numerical guideline of 5E-8 for ICLERP and (as explained in Section 4) in agreement with guidance provided in RG 1.177; and (4) the LERF will increase by about 1E-8/year, which is significantly less than the acceptance guideline of 1E-7/year for ΔLERF. Furthermore, the proposed time extension may actually be risk neutral or result in a decrease in risk if credit for avoiding the transition to shutdown risk is taken.

WITHDRAWN

The risk impact argument is consistent with the following defense-in-depth argument where the impact of ISTS LCO 3.4.11 Conditions E and F on defense-in-depth is discussed. The primary purpose of this LCO is to ensure that the PORVs and the BVs are operable so the potential for a small break LOCA through the PORV pathway is minimized, or if a small LOCA were to occur through a failed open PORV, the block valve could be manually operated to isolate the path. In addition, one of the functions of the PORVs is to limit the number of pressure transients that may challenge the primary safety valves (PSVs) since the PSVs, unlike the PORVs, cannot be isolated.

When both PORVs are found inoperable (i.e., ISTS LCO 3.4.11 Condition E or equivalent), the associated BVs are manually closed, within one hour, to isolate both PORV paths. With none of the PORVs available to open, the PSVs could be challenged to provide overpressure protection. However, a challenge to the PSVs during the proposed completion time extension to restore one PORV is extremely unlikely and the PSVs are available and highly reliable (i.e., even if they are challenged, they would close properly when the pressure is reduced below their setpoint). It should be noted that overpressure protection is provided by the PSVs in the design basis analyses, without any credit for PORV opening for accident mitigation (in fact there are some plants built without PORVs). For these reasons, there is defense-in-depth against LOCA accidents through the PORV and the PSV paths as well as against overpressure accidents

during the very short time interval when ISTS LCO 3.4.11 Condition E is proposed to be allowed with the plant operating at power.

When both BVs are found inoperable (i.e., ISTS LCO 3.4.11 Condition F or equivalent), the PORVs are placed in manual control, within one hour, to ensure that they do not open automatically in the unlikely event they are challenged. Therefore, there is defense-in-depth against small LOCA accidents through the PORV paths. However, in the unlikely event of a pressure transient during the proposed completion time extension, the PSVs could be challenged to provide overpressure protection. This is the same scenario discussed above for ISTS LCO 3.4.11 Condition E. For these reasons, there is defense-in-depth against LOCA accidents through the PORV and the PSV paths as well as against overpressure accidents during the very short time interval when ISTS LCO 3.4.11 Condition F is proposed to be allowed with the plant operating at power.

The PORV paths provide an alternative means of core cooling by 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. The unavailability of feed and bleed for core cooling, the dominant contributor to risk associated with the proposed changes to LCO 3.4.11. As discussed above, such risk is very small.

WITHDRAWN

Finding: The requested changes to allow 8 hours for completing the actions required by TS (i.e., actions associated with ISTS LCO 3.4.11 Conditions E and F or equivalent) are acceptable.

Tier 2 Restrictions: None.

5.4 Safety Injection Tanks (LCO 3.5.1)

The Safety Injection Tanks (SITs) are pressurized passive injection devices whose primary safety function is to inject large quantities of borated water into the reactor vessel during the blowdown phase of a large LOCA and to provide inventory to help accomplish the refill phase that follows the blowdown phase.

Plant Applicability: Applicable to all WOG member plants with CE NSSS designs.

Limiting Condition for Operation (LCO): All SITs shall be operable during MODES 1 and 2 as well as during MODE 3 when the pressurizer pressure is above [700] psia.

Condition Requiring Entry into Shutdown Required Action: When two or more SITs are inoperable (ISTS LCO 3.5.1 Condition D), immediate entry into LCO 3.0.3 is required.

Proposed Modification to Shutdown Required Actions: Increase the time available to restore one SIT before entry into LCO 3.0.3 to 24 hours.

Assessment: The risk assessment results, documented in Section 4, indicate that the proposed 24-hour completion time for restoring one SIT before entering LCO 3.0.3 will not lead to a significant increase in risk and may actually decrease risk. The risk impact of the proposed 23-hour extension, without credit for avoiding the transition to shutdown risk, was assessed to be well within the acceptance criteria reported in Regulatory Guides 1.174 and 1.177. Specifically, the proposed time extension would lead to the following risk increases: (1) the probability of

core damage will increase by about 1E-8, which is less than the numerical guideline of 5E-7 for ICCDP; (2) the CDF will increase by about 3E-9/year, which is significantly less than the acceptance guideline of 1E-6/year for ΔCDF; (3) the large early release probability will increase by about 4E-11, which is much less than the numerical guideline of 5E-8 for ICLERP; and (4) the LERF will increase by about 9E-12/year, which is much less than the acceptance guideline of 1E-7/year for ΔLERF. Furthermore, the proposed time extension would, most likely, result in a risk reduction if credit for avoiding the transition to shutdown risk is taken.

The risk impact argument is also supported by the following defense-in-depth discussion. The SITs are needed primarily to mitigate large LOCAs. The unavailability of two or more SITs will compromise the ability of the plant to respond to a large LOCA. However, as discussed above, even if it is conservatively assumed that all large LOCAs proceed to core damage, the risk impact is negligible (much less than the risk estimated to incur during plant transition to shutdown). On the other hand, the unavailability of two or more SITs may alter the progression of some smaller break size LOCAs and the extent of core damage. However, their impact on the core damage potential is negligible. In addition, long term core cooling, provided via the plant's LPSI and HPSI systems, partially offsets the impact of SIT unavailability.

Finding: The requested change to increase the time available to take action to restore all SITs (from one to 24 hours) for cases when two or more SITs are inoperable is acceptable.

WITHDRAWN

Tier 2 Restrictions: None.

5.5 Low Pressure Safety Injection (LCO 3.5.2)

The low pressure safety injection (LPSI) system is part of the emergency core cooling system (ECCS). The function of the ECCS is to provide core cooling and negative reactivity to ensure that the reactor core is protected following certain accidents, such as LOCAs, SGTRs and loss of feedwater. There are two phases of ECCS operation: injection and recirculation. In the injection phase, borated water is injected into the RCS via the cold legs. After the blowdown stage of the LOCA stabilizes, injection flow is split equally between the hot and cold legs. After the RWST is depleted, the ECCS recirculation phase is entered as the ECCS suction is automatically transferred to the containment sump. TS require that in MODES 1, 2 and 3, with pressurizer pressure greater than or equal to [1700] psia, both redundant (100% capacity) ECCS trains must be operable. Each ECCS train consists of a high pressure safety injection (HPSI) subsystem, a low pressure safety injection (LPSI) subsystem and a charging subsystem.

Plant Applicability: Applicable to all WOG member plants with CE NSSS designs.

Limiting Condition for Operation (LCO): Two redundant, 100% capacity LPSI trains must be operable in MODES 1 and 2 as well as in MODE 3 when the pressurizer pressure is greater than or equal to [1700] psia.

Condition Requiring Entry into Shutdown Required Action: When both LPSI trains are inoperable, the design basis assumptions for the large break LOCA analyses are not met and a default entry into LCO 3.0.3 is required.

Proposed Modification to Shutdown Required Actions: Add separate condition for both LPSI trains inoperable to restore at least one LPSI train to operable in 24 hours. In addition, with the proposed condition taken with the proposed changes to HPSI discussed below, the existing condition (ISTS LCO 3.5.2 Condition D) of "Less than 100% of the ECCS flow equivalent to a single OPERABLE train available" will no longer be required since that condition will be addressed by the conditions for two HPSI subsystems inoperable or two LPSI subsystems inoperable.

Assessment: The risk assessment results, documented in Section 4, indicate that the proposed 24-hour completion time for restoring one LPSI train will not lead to a significant increase in risk and may actually decrease risk. The risk impact of the proposed completion time extension, without credit for avoiding the transition to shutdown risk, was assessed to be well within the acceptance criteria reported in Regulatory Guides 1.174 and 1.177. Specifically, the proposed completion time extension would lead to the following risk increases: (1) the probability of core damage will increase by about 1E-7, which is less than the numerical guideline of 5E-7 for ICCDP; (2) the CDF will increase by about 2E-8/year, which is significantly less than the acceptance guideline of 1E-6/year for ΔCDF; (3) the large early release probability will increase by about 4E-10, which is much less than the numerical guideline of 5E-8 for ICLERP; and (4) the LERF will increase by about 8E-11/year, which is much less than the acceptance guideline of 1E-7/year for ΔLERF. Furthermore, the proposed completion time extension would most likely, result in a risk reduction if credit for avoiding the transition to shutdown risk is taken.

WITHDRAWN

The risk impact argument is also supported by the following defense-in-depth discussion. The primary impact of the unavailability of the LPSI system will be the reduction in the capability of the plant to provide RCS inventory makeup to mitigate a large LOCA. However, the unavailability of the LPSI system will impair the ability of the plant to maneuver to shutdown cooling. Therefore, the proposed 24-hour completion time to repair one LPSI train is reasonable due to the very small incremental risk associated with the continued plant operation at power and the inadvisability of a plant shutdown without the LPSI pumps which are needed for shutdown cooling.

ISTS LCO 3.5.2 Condition D requires that for a condition where the ECCS flow is less than 100% of the ECCS flow assumed in the LOCA analysis. WCAP-16125-NP proposed to delete this condition because it would no longer be necessary based on the new conditions for two HPSI trains or two LPSI trains inoperable. The NRC staff has concluded that an adequate basis has not been provided to justify the deletion of ISTS LCO 3.5.2 Condition D. Specifically, the WOG should discuss the functions of the HPSI and LPSI systems in terms of reactivity control, RCS inventory control, RCS pressure control, and core heat removal for system operations such as safety injection and recirculation, hot leg injection and once through core cooling to mitigate the consequences of LOCAs, SLB, and SGTR events. The WOG should also discuss the safety and nonsafety related accident mitigation systems, and show that, for a condition when the ECCS flow is less than 100% of the ECCS flow equivalent to a single OPERABLE train, alternative flow injection systems and backup accident management strategies are available and effective. Licensees should also list specific compensatory measures (including a description of pertinent operating procedures, maintenance process and training programs) and contingency plans with acceptable justification for the proposed deletion of ISTS LCO 3.5.2 Condition D.

Finding: The requested change to increase the time available to restore an LPSI train to operable is acceptable. The proposed change to delete ISTS LCO 3.5.2 Condition D is not adequately justified and therefore, is not acceptable without further justification.

Tier 2 Restrictions: None.

5.6 High Pressure Safety Injection (LCO 3.5.2)

The high pressure safety injection system is part of the ECCS. The function of the ECCS is to provide core cooling and negative reactivity to ensure that the reactor core is protected following certain accidents, such as LOCAs, SGTRs and loss of feedwater. There are two phases of ECCS operation: injection and recirculation. In the injection phase, borated water is injected into the RCS via the cold legs. After the blowdown stage of the LOCA stabilizes injection flow is split equally between the hot and cold legs. After the RWST is depleted, the ECCS recirculation phase is entered as the ECCS suction is automatically transferred to the containment sump. TS require that in MODES 1, 2 and 3, with pressurizer pressure greater than or equal to [1700] psia, both redundant (100% capacity) ECCS trains must be operable. Each ECCS train consists of a high pressure safety injection subsystem, a low pressure safety injection subsystem and a charging subsystem.

WITHDRAWN

Plant Applicability: Applicable to all WOG member plants with CE NSSS designs.

Limiting Condition for Operation (LCO): In MODES 1 and 2 as well as in MODE 3 when the pressurizer pressure is greater than or equal to [1700] psia, both trains of HPSI must be operable.

Condition Requiring Entry into Shutdown Required Action: When both HPSI trains are inoperable, a default entry into LCO 3.0.3 is required.

Proposed Modification to Shutdown Required Actions: Increase the time for restoring one HPSI pump or subsystem, before initiating shutdown per LCO 3.0.3, to four hours.

Assessment: The risk assessment results, documented in Section 4, indicate that the proposed 4-hour completion time for the actions required by TS before entering LCO 3.0.3 will not lead to a significant increase in risk and, actually, may decrease risk by avoiding the risk associated with the transition to shutdown. The risk impact of the proposed completion time extension, without credit for avoiding the transition to shutdown risk, was assessed to be in agreement with the acceptance guidelines reported in Regulatory Guides 1.174 and 1.177. Specifically, the proposed completion time extension would lead to the following risk increases: (1) An ICCDP of 1.7E-6 for plants with PORVs and 1.1E-6 for plants without PORVs, which are close to the numerical guideline of 5E-7 for ICCDP used in RG 1.177; (2) a ΔCDF of 3.5E-7/year for plants with PORVs and 2.1E-7 for plants without PORVs, which are significantly less than the acceptance guideline of 1E-6/year for ΔCDF; (3) an ICLERP of about 4E-8 for plants with PORVs and less than 3E-8 for plants without PORVs, which are less than the numerical guideline of 5E-8 for ICLERP; and (4) a ΔLERF of about 8E-9/year for plants with PORVs and about 5E-9 for plants without PORVs, which are much less than the acceptance guideline of 1E-7/year for ΔLERF. Furthermore, the proposed time extension may actually be risk neutral or result in a decrease in risk if credit for avoiding the transition to shutdown risk is taken.

The risk impact argument is also supported by the following defense-in-depth discussion. The subject LCO requires the operability of a number of independent subsystems. In many instances due to the redundancy of trains and the diversity of subsystems, the inoperability of one component in a train does not necessarily render the HPSI incapable of performing its function. Neither does the inoperability of two different components, each in a different train, necessarily result in a loss of function for the ECCS. Examples of typical inoperabilities would include the unavailability of a single header injection valve or degradation of HPSI delivery curves below minimum design basis levels. The proposed completion time extension allows for potential resolution of minor HPSI system inoperabilities and provides time to prepare for a controlled plant shutdown without increasing the plant's risk significantly.

Finding: The requested change to allow four hours to resolve the inoperability and restore one pump or subsystem of HPSI capability before required to commence a plant shutdown per LCO 3.0.3, is acceptable.

Tier 2 Restrictions: None.

5.7 Containment (LCO 3.6.1)

WITHDRAWN

The requirements stated in this LCO define the performance of the containment as a fission barrier. Specifically, LCO 3.6.1 requires that the containment maximum leakage rate be limited in accordance with 10CFR50 Appendix J. Other LCOs place additional restrictions on containment air locks and containment isolation valves. The integrated effect of these TSs is to ensure that the containment leakage is well controlled within limits which assure that the post accident whole body and thyroid dose limits of 10 CFR 100.11 or 10 CFR 50.67, as applicable, are satisfied following a Maximum Hypothetical Accident (MHA) initiated from full power. Inability to meet this leakage limit renders the containment inoperable.

Plant Applicability: Applicable to all WOG member plants with CE NSSS designs.

Limiting Condition for Operation (LCO): Containment shall be operable in MODES 1, 2, 3 and 4.

Condition Requiring Entry into Shutdown Required Action: Containment is declared to be inoperable due to excessive leakage, including leakage from air locks and isolation valves, for a time period greater than one hour. If the containment is not restored to operable status within one hour, a plant shutdown is required.

Proposed Modification to Shutdown Required Actions: Define a specific action to allow 8 hours to restore an inoperable containment to operable. Allow MODE 4 to become a designated end state for correcting containment impairments for conditions where the containment leakage is excessive due to reasons other than the inoperability of two or more containment isolation valves (CIVs) in the same flow paths.

Assessment: The risk assessment results, documented in Section 4, indicate that the proposed 8-hour completion time for restoring an inoperable containment to operable status will not lead to a significant increase in risk and may actually decrease risk. The risk impact of the proposed completion time extension was assessed to be well within the acceptance criteria reported in Regulatory Guides 1.174 and 1.177. Specifically, the proposed time extension would lead to the following conservatively assessed risk increases: (1) the large early release probability will

increase by about 9E-8, which is close to the numerical guideline of 5E-8 for ICLERP; and (2) the LERF will increase by about 2E-8/year, which is significantly less than the acceptance guideline of 1E-7/year for ΔLERF. Furthermore, the proposed completion time extension may actually be risk neutral or result in a decrease in risk if credit for avoiding the transition to shutdown risk is taken.

The proposed changes apply to containment conditions where containment integrity is essentially maintained and adequate ECCS net positive suction head (NPSH) is expected following an event. Containment "leakage" at or near design basis levels is not explicitly modeled in PRAs. The PRA implicitly requires that containment "gross" integrity must be available to ensure adequate NPSH for ECCS pumps. Even though the PRA models do not consider that containment "leakage" contributes to a large early release, the assessed risk impact of the proposed completion time extension is based on the assumption that all core damage events will proceed to a large early release.

The requirement for an immediate (within one hour) shutdown is based on the philosophy that inoperability of the containment is a violation of the plant design basis and, therefore, a plant shutdown must be initiated as soon as possible. The selection of one hour was based on the requirement for "immediate shutdown" and the assumption that one hour is adequate time for operators to effect shutdown plans. The goal was to place the plant in a condition where the health and safety of the public could be better assured. No specific risk assessments were performed. In fact it is more appropriate from the health and safety objective viewpoint to consider the risk of continued plant operation as well as that introduced by the shutdown. In consideration of the total plant risk, it is more risk beneficial to allow a small increase in risk at power to resolve a TS inoperability rather than to undertake an immediate (within one hour) shutdown.

WITHDRAWN

In addition to the completion time extension, it is also proposed that MODE 4 be allowed as the end state to repair the containment. This is supported by the following arguments. If accidents were to occur in MODE 4, resulting containment pressures would be significantly less than the design basis accident (DBA) conditions. Hence, leakage would be further reduced. While in MODE 4, the probability of LOCA or MSLB is significantly reduced from MODE 1 levels. The implied licensing basis assumption that MODE 5 is inherently of lower operational risk than MODE 4 is not supported by risk evaluations (Reference 4). 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. Thus, remaining in MODE 4, while the containment excess leakage condition is being corrected, is an appropriate action.

The ISTS LCO 3.6.1 requirement that the plant be brought to MODE 5 end state is not based on consideration of risks. Accidents initiated from MODE 4 are far less challenging to the containment than those initiated from MODE 1. The lower energy content in MODE 4 results in containment pressures and potential leakage approximately one half of that associated with MODE 1 releases. Furthermore, by having the plant in a shutdown condition in advance, fission product releases are significantly reduced. Thus, while leakage restrictions should be maintained, MODE 4 leakage in excess of that allowed in MODE 1 can be safely allowed for a limited time sufficient to resolve the inoperability and return the plant to power operation.

From a deterministic perspective, MODE 4 with SG heat removal would maintain more mitigating systems available, as compared to MODE 5, to respond to loss of RCS inventory or decay heat removal events and therefore reduce the overall public risk. In MODE 4, the Safety Injection Actuation Signal (SIAS) and the Containment Isolation Actuation Signal (CIAS) will be available to aid the operators in responding to events that threaten the reactor or containment integrity. Therefore, the proposed TS end state change does not adversely affect the plant defense-in-depth.

Finding: The requested changes to (1) increase the time available to take action to restore the containment to 8 hours and (2) allow MODE 4 as the repair end state, are acceptable.

Tier 2 Restrictions: None.

5.8 Containment Spray System (LCO 3.6.6 A)

The containment spray (CS) and containment cooling (CC) systems provide containment atmosphere cooling to limit post accident pressure and temperature in containment to less than the design values. For most CE NSSS design plants the containment sprays represent a portion of a diverse and redundant heat removal system. In addition to containment heat removal, CSs enhance post-accident fission product removal.

WITHDRAWN

Plant Applicability: Applicable to all WOG member plants with CE NSSS designs.

Limiting Condition for Operation (LCO): Two containment spray trains and two containment cooling (CAC or CARC) trains shall be operable in MODES 1, 2, 3 and [4].

Condition Requiring Entry into Shutdown Required Action: Inoperability of both CS trains or any combination of three or more trains inoperable (ISTS LCO 3.6.6.A Condition F), immediate entry into LCO 3.0.3 is required.

Proposed Modification to Shutdown Required Actions: (1) increase the time available for restoring one CS train to 72 hours when at least one CARC train is available for containment heat removal; (2) increase the time available for restoring one CS train to 12 hours when two trains of the CARC system is unavailable for containment heat removal. Based on Table 5.2.3-2 of WCAP-16125-NP, ISTS LCO 3.6.6.A would be revised to allow shutdown modes of MODE 3 in 6 hours and MODE 5 in 36 hours versus the current requirement of immediate entry into LCO 3.0.3 if the Required Action and associated Completion Time not met.

Assessment: The risk assessment results, documented in Section 4, indicate that the proposed 12-hour completion time for restoring one CS train when two trains of the CARC system is unavailable for containment heat removal before entering LCO 3.0.3 will not lead to a significant increase in risk and may actually decrease risk. The risk impact of the proposed completion time extension was assessed to be well within the acceptance criteria reported in Regulatory Guides 1.174 and 1.177. Specifically, the proposed completion time extension would lead to the following risk increases: (1) the probability of core damage will increase by less than 7E-7 which is close to the numerical guideline of 5E-7 for ICCDP used in RG 1.177; (2) the CDF will increase by about 1.4E-7/year (acceptance criteria for ΔCDF about 1E-6/year); (3) the large early release probability during the condition will increase by about 1E-8 (acceptance criteria for

ICLERP is 5E-8); and (4) the LERF will increase by about 2.5E-9/year (acceptance criteria for ΔLERF is 1E-7/year). Furthermore, the proposed completion time extension may actually be risk neutral or result in a decrease in risk if credit for avoiding the transition to shutdown risk is taken.

When at least one CARC train is available for containment heat removal, the risk impact in terms of CDF and LERF is insignificant. However, credit is taken for post accident fission product removal by the CS system. The radiation release "non-LER" risk impact associated with the proposed increase of the time available for restoring one CS train to 72 hours was conservatively assessed, as discussed in Section 4. Specifically, the proposed completion time extension would lead to the following "non-LER" risk increases: (1) the probability of a "non-LER" release during the completion time extension would increase by about 8E-7; and (2) the "non-LER" frequency would increase by 1.6E-7/year. These increases in "non-LER" risk are slightly above the values used in the criteria discussed in Section 3.1.1 of this report. However, such increases in "non-LER" risk are still comparable in magnitude to what is considered acceptable for increases in the much higher consequence risks associated with core damage and large early release. Furthermore, the proposed completion time extension is definitely risk beneficial when the averted core damage and large early release risks associated with avoiding plant shutdown are taken into consideration.

In addition to the risk argument, the proposed 72-hour completion time is selected for compatibility with improved standard technical specification (ISTS) LCO 3.6.6B. ISTS LCO 3.6.6B calls for a Completion Time of 72 hours when two CS trains are inoperable (Condition C) and is applicable to conditions where the sprays are not credited for fission product removal.

WITHDRAWN

Inoperability of the CS or CARC will degrade the capability of the plant to respond to a containment threat. However, provided the other system is available the plant remains capable of controlling pressure. The loss of sprays will expose some plant equipment to beyond environmental qualification temperature limits should a MSLB occur. However, the probability of such an event during the proposed completion time extension is very small (about 1E-3/year or less than 1E-5 per 71 hours). Furthermore, the ability of the plant to cope with a MSLB event is not compromised.

Finding: The requested changes to (1) increase the time available for restoring one CS train to 72 hours when at least one CARC train is available for containment heat removal; and (2) increase the time available for restoring one CS train to 12 hours when two trains of the CARC system is unavailable for containment heat removal, are acceptable. The requested change described in Table 5.2.3-2 of WCAP-16125-NP, that is, ISTS LCO 3.6.6.A would be revised to allow shutdown modes of MODE 3 in 6 hours and MODE 5 in 36 hours versus the current requirement of immediate entry into LCO 3.0.3 if the Required Action and associated Completion Time is not met, was not justified in the topical report. Therefore, the proposed change is not acceptable without further justification.

Tier 2 Restrictions: None.

5.9 Iodine Cleanup System (ICS) (LCO 3.6.10)

The purpose of the ICS is to remove elemental iodine from the post-accident containment atmosphere. These systems were initially incorporated into plants in the belief that radiological

iodine releases would be predominantly in elemental form. However, extensive research has indicated that most iodine will be released in the form of Cesium Iodine (CsI) particulates. Consequently, the actual impact of system functionality on actual public doses is negligible. ICS consists of two 100% capacity trains.

Plant Applicability: Calvert Cliffs 1 & 2, St Lucie 1 & 2.

Limiting Condition for Operation (LCO): Two ICS trains shall be operable in MODES 1, 2, 3 and 4.

Condition Requiring Entry into LCO 3.0.3: Both ICS trains inoperable. Currently a default entry into LCO 3.0.3 is required.

Proposed Modification to Shutdown Required Actions: Add a condition to (1) allow 24 hours to restore one train to operable status, and (2) allow MODE 4 as the final end state for repairing the inoperable system.

Assessment: The risk assessment results, documented in Section 4, indicate that the proposed 24-hour completion time for restoring one train of ICS will not lead to a significant increase in risk and may actually decrease risk. The proposed completion time extension will not contribute to any risk increases, in terms of core damage and large early release. The radiation release "non-LER" risk impact associated with the proposed time increase was conservatively assessed, as discussed in Section 4. Specifically, the proposed completion time extension would lead to the following "non-LER" risk increases: (1) the probability of a "non-LER" release during the completion time extension would increase by about 2.6E-7; and (2) the "non-LER" frequency would increase by about 5.0E-8/year. These increases in "non-LER" risk, which are comparable in magnitude to what is considered acceptable for core damage and large early release risk increases, are very small. Furthermore, the proposed completion time extension is risk beneficial when the averted core damage and large early release risks associated with avoiding plant shutdown are taken into consideration.

WITHDRAWN

The proposed change to allow MODE 4 as the final end state for repairing the inoperable system is supported by risk assessments (Reference 4) which indicated that, in general, there is less risk associated with staying in MODE 4 to repair the inoperable system than proceeding to MODE 5. This is due to the fact that there are more systems available in MODE 4 than in MODE 5 to mitigate accidents initiated at shutdown and the risk of transition between MODES 4 and 5 is avoided.

The ICS functions together with the containment spray and the containment cooling systems following a design basis accident (DBA) that causes failure of the fuel cladding, and release of radioactive material (principally iodine) to the containment. The ICS is specifically designed to respond to the maximum hypothetical accident with a large assumed contribution due to elemental iodine. The DBAs that result in a release of radioactive iodine within containment are LOCA and MSLB or a control element assembly (CEA) ejection accident. In the analysis for each of these accidents, it is assumed that adequate containment leak tightness is present at event initiation to limit potential leakage to the environment. Additionally, it is assumed that the amount of radioactive iodine release is limited by reducing the iodine concentration in the containment atmosphere via use of containment sprays. The unavailability of the ICS will have no significant impact on anticipated radiological releases to the public or the control room. This

is due to the fact that: (1) iodine releases are predominantly particulate and removal via sprays and precipitation is effective, (2) availability of elemental iodine is low so that ICS has limited utility, and (3) containment leak tightness significantly limits potential releases. Significant release events that contribute to large early release, such as containment bypass and SGTR with loss of secondary isolation events, will bypass these filters regardless of their availability.

Finding: The requested changes to (1) increase the time available to restore one ICS train to 24 hours and (2) allow MODE 4 as the final end state, for cases when both ICS trains are inoperable, are acceptable.

Tier 2 Restrictions: None.

5.10 Shield Building Exhaust Air Cleanup System (LCO 3.6.13)

The shield building exhaust air cleanup system (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 absorber section for removal of radionuclides and a fan. Ductwork, valves and/or dampers and instrumentation also form part of the system.

WITHDRAWN

Plant Applicability: St Lucie 1 & 2, Waterford 3 and Millstone 2.

Limiting Condition for Operation (LCO): Two SBEACS trains shall be operable in MODES 1, 2, 3 and 4.

Condition Requiring Entry into Shutdown Required Action: Both SBEACS trains inoperable. Currently a default entry into LCO 3.0.3 is required.

Proposed Modification to Shutdown Required Actions: Add a condition to (1) allow 24 hours to take action for both SBEACS trains unavailable, and (2) allow MODE 4 as the final end state for repairing the inoperable system.

Assessment: The risk assessment results, documented in Section 4, indicate that the proposed 24-hour completion time for restoring one train of SBEACS will not lead to a significant increase in risk and may actually decrease risk. The proposed completion time extension will not contribute to any risk increases, in terms of core damage and large early release. The radiation release "non-LER" risk impact associated with the proposed time increase was conservatively assessed, as discussed in Section 4. Specifically, the proposed completion time extension would lead to the following "non-LER" risk increases: (1) the probability of a "non-LER" release during the completion time extension would increase by about 2.6E-7; and (2) the "non-LER" frequency would increase by about 5.0E-8/year. These increases in "non-LER" risk, which are comparable in magnitude to what is considered acceptable for core damage and large early release risk increases, are very small. Furthermore, the proposed completion time extension is definitely risk beneficial when the averted core damage and large early release risks associated with avoiding plant shutdown are taken into consideration.

The proposed change to allow MODE 4 as the final end state for repairing the inoperable system is supported by risk assessments (Reference 4) which indicated that, in general, there is less

risk associated with staying in MODE 4 to repair the inoperable system than proceeding to MODE 5. This is due to the fact that there are more systems available in MODE 4 than in MODE 5 to mitigate accidents initiated at shutdown and the risk of transition between MODES 4 and 5 is avoided.

The proposed changes are also supported by the following qualitative discussion. The SBEACS is required to ensure that the radioactive material leaking from the primary containment of a dual containment into the Shield Building (secondary containment) following a DBA are filtered and absorbed prior to exhausting to the environment. Loss of the SBEACS could cause site boundary doses, in the event of a DBA, to exceed the values given in the licensing basis. However, containment "leakage" at or near design basis levels is not explicitly modeled in PRAs. PRAs implicitly require that containment "gross" integrity must be available to ensure NPSH for ECCS pumps. In the PRA Level 2 models, containment "leakage" is not considered to contribute to large early release. If accidents were to occur in MODE 4, resulting containment pressures would be significantly less than the DBA conditions. Hence, leakage would be further reduced. In addition, while in MODE 4, the probability of LOCA and MSLB is significantly reduced from MODE 1 levels. By keeping the plant in MODE 4, operator actions required for entry into shutdown cooling and which introduce potential containment bypass risks are avoided.

Finding: The requested changes to (1) increase the time available to restore one SBEACS train to 24 hours and (2) allow MODE 4 as the final end state, for cases when both SBEACS trains are inoperable, are acceptable.

WITHDRAWN

Tier 2 Restrictions: None.

5.11 Control Room Emergency Air Cleanup System (LCO 3.7.11)

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 CREACUS, CREACS, CREVAS, CREVS, or CREAFS. The current TS require operability of CREACS from MODE 1 through MODE 4 to support operator response to a DBA. The system's operability in MODES 5 and 6 may also be required at some plants for chemical and toxic gas concerns. The CREACS is needed to protect the control room (CR) in a wide variety of circumstances.

Plant Applicability: Applicable to all WOG member plants with CE NSSS designs.

Limiting Condition for Operation (LCO): Two CREACS trains shall be operable in MODES 1, 2, 3 and 4 and during movement of [recently] irradiated fuel assemblies in MODES [5 and 6].

Condition Requiring Entry into Shutdown Required Action: Both trains inoperable for conditions other than inoperable control room boundary in MODES 1, 2, 3, and 4. Explicit entry into LCO 3.0.3 required (ISTS LCO 3.7.11 Condition F).

Proposed Modification to Shutdown Required Actions: (1) Increase the time available to take action to 24 hours (or the time to reach 5 REM, which may be less than 24 hours, from the radiation field associated with main steam safety valves lifting concurrent with a SGTR) for the cases in which both CREACS trains are unavailable, and (2) allow MODE 4 as the final end state for repairing the inoperable system. This modification applies to the radiation protection function

only. Site specific validation is necessary to support extension to toxic gas and chemical protection functions.

Assessment: The risk assessment results, documented in Section 4, indicate that the proposed 24-hour completion time for restoring one train of CREACS before entering LCO 3.0.3 will not lead to a significant increase in risk and may actually decrease risk. The proposed completion time extension will not contribute to any risk increases, in terms of core damage and large early release. The radiation release "non-LER" risk impact associated with the proposed time increase was conservatively assessed, as discussed in Section 4. Specifically, the proposed completion time extension would lead to the following "non-LER" risk increases: (1) the probability of a "non-LER" release during the completion time extension would increase by about 2.6E-7; and (2) the "non-LER" frequency would increase by about 5.0E-8/year. These increases in "non-LER" risk, which are comparable in magnitude to what is considered acceptable for core damage and large early release risk increases, are very small. Furthermore, the proposed completion time extension is definitely risk beneficial when the averted core damage and large early release risks associated with avoiding plant shutdown are taken into consideration.

The proposed change to allow MODE 4 as the final end state for repairing the inoperable system is not justified. ISTS LCO 3.7.11 Condition F has an explicit LCO 3.0.3 entry. WCAP-16125-NP does not provide justification for modifying Condition F Required Action from "Enter LCO 3.0.3" to an end state of MODE 4.

WITHDRAWN

Finding: The requested change to increase the time available to take action to restore one CREACS train to 24 hours for the radiation protection function only is acceptable. The requested change to allow MODE 4 as the final end state, for cases when both CREACS trains are inoperable, is not justified in WCAP-16125-NP and is not acceptable.

Tier 2 Restrictions: None.

5.12 Control Room Emergency Air Temperature Control System (LCO 3.7.12)

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

Plant Applicability: Applicable to Calvert Cliffs 1 & 2, Fort Calhoun, Palisades, PVNGS 1, 2, & 3, Waterford 3 and ANO 2. It is noted that cooling for the St Lucie units are included in the air cleanup system discussed in TS 3.7.11 but the cooling system arguments contained in this section apply to St Lucie Units 1 & 2).

Limiting Condition for Operation (LCO): Two CREATCS trains shall be operable in MODES 1, 2, 3 and 4 and during movement of [recently] irradiated fuel assemblies in MODES [5 and 6].

Condition Requiring Entry into Shutdown Required Action: Both trains inoperable in MODES 1, 2, 3, and 4 requires an explicit LCO 3.0.3 entry (ISTS LCO 3.7.12 Condition E).

Proposed Modification to Shutdown Required Actions: Modify ISTS LCO 3.7.12 Condition E to (1) increase the time available to take action under LCO 3.0.3 to 24 hours for the cases in which

both CREATCS trains are unavailable, and (2) allow MODE 4 as the final end state for repairing the inoperable system.

Assessment: The risk assessment results, documented in Section 4, indicate that the proposed 24-hour completion time for restoring one train of CREATCS before entering LCO 3.0.3 will not lead to a significant increase in risk and may actually decrease risk. The proposed completion time extension will not contribute to any risk increases, in terms of core damage and large early release. The radiation release "non-LER" risk impact associated with the proposed completion time increase was conservatively assessed, as discussed in Section 4. Specifically, the proposed completion time extension would lead to the following "non-LER" risk increases: (1) the probability of a "non-LER" release during the completion time extension would increase by about 2.6E-7; and (2) the "non-LER" frequency would increase by about 5.0E-8/year. These increases in "non-LER" risk, which are comparable in magnitude to what is considered acceptable for core damage and large early release risk increases, are very small. Furthermore, the proposed completion time extension is definitely risk beneficial when the averted core damage and large early release risks associated with avoiding plant shutdown are taken into consideration.

The proposed change to allow MODE 4 as the final end state for repairing the inoperable system is not justified. ISTS LCO 3.7.12 Condition E has an explicit LCO 3.0.3 entry. WCAP-16125-NP does not provide justification for modifying Condition E Required Action from "Enter LCO 3.0.3" to an end state of MODE 4.

WITHDRAWN

Several short term actions associated with cooling the CR may be implemented to mitigate risk consequences further. These actions include use of portable fans and propping open doors. Several plants have such actions in procedures.

Finding: The requested change to increase the time available to take action to restore one CREATCS train to 24 hours is acceptable. The requested change to allow MODE 4 as the final end state, for cases when both trains are inoperable, is not justified in WCAP-16125-NP and is not acceptable.

Tier 2 Restrictions: None.

5.13 Emergency Core Cooling System (ECCS) Pump Room Exhaust Air Cleanup System (PREACS) (LCO 3.7.13)

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

Plant Applicability: Calvert Cliffs 1 & 2, St Lucie 1 & 2, Waterford 3. It is noted that at Waterford 3 the functions of the ECCS PREACS and Penetration Room Exhaust Air Cleanup System (PREACS), which is discussed below under LCO 3.7.15, are combined within the Controlled Ventilation Area System (CVAS) TS.

Limiting Condition for Operation (LCO): Two ECCS PREACS trains shall be operable in MODES 1, 2, 3 and 4.

Condition Requiring Entry into Shutdown Required Action: Both trains inoperable, default entry into LCO 3.0.3.

Proposed Modification to Shutdown Required Actions: (1) Increase the time available to restore one train to 24 hours, and (2) allow MODE 4 as the final end state for repairing the inoperable system.

Assessment: The risk assessment results, documented in Section 4, indicate that the proposed 24-hour completion time for restoring one train of ECCS PREACS will not lead to a significant increase in risk and may actually decrease risk. The proposed completion time extension will not contribute to any risk increases, in terms of core damage and large early release. The radiation release "non-LER" risk impact associated with the proposed completion time increase was conservatively assessed, as discussed in Section 4. Specifically, the proposed completion time extension would lead to the following "non-LER" risk increases: (1) the probability of a "non-LER" release during the completion time extension would increase by about 1.1E-7; and (2) the "non-LER" frequency would increase by about 2.0E-8/year. These increases in "non-LER" risk, which are comparable in magnitude to what is considered acceptable for core damage and large early release risk increases, are very small. Furthermore, the proposed completion time extension is definitely risk beneficial when the averted core damage and large early release risks associated with avoiding plant shutdown are taken into consideration.

WITHDRAWN

The proposed change to allow MODE 4 as the final end state for repairing the inoperable system is supported by risk assessments (Reference 4) which indicated that, in general, there is less risk associated with staying in MODE 4 to repair the inoperable system than proceeding to MODE 5. This is due to the fact that there are more systems available in MODE 4 than in MODE 5 to mitigate accidents initiated at shutdown and the risk of transition between MODES 4 and 5 is avoided.

The unavailability of the ECCS PREACS only impacts radiation releases to the public when the ECCS recirculation is in progress during a LOCA. Since successful recirculation also implies successful event mitigation, the releases this system is designed to mitigate are relatively low.

Finding: The requested changes to (1) increase the time available to take action to restore one ECCS-PREACS train to 24 hours and (2) allow MODE 4 as the final end state, for cases when both trains are inoperable, are acceptable.

Tier 2 Restrictions: None.

5.14 Penetration Room Exhaust Air Cleanup System (PREACS) (LCO 3.7.15)

The Penetration Room Exhaust Air Cleanup System (PREACS) filters air from the penetration area between the containment and the auxiliary building. The PREACS consists of two independent, redundant trains. Each train consists of a heater, demister or prefilter, HEPA filter, activated charcoal absorber and a fan.

Plant Applicability: Calvert Cliffs 1 & 2, and Waterford 3. It is noted that at Waterford 3 the functions of the PREACS and ECCS PREACS, which is discussed above under LCO 3.7.13, are combined within the Controlled Ventilation Area System (CVAS) TS.

Limiting Condition for Operation (LCO): Two PREACS trains shall be operable in MODES 1, 2, 3 and 4.

Condition Requiring Entry into Shutdown Required Action: Both trains inoperable for reasons other than an inoperable penetration room boundary, default entry into LCO 3.0.3 is required.

Proposed Modification to Shutdown Required Actions: (1) Increase the time available to restore one train to 24 hours, and (2) allow MODE 4 as the final end state for repairing the inoperable system.

Assessment: The risk assessment results, documented in Section 4, indicate that the proposed 24-hour completion time for restoring one train of PREACS will not lead to a significant increase in risk and may actually decrease risk. The proposed completion time extension will not contribute to any risk increases, in terms of core damage and large early release. The radiation release "non-LER" risk impact associated with the proposed completion time increase was conservatively assessed, as discussed in Section 4. Specifically, the proposed completion time extension would lead to the following "non-LER" risk increases: (1) the probability of a "non-LER" release during the completion time extension would increase by about 2.6E-7; and (2) the "non-LER" frequency would increase by about 5.0E-8/year. These increases in "non-LER" risk, which are comparable in magnitude to what is considered acceptable for core damage and large early release risk increases, are very small. Furthermore, the proposed completion time extension is definitely risk beneficial when the averted core damage and large early release risks associated with avoiding plant shutdown are taken into consideration.

WITHDRAWN

The proposed change to allow MODE 4 as the final end state for repairing the inoperable system is supported by risk assessments (Reference 4) which indicated that, in general, there is less risk associated with staying in MODE 4 to repair the inoperable system than proceeding to MODE 5. This is due to the fact that there are more systems available in MODE 4 than in MODE 5 to mitigate accidents initiated at shutdown and the risk of transition between MODES 4 and 5 is avoided.

Finding: The requested changes to (1) increase the time available to take action to restore one PREACS train to 24 hours and (2) allow MODE 4 as the final end state, for cases when both trains are inoperable, are acceptable.

Tier 2 Restrictions: None.

6.0 SUMMARY AND CONCLUSIONS

The required action for conditions that imply a loss of function, is entry into LCO 3.0.3. Currently, upon entering LCO 3.0.3, one hour is allowed to prepare for an orderly shutdown before initiating a change in plant operation. The WOG is proposing to define or modify various TS Conditions to accommodate extension of the currently required time of one hour to initiate plant shutdown for member plants with CE NSSS designs. The proposed extension, related to specific systems or components, is based on the system's risk significance. In addition, WCAP-16125-NP provides a proposal to modify several Required Action statements, related to specific systems or components, to allow for a MODE 4 (hot shutdown) end state for repair purposes of two-train redundant systems that do not have explicit LCO 3.0.3 entry requirements, when the time requirements of the action statement for staying at power cannot be met.

The intent of the proposed TS changes is to provide needed flexibility in the performance of corrective maintenance during power operation to fully evaluate the situation or restore loss of function and at the same time enhance overall plant safety by:

- avoiding unnecessary unscheduled plant shutdowns,
- minimizing plant transitions and associated transition and realignment risks,
- providing increased flexibility in scheduling and performing maintenance and surveillance activities, and
- providing explicit guidance in areas that currently does not exist.

It should be noted that many of the proposed TS changes affect the existing plant shutdown requirements for plant conditions where the plant operation is not in explicit compliance with the plant design basis. The proposed actions provide a risk-informed process for establishing shutdown priorities aiming at reducing overall plant risk and increasing public health and safety protection. In performing the risk-informed assessments and interpreting the results, the following assumptions were made:

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- A condition resulting in the inoperability of a system or component which currently results in the need for an immediate shutdown is a low frequency event.
- The frequency of events leading to LCO 3.0.3 is not expected to increase significantly following the proposed change because such events may be reportable and may require a licensee event report. In addition, events leading to LCO 3.0.3 are used in performance indicators and the reactor oversight program. Therefore, licensees will have no incentive to allow the current low frequency of these events to increase after the proposed extensions are granted.
- The risk incurred by increasing the required shutdown action time is controlled to acceptable levels using a risk informed approach that considers the component risk worth and offsetting benefits of avoiding plant transitions.

The risk impact of the proposed TS changes was assessed following the three-tiered approach recommended in RG 1.177 for evaluating proposed extensions in currently allowed Completion Times (CTs):

- The first tier involves the assessment of the change in plant risk due to the proposed TS change;
- The second tier involves the identification of potentially high-risk configurations that could exist if equipment in addition to that associated with the change were to be taken out of service simultaneously;
- The third tier involves the implementation of the proposed changes in conjunction with a configuration risk management program (CRMP).

The impact of each proposed system-specific TS change on defense-in-depth was evaluated in conjunction with the risk assessment results. Due to the nature of the plant conditions associated with the proposed TS changes (i.e., loss of a system's or component's function), the redundancy and diversity typically associated with ensuring the deterministic aspect of defense-in-depth position is not always strictly possible. In these cases defense-in-depth was considered by identifying specific restrictions to the implementation of the proposed changes. Such restrictions aim at (1) controlling the outage time for related equipment, (2) restricting activities which may challenge the unavailable systems or functions, (3) allowing only small time intervals for plant operation at power with a system or function unavailable, (4) using, whenever possible, contingency actions to limit concurrent outages, and (5) evaluating repair activities and alternatives.

Based on this integrated evaluation, the staff concludes that the proposed system-specific TS changes would at most lead to acceptably small risk increases. In addition, defense-in-depth is taken into consideration. This conclusion is a consequence of the low expected challenge frequency of the systems or functions associated with the proposed TS changes, the very short proposed exposure times to the specified plant conditions, the offsetting benefits of avoiding plant transitions, and the identification of specific restrictions to the implementation of the proposed changes.

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The staff approval applies only to operation as described in the WOG requests documented in Reference 3 and acceptably justified in References 3 and 4. To be consistent with the staff's approval, licensees interested in implementing these changes should commit to operate in accordance with the following stipulations:

1. Implementation guidance will be developed for the proposed change. The guidance will discuss for the applicable TS changes the safety and non-safety related accident mitigation systems, and show that alternative systems and backup accident management strategies are available and effective. It will also provide specific compensatory measures (including a description of pertinent operating procedures, maintenance processes and training programs) and contingency plans with adequate justification for the proposed action times. In addition, this guidance will identify any restrictions or requirements that may be needed for the safe implementation of the proposed TS changes. Examples of restrictions are listed in Appendix A.

2. Licensees should implement the proposed changes, which include increases in required shutdown Completion Times as well as changes in shutdown action statement end states, in accordance with an overall configuration risk management program (CRMP) to ensure that potentially risk-significant configurations resulting from maintenance and other operational activities are identified and avoided. This includes the conduct of an 10 CFR 50.65(a)(4) reevaluation for emergent conditions. This objective is met by licensee programs to comply with the 10 CFR 50.65(a)(4) requirement to assess and manage risk resulting from maintenance and other operational activities.

3. Entry into shutdown MODE 4 shall be for the primary purpose of accomplishing short-duration repairs which necessitated exiting the original operating mode. The requested changes in end state for repair purposes do not prohibit licensees from entering cold shutdown if they wish to do so for operational reasons or maintenance requirements.

4. The following end state mode changes were found to be not acceptable without further plant specific justification. Licensees must provide plant specific justification for the adoption of these proposed end states.

   a. LCO 3.1.9, Boration Systems: MODE 3 as the final end state for the cases in which both boration paths are inoperable and the borated water storage tank contents are out of the specified limits.

   b. LCO 3.6.6.A, Containment Spray System: MODE 3 in 6 hours and MODE 5 in 36 hours versus the current requirement of immediate entry into LCO 3.0.3 if the Required Action and associated Completion Time not met.

   c. LCO 3.7.11, Control Room Emergency Air Cleanup System: MODE 4 as the final end state for the radiation protection function only versus the current requirement of immediate entry into LCO 3.0.3 for two CREACS trains inoperable for reasons other than an inoperable control room boundary.

   d. LCO 3.7.12, Control Room Emergency Air Temperature Control System: MODE 4 as the final end state versus the current requirement of immediate entry into LCO 3.0.3 for two CREATCS inoperable in MODES 1, 2, 3, and 4.

WITHDRAWN

5. WCAP-16125-NP proposed to delete ISTS LCO 3.5.2 Condition D, "Less than 100% of the ECCS flow equivalent to a single OPERABLE train available." This proposed changed was not justified in WCAP-16125 and therefore, the proposed change is not acceptable without further plant specific justification.

REFERENCES

1. Pietrangelo, A. R., NEI, to W. D. Beckner, "Transmittal of CEOG Technical Report, CE NPSD-1208, Revision 0," January 24, 2001.

2. CE NPSD-1208, "Justification for Risk-Informed Modifications to Selected Technical Specifications for Conditions Leading to Exigent Plant Shutdown," CE Owner's Group, December 2000.

3. WCAP-16125-NP, Revision 0, "Westinghouse Owners Group Transmittal of WCAP-16125, Rev. 0, 'Justification for Risk-Informed Modifications to Selected Technical Specifications for Conditions Leading to Exigent Plant Shutdown,'" October 3, 2003.

4. CE-NPSD-1186, "Technical Justification for the Risk-Informed Modification to Selected Required Action End States for CEOG PWRs," CE Owner's Group, April 2000.

5. NUREG-1432, "Standard Technical Specifications, Combustion Engineering Plants," Revision 2, USNRC, June 2001.

6. Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decision Making on Plant Specific Changes to the Licensing Basis," USNRC, August 1998.

7. Regulatory Guide 1.177, "An Approach for Pant Specific Risk-Informed Decision Making: Technical Specifications," USNRC, August 1998.

8. NUREG-1022, Revision 2, "EVENT REPORTING GUIDELINES 10 CFR 50.72 and 50.73" USNRC, October 2000.

9. Regulatory Guide 1.182, "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants," USNRC, May 2000.

Date Withdrawn: April 3, 2007

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APPENDIX A

EXAMPLES OF REQUIREMENTS AND/OR RESTRICTIONS FOR THE IMPLEMENTATION OF THE PROPOSED RISK-INFORMED TS CHANGES FOR CONDITIONS LEADING TO EXIGENT PLANT SHUTDOWN (LCO 3.0.3)

1. LCO 3.4.9, Pressurizer Heaters

   The extension of the time to take action to restore one pressurizer heater group to 24 hours will be allowed only after verification that the plant can be controlled via backup equipment. In case it is found that additional equipment failures are present which increase significantly the likelihood of plant trip, a controlled plant shutdown should be initiated.

2. LCO 3.4.11, Pressurizer PORVs and Associated Block Valves

   The proposed change does not apply to (a) PORVs that are leaking, (b) PORVs that cannot be isolated by block valves, and (c) PORVs that are not expected to be isolable following a demand.

3. LCO 3.5.1, Safety Injection Tanks

   Compensatory actions will be taken to ensure both LPSI and all HPSI trains are available to partially offset the impact of SIT unavailability.

WITHDRAWN

4. LCO 3.5.2, Low Pressure Safety Injection

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

5. LCO 3.5.2, High Pressure Safety Injection

   Ensure at least two charging pumps are available during TS entry. Charging pumps may be used to support accident responses to smaller sized pipe failure events and for events with one or more stuck open PORVs or PSVs and for SGTRs. Good maintenance practices minimize the simultaneous unavailability of similar equipment (e.g., SITs, LPSIs, and swing HPSIs if available). In addition, appropriate measures should be taken to protect the AFW system from both internal and external events (e.g., fires) during TS entry in order to minimize the challenge probability to the inoperable HPSI system.

6. LCO 3.6.1, Containment

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

7. LCO 3.7.11, Control Room Emergency Air Cleanup System

Using configuration risk management program ensure plant staff is aware of the system inoperability and that respiratory units and control room pressurization systems are available and operational and that leakage pathways are properly controlled. Equipment must be available to identify the onset of a radiological release (or if applied to non-radiation atmospheric cleanup, a toxic gas release). Also, ensure availability of alternate shutdown panels and local shutdown stations should remote actions become necessary.

8. LCO 3.7.12, Control Room Emergency Air Temperature Control System

Administrative actions should be taken to ensure plant staff is aware of the system inoperability and that respiratory units and CR pressurization systems are available and operational and that leakage pathways are properly controlled. Temporary cooling may also be established via use of portable fans, propping open doors, or similar actions. Also, availability of alternate shutdown panels and local shutdown stations should be ensured.

WITHDRAWN

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

January 27, 2009

Mr. Anthony Nowinowski, Manager
Owners Group Program Management Office
Westinghouse Electric Company
P.O. Box 355
Pittsburgh, PA 15230-0355

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION RE: PRESSURIZED WATER REACTOR OWNERS GROUP (PWROG) TOPICAL REPORT (TR) WCAP-16125-NP, REVISION 1, "JUSTIFICATION FOR RISK-INFORMED MODIFICATIONS TO SELECTED TECHNICAL SPECIFICATIONS FOR CONDITIONS LEADING TO EXIGENT PLANT SHUTDOWN" (TAC NO. MD8138)

Dear Mr. Nowinowski:

By letter dated January 7, 2008, the PWROG submitted for U.S. Nuclear Regulatory Commission (NRC) staff review TR WCAP-16125-NP, Revision 1, "Justification for Risk-Informed Modifications to Selected Technical Specifications for Conditions Leading to Exigent Plant Shutdown." Upon review of the information provided, the NRC staff has determined that additional information is needed to complete the review. On January 12, 2009, Chad Holderbaum, PWROG Project Manager, and I agreed that the NRC staff will receive your response to the enclosed Request for Additional Information (RAI) questions by April 15, 2009. If you have any questions regarding the enclosed RAI questions, please contact me at 301-415-1053.

Sincerely,

Holly D. Cruz, Project Manager
Special Projects Branch
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

Project No. 694

Enclosure: RAI questions

cc w/encl: See next page

cc:
Mr. James A. Gresham, Manager
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REQUEST FOR ADDITIONAL INFORMATION

BY THE OFFICE OF NUCLEAR REACTOR REGULATION

TOPICAL REPORT (TR) WCAP-16125-NP, REVISION 1

"JUSTIFICATION FOR RISK-INFORMED MODIFICATIONS TO SELECTED

TECHNICAL SPECIFICATIONS FOR CONDITIONS LEADING TO EXIGENT

PLANT SHUTDOWN" (TAC NO. MD8138)

PRESSURIZED WATER REACTOR OWNERS GROUP (PWROG)

PROJECT NO. 694

1. The U. S. Nuclear Regulatory Commission (NRC) staff has determined that the exclusion area boundary (EAB) thyroid dose values calculated with use of the more realistic χ/Q values (10^{-5} to 10^{-4}), and consideration of containment sprays operational, range between 10^{+4} to 10^{+6}. However, your assessment noted that these values range from 10^{+4} to 10^{+5}, resulting in an order of magnitude lower than the EAB thyroid dose values assessed by the NRC staff. Please justify your determination of the EAB thyroid dose values presented in Table 4.3-1 for large early release associated with quantitative health objective by TR WCAP-16125-NP, Revision 1. In addition, please provide detailed calculations showing the derivation of each of the release values (Curies) as shown in Table 4.3-1.

2. The TR proposes that the end states for some of the new Technical Specifications (TS) actions be Mode 4 instead of Mode 5. The NRC staff has previously approved for implementation TS Task Force Traveler (TSTF)-422-A which revises TS action end states. However, the proposed end states identified in Table 2-1 are not, in all cases, consistent with TSTF-422-A. Specifically, limiting conditions for operation (LCOs) 3.6.8 and 3.6.10 are not in the scope of TSTF-422-A, and should retain a Mode 5 end state. However, LCO 3.7.11 and 3.7.12 are in the scope of TSTF-422-A with a Mode 4 end state, this end state could be retained.

   Further, a licensee must implement TSTF-422-A to justify the revised end states of this TR for these new TS actions. If a licensee chose to request implementation of this TR without having applied for TSTF-422-A, then the existing end states (i.e., Mode 5) of its TS should be retained for consistency, with the end state for the one inoperable train action requirement.

   The TR should be revised to: 1) make the end states consistent with the previously approved TSTF-422-A end states and 2) to identify the requirement to retain plant-specific consistency of the LCO action end states if TSTF-422-A has not been implemented.

ENCLOSURE

3. In Table 4.1-1 for the low pressure safety injection (LPSI) system and power-operated relief valve (PORV), the NRC staff requests the following clarifications:

   a) LPSI is identified as a backup mitigation function for small break loss-of-coolant accidents (LOCAs) and steam generator tube ruptures (SGTRs). Further details describing how the risk analyses credit this system for mitigation of these initiating events are requested.

   b) The pressurizer PORV is identified as not required for mitigation of a SGTR. Typically, a PORV is used for depressurization of the reactor coolant system (RCS) and termination of primary-to-secondary leakage.

4. Table 4.3-2 identifies the probability of a system challenge during the proposed new action requirement of 24 hours. Please provide an explanation as to why this not then multiplied by the frequency of entering the new action requirement (i.e., 1-in-3 years or 1-in-5 years) as was done with the core damage mitigation systems in Table 4.1-3.

5. In all of the subparts of Section 5.1 for the core damage mitigating systems, the basis for the proposed change is not adequately described. Specifically, the incremental conditional core damage probability (ICCDP) and incremental conditional large early release probability (ICLERP) are typically identified, but the TR in Section 4.0 (Page 9, third paragraph) had previously stated that Regulatory Guide (RG) 1.174 requirements for core damage frequency (ΔCDF) and large early release frequency (ΔLERF) were the applicable basis for these changes. Tables 4.1-3 and 4.2-1b present these metrics, although there is no comparison to the RG 1.174 guidance. The TR needs to clearly identify the basis for which the NRC staff is being requested to approve these changes.

6. Section 5.1.3 for Tier 2 Restrictions refers to the Defense-in-Depth section, which states that the TS will include an action to confirm that LCO 3.7.5 is met. It is not clear to the NRC staff what this means (i.e., are all auxiliary feedwater pumps required to be operable?). In addition, LCO 3.7.5 is not always applicable in Mode 4 in the standard TS. It is not clear whether application of the new TS action in Mode 4 would therefore be acceptable. The TR needs to address these issues.

7. In Section 4.3.1.2, the TR states that availability of containment spray (CS) reduces the need for the additional fission product filtering afforded by the shield building exhaust air cleanup system (SBEACS), penetration room exhaust air cleanup system (PREACS), and integrated control system (ICS). However, in Sections 5.2.1 (SBEACS), 5.2.2 (ICS), and 5.2.6 (PREACS), the availability of the CS system is not discussed, nor is any Tier 2 or Tier 3 requirement identified. The availability of CS would seem to be an appropriate defense-in-depth consideration for these systems. The TR needs to address this concern.

8. Section 5 identifies both Tier 2 restrictions and Tier 3 restrictions for each TS LCO. However, there are also recommendations found in other sections such as Defense-in-Depth, which are not reflected in either the Tier 2 or Tier 3 sections. This seems to establish a third layer of restrictions. It is not clear how the NRC staff is

intended to consider these items in its safety evaluation (SE). Are these requirements for implementation? The TR needs to clarify its intent.

9. The subsections in Section 5.2 addressing the radiological control systems, the basis for the proposed changes are not clear. For example, in Section 5.2.1 for the SBEACS, the only basis for the new TS action is that there is no impact on ICCDP and ICLERP. All the other statements in this section are statements of fact about the system function. A clear basis must be established for acceptability of the proposed changes. This will involve not only the zero impact on severe accidents, but should also clearly identify the defense-in-depth afforded by the other operable systems (i.e., CS for the SBEACS), and the calculated frequency of having a "less-than-LERF" release meeting the RG 1.174 acceptance guidelines for LERF.

10. The NRC staff review noted the following items during its review of the TR which may warrant some revisions to the document.

    a) In Section 1.0, Page 1, and in other places throughout the TR, a distinction is drawn between Tier 2 actions and Tier 3 actions, based on the inclusion of Tier 2 actions in the TS action statement. The NRC staff notes that this is not consistent with RG 1.177, Tier 2 and Tier 3 considerations. RG 1.177 defines Tier 2 considerations as any restriction on equipment unavailability which is required to avoid high risk configurations. Tier 3 considerations involve the application of a configuration risk management program to assess emergent equipment unavailabilities. The redefinition of Tier 2 and Tier 3 in this TR may lead to confusion during implementation by licensees and will not be consistent with the NRC staff SE, which will use the RG 1.177 definitions.

    b) In Section 4.1.1, sentences five through seven ("In this bounding risk approach,...") and sentence 11 ("In general, it is assumed...") conflict, in that the sentences five through seven states that core damage is assumed based on unavailability of the systems, and sentence 11 says this is true only "in general."

    c) In Section 4.1.2, a separate calculation of the ICCDP for a 23-hour period is provided. This is inconsistent with RG 1.177, which defines the ICCDP as the incremental risk over the entire action time, not just for the proposed increase in time.

    d) In Section 4.2, Category 2 (severe accidents accompanied by loss of containment isolation) is identified as resulting in an early release (last sentence of first paragraph), then it is further clarified that Category 2 encompasses a range of events varying from early to late, and scrubbed and unscrubbed. These two sentences are inconsistent.

    e) In Section 4.2.4, Page 28, under Final Comments, the term ΔLERP is assumed to be a typographical error, and should be ΔLERF.

    f) In Section 4.3, Page 28, second paragraph after the bullet list of systems, it is not clear to what "containment iodine controlled atmospheric cleanup systems" refers,

and it is also not clear why this and the control room emergency air temperature control system are specifically described in the second paragraph.

g) In Section 5.0, Page 39, the paragraph after the list of subsections identifies that defense-in-depth is not maintained. The prior discussions which identify that there is significant redundancy of mitigation capability among the various systems being evaluated with regards to their impact on mitigation of offsite dose made a case that there is defense-in-depth maintained for achieving the dose mitigation function.

h) In Section 5.0, Page 39, the statement that the one hour plant shutdown (assumed to refer to TS 3.0.3 action requirement) is based on an inoperable containment being a violation of plant design is without basis.

i) In Section 5.1.2, Page 44, it is stated that the reactor vessel head vents may be cycled during a plant cooldown to eliminate steam voids which may have formed. The NRC staff considers this statement to be inaccurate since the purpose of the system is to vent non-condensable gasses only. Opening the head vents in a saturated RCS will only further increase voiding in the head.

j) In Section 5.1.5,Page 52, in the Basis for Proposed Change subsection, the third and fourth sentences regarding LPSI impact on SGTR events appear to be incomplete:

   i. The clause "as for many systems" needs further explanation as to what is intended to be conveyed.

   ii. The statement that LPSI is "required to be aligned to the SDC [shutdown cooling]" is not understood in the context of a justification for condition where neither train of LPSI is available.

   iii. The statement regarding the "risk impact of a plant shutdown with availability of SDCS [SDC system]" is not understood since neither LPSI is available for SDC.

PWROG
PWR Owners Group

Program Management Office
4350 Northern Pike
Monroeville, Pennsylvania 15146

WCAP-16125-NP, Rev. 2 (Non-Proprietary)
Project No. 694

May 8, 2009

OG-09-186

Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, DC 20555-001

Subject: Pressurized Water Reactor Owners Group
Responses to the NRC Request for Additional Information (RAI) on Topical Report (TR) WCAP-16125-NP, Revision 1, "Justification for Risk-Informed Modifications to Selected Technical Specifications For Conditions Leading to Exigent Plant Shutdown" (PA-LSC-0364 Revision 2)

References:

1. PWROG Letter, F. Schiffley to Document Control Desk, "Transmittal of WCAP-16125-NP, Revision 1, "Justification for Risk-Informed Modifications to Selected Technical Specifications For Conditions Leading to Exigent Plant Shutdown," OG-08-7, January 7, 2008.

2. NRC Letter, Holly D. Cruz of NRR to Mr. Anthony Nowinowski of the PWR Owners Group Program Management Office, "Request for Additional Information Re: Pressurized Water Reactor Owners Group (PWROG) Topical Report (TR) WCAP-16125-NP, Revision 1, "Justification for Risk-Informed Modifications to Selected Technical Specifications For Conditions Leading to Exigent Plant Shutdown", (TAC No. MD8138), January 27, 2009.

In January 2008, the Pressurized Water Reactor Owners Group (PWROG) submitted WCAP-16125-NP (Non-Proprietary), Rev. 1, "Justification for Risk-Informed Modifications to Selected Technical Specifications For Conditions Leading to Exigent Plant Shutdown," for review and approval (Reference 1). In January 2009, NRC staff provided a formal Request for Additional Information (RAI) (Reference 2) for WCAP-16125-NP Revision 1.

Enclosure 1 to this letter provides the RAI responses to the 10 questions received in Reference 2. Enclosure 2 to this letter provides the primary changes and markups included in section 1 of WCAP-16125-NP Revision 2.

These RAI responses are being provided to support issuance of the draft Safety Evaluation on WCAP-16125-NP Revision 2.

Following receipt of the Safety Evaluation for WCAP-16125-NP Revision 2, this letter will be incorporated into the approved version and will be issued as WCAP-16125-NP-A, Revision 2.

If you have any questions concerning this matter, please feel free to call Chad Holderbaum at 412-374-6230.

Sincerely,

for Dennis Buschbaum

Dennis Buschbaum, Chairman
Pressurized Water Reactor Owners Group

DEB:CMH:rfn

Enclosures 2

cc: PWROG Management Committee
PWROG Licensing Subcommittee
PWROG Project Management Office
R.E. Schneider – Westinghouse
P.J. Hijeck - Westinghouse
R.J. Schomaker – AREVA NP
J.D. Andrachek – Westinghouse
J.A. Gresham – Westinghouse

Westinghouse

To: File
cc:

Date: August 10, 2009

From: Raymond Schneider
Ext: 860-731-6461
Fax: 860-731-6439

Your ref:
Our ref: LTR-RAM-II-09-020-NP

Subject: Transmittal of Response to Request for Additional Information Regarding Topical Report (TR) WCAP-16125-NP, Revision 1, "Justification for Risk-Informed Modifications to Selected Technical Specifications for Conditions Leading to Exigent Plant Shutdown" (TAC No. MD8138)

Attached is the Response to Request for Additional Information Regarding Topical Report (TR) WCAP-16125-NP, Revision 1, "Justification for Risk-Informed Modifications to Selected Technical Specifications for Conditions Leading to Exigent Plant Shutdown". This response has been reviewed and determined to be Non-Proprietary. No other changes have been made to this letter.

This work has been performed in support of PWROG Task PA-LSC-0364. The RAIs specifically refer to questions with regard to WCAP-16125-NP, Revision 1. This report is applicable to Combustion Engineering Designed PWRs.

Regards,

Raymond Schneider*
Author
Risk Applications & Methods II

* Electronically Approved Records are authenticated int eh Electronic Document Management System

Response to Request for Additional Information Regarding Topical Report (TR) WCAP-16125-NP, Revision 1 "Justification for Risk-Informed Modifications to Selected Technical Specifications for Conditions Leading to Exigent Plant Shutdown" (TAC No. MD8138)

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1. The U. S. Nuclear Regulatory Commission (NRC) staff has determined that the Exclusion Area Boundary (EAB) thyroid dose values calculated with use of the more realistic χ/Q values (10^{-5} to 10^{-4}), and consideration of containment sprays operational, range between 10^{+4} to 10^{+6}. However, your assessment noted that these values range from 10^{+4} to 10^{+5}, resulting in an order of magnitude lower than the EAB thyroid dose values assessed by the NRC staff. Please justify your determination of the EAB thyroid dose values presented in Table 4.3-1 for large early release associated with quantitative health objective by TR WCAP-16125-NP, Revision 1. In addition, please provide detailed calculations showing the derivation of each of the release values (Curies) as shown in Table 4.3-1.

Response

Justify your determination of the EAB thyroid dose values presented in Table 4.3-1 for large early release.

The stated assumption in Table 4.3-1 of the TR is that sprays are assumed to be available for both the bounding χ/Q dose and the realistic range χ/Q dose cases. For the "Approximate 2 hour Ci release (only CSS operable)" entry in Table 4.3-1, the Ci release in Table 4.3-1 is given as a range of 2.5E6 to 30E6 Ci. The upper bound of this range is actually the value calculated for the Large Early Release (LER) <u>without</u> sprays operating (see Section 5.2.1). In an upper bound sense then (taking this release uncertainty into account), the dose could exceed that due to the expected release with sprays; i.e., 2.5E6 Ci of I-131 (see also Section 5.2.1). Therefore, for this case, the dose is calculated as follows:

Bounding χ/Q Dose with Sprays = Ci Release•χ/Q•Breathing Rate•Dose Conversion Factor > 2.5E6 Ci•1E-3 sec/m^3•3.5E-4 m^3/sec•1.1E6 rem/Ci > 1E6 rem thyroid

For the more realistic case, the release is assumed to be limited to that calculated for the case with sprays assumed to be available (2.5E6 Ci of I-131) such that the only uncertainty is in the assumed χ/Q:

Realistic Range χ/Q Dose with Sprays = Ci Release•χ/Q•Breathing Rate•Dose Conversion Factor = 2.5E6 Ci•(1E-5 to 1E-4 sec/m^3)•3.5E-4 m^3/sec•1.1E6 rem/Ci = 1E4 to 1E5 rem thyroid

Provide detailed calculations showing the derivation of each of the release values (Curies) as shown in Table 4.3-1.

1. Large Early Release

<u>Spray Credit</u>

A number of judgments and assumptions have been made in order to determine a representative activity release to the environment associated with a LER. Key to making that determination is the estimate of containment spray impact on the LER release. The spray impact is estimated to be approximately a factor of 12 reduction in the Ci of I-131 released from the containment as compared to the LER case without sprays.

In order to demonstrate that the estimated spray removal impact is not dependent specifically on the TID-14844/Regulatory Guide 1.4 methodology, the Alternative Source Term (AST) spray methodology is evaluated, as well.

Considering first the TID-14844/Regulatory Guide 1.4 methodology (since that is the principal methodology employed in Table 4.3-1), the spray removal rate for iodine is limited to 20 per hour by the applicable regulatory guidance. Usually, that rate is achievable. However, only 91% of the total iodine (the elemental fraction stipulated in Regulatory Guide 1.4) is assumed to be removed at this rate; organic iodine is not removed at all and particulate iodine is removed at a substantially lower rate. Therefore, the integrated iodine airborne (normalized to the release from the RCS so the units of the integral is "hours") over the first two hours may be estimated to be 0.09•2 hours + 0.91• $(1-e^{-(20\ /\text{hour}\cdot 2\ \text{hours})})/20$ /hour = 0.18 hours + 0.05 hours = 0.23 hours. If the full release were airborne for two hours, the integrated iodine airborne (normalized to the release from the RCS) would be 2.0 hours. Therefore, the spray impact is about an order of magnitude reduction in what the two-hour release would be assuming a uniform fractional release rate from the containment over the two-hour dose period.

For the Alternative Source Term (AST), the results would be similar. Assuming a uniform rate of release from the RCS over ~ two hours as an approximation of the AST, the release rate (normalized to the release from the RCS) would be ~0.5 /hour. Spray removal rates are lower for the particulate-dominated AST than for the TID-14844 elemental iodine. One may estimate the particulate removal value to be between 5 and 10 per hour; an average of 7.5 per hour is used in the example below. In the example, 95% of the activity (the particulate iodine) is removed at 7.5 per hour and ~5% of the activity (the elemental iodine, 4.85% to be exact) is removed at 20 per hour.

Equilibrium is reached rapidly, so the steady-state expression $A = S/\lambda$ (where $S = \lambda A$ at steady state) is used for the spray case to characterize the fraction of the release that's airborne over the first two hours (A) in terms of the fractional release rate (S = 0.5 /hour) and the removal rate (λ = 7.5 /hour for particulate and 20 /hour for elemental iodine). The time-integrated result for the case with sprays is A•2 hours = 0.5 /hour•(0.95/7.5 /hour + 0.05/20 /hour)•2 hours = 0.13 hours (normalized, as before, to the release from the RCS). This may be compared to the integrated A over the first two hours without sprays which would be 1.0 hours (again, normalized to the release from the RCS). However, for the first two hours after the release is <u>complete</u> (rather than the first two hours from the start of release), the result would be 2.0 hours without sprays, same as for TID-14844/Regulatory Guide 1.4. So here again, the spray impact is about an order of magnitude reduction in the average amount of activity airborne over the two-hour dose period (i.e., 0.13 hours with sprays vs. 1.0 to 2.0 hours without sprays).

<u>Large Early Release Containment Leakage</u>

Having estimated the I-131 activity in the containment over the first two hours (normalized by the activity release) both with and without sprays, one must estimate the LER uniform containment leak rate over that same period in order to obtain the integrated release to the environment. The fraction released to the environment is assumed to be 30% for the case without sprays and 2.5% for the case with sprays.

To justify these two estimates, consider, for example, a representative LER pathway consisting of a 6" diameter (0.2 ft^2) opening (e.g., a failed-open mini-purge flow path producing a pre-existing containment failure). Assuming a sonic velocity of ~1100 fps through this pathway, it would exhibit a leak rate of about 220 cfs, amounting to a fractional leak rate of ~6 /day or ~0.25 /hour for a typical containment free volume of 3.0E6 ft^3. This leak rate is small enough (relative to the spray removal rates of >7.5 per hour) to not significantly alter the impact of sprays, but large enough to result in substantial removal of containment activity in two hours (i.e., $1-e^{-0.25\ /\text{hour}\cdot 2\ \text{hours}}$ = ~40% release).

In the absence of sprays, some natural removal would be expected to occur, reducing the 40% release to an estimated 30% release over two hours for the given leak rate.

For a 24-hour release, the 0.25 per hour containment leak rate is not sustainable because of decreasing containment pressure. Under these conditions, natural removal would be expected to have an even greater impact on the amount released than that assumed for the two-hour release. However, no attempt has been made to estimate more closely what the potential 24-hour release would be except to indicate >30% for the 24-hour release in Table 4.3-1.

With sprays in operation, the average amount of activity airborne in the containment over the first two hours (and the amount leaked) would be reduced by nearly an order of magnitude as documented above. Since sprays also tend to reduce the containment pressure and leak rate (as well as the airborne activity), the overall spray impact has been estimated to be about a factor of 12 (20% greater than the estimated factor of ten impact of sprays on airborne activity alone), reducing the fraction of core inventory released over the first two hours from 30% to 2.5%. Importantly, the nature of sprays is to reduce releases beyond two hours even more dramatically than the two-hour releases. For this reason, the 24-hour release for the spray case is stated to be the same as the two-hour release in Table 4.3-1.

Large Early Release Activity Release

The conversion of percentages to Ci of I-131 has been done on the basis of 1E8 Ci of I-131 in the core (typical for a 4000 MWt core). Therefore, the 30% release is stated as 30E6 Ci of I-131 and the 2.5% release is stated as 2.5E6 Ci of I-131.

Note that the range of I-131 releases has been given as 2.5E6 to 30E6 Ci of I-131 for all five of the applicable entries in Table 4.3-1 for the Large Early Release. This is because of the uncertainties in the above estimates. For cases without sprays, one would expect the release to be near the upper end of the stated range. For cases with sprays, one would expect the release to be at the lower end of that range, particularly when "realistic" doses are being calculated.

Regarding the impact of the Iodine Cleanup System (ICS), the Shield Building Emergency Air Cleanup System (SBEACS), the Penetration Room Emergency Air Cleanup System (PREACS), and/or the ECCS Pump Room Emergency Air Cleanup System (ECCS-PREACS), it is doubtful these systems would be effective in reducing the activity released for a large, pre-existing opening, an Interfacing-system LOCA (ISLOCA) initiator, or an energetic containment failure. These are the containment failure modes typically considered in connection with a LER.

2. MHA Release

The MHA release is similar to that for the LER except the release from the core is reduced by about a factor of 2 to 2.5, the containment leak rate is reduced by a factor of 1200 (~6 per day to 0.005 per day), and natural removal would be expected to be somewhat more effective than the LER case because of greater residence time inside containment. These three effects are estimated to reduce the LER value without sprays (30E6 Ci I-131 leaked) by a factor in excess of 3000. The estimated release for the MHA without sprays (and no other mitigation system) is a value <1E4 Ci I-131.

For the MHA with some or all mitigation systems available, sources other than containment leakage (specifically, ESF leakage) are likely to become important. A sump volume of 500,000 gallons and an ESF leak rate of 2 gpm are assumed, leading to a fractional leak rate of 4E-6 per minute.

Assuming 50% of the core inventory of I-131 is in the sump (5E7 Ci), the ESF leakage release rate for I-131 (assuming 10% iodine re-evolution) is 20 Ci per minute or 2400 Ci over two hours. The order of magnitude reduction for sprays is taken into account reducing the containment leakage contribution to <1E3 Ci I-131 from <1E4 Ci I-131 without sprays. The sum of the containment leakage and the ESF leakage contributions is presented as <3.5E3 Ci I-131 if the Containment Spray System (CSS) is available, but no other mitigation systems are available.

For the case of CSS available as well as the ICS, the SBEACS, or the PREACS being available (but not all), an additional DF of 2 is applied to containment leakage I-131 release. This is based on virtually complete removal of the iodine in any of the containment leakage captured by these containment leakage collection systems, but the potential for as much as 50% bypass with not all systems being available. The DF of 2 is viewed as a minimum DF value. With the DF of 2 credit for the containment leakage contribution, the overall I-131 release is reduced from <3.5E3 Ci to <3.0E3 Ci.

If there is no secondary treatment of containment leakage (i.e., no ICS, SBEACS, or PREACS), but the ECCS-PREACS is available, the ESF leakage contribution is expected to be reduced by a factor of at least 20 (120 Ci of I-131 released) since there is little potential for bypass of systems designed to collect iodine released from ESF leakage. Since there is already conservatism in the <1E3 Ci of I-131 from containment leakage, the result for this combination (CSS + ECCS-PREACS) is stated as <1E3 Ci of I-131.

Finally, if all mitigation systems are available, a DF of 20 is assumed to be available for all containment and ESF leakage, reducing the <3.5E3 Ci I-131 released (sprays only) to <1.5E2 Ci I-131 released (recognizing the conservatism in the <3.5E3 Ci value).

3. DBA with Limited Fuel Damage

For the DBA with limited fuel damage case, two events were compared. One is a large LOCA with successful core cooling (estimated to involve a release from the fuel of less than 0.1% of the core inventory of I-131) and the second is a rod ejection accident. The second case (the rod ejection accident) was estimated to be more limiting.

The two-hour release without mitigation was estimated on the basis of 7% cladding failure and a gap fraction of 10%, leading to a total release from the core of 0.7% of the core inventory of I-131 or 7E5 Ci of I-131.

Both the steam generator release pathway and the containment release pathway were evaluated for the rod ejection accident with the containment release pathway being judged as more significant by about a factor of 40 (without mitigation). The two-hour, unmitigated release was estimated by assuming a containment leakrate of 0.005 per day (~0.00021 per hour) and a DF of 2 for plateout inside containment. Therefore, the unmitigated release is 7E5 Ci I-131•0.00021 per hour•2 hours•0.5 = 147 Ci of I-131. This value has been entered in Table 4.3-1 as <200 Ci of I-131.

Applying a DF of 10 for sprays reduces the release to <20 Ci of I-131.

For consideration of ICS, SBEACS, or PREACS, the same DF of 2 (considering the potential for bypass) has been applied as was applied for the MHA. This consideration reduces the release to <10 Ci I-131.

Because ESF leakage is not considered for the rod ejection accident, the ECCS-PREACS has no impact, and the release remains the same as sprays-only; i.e., <20 Ci of I-131, if only sprays and ECCS-PREACS are available.

With all mitigation systems available, the <20 Ci of I-131 could nominally be reduced to <1 Ci I-131. However, the potential for the steam generator release pathway precludes this reduction. If all of the activity were released via the steam generator pathway, the release to the environment would be ~10 Ci of I-131. With some of the activity released via each pathway, the result would be a release to the environment of <10 Ci I-131, but not <1 Ci I-131.

2. The TR proposes that the end states for some of the new Technical Specifications (TS) actions be Mode 4 instead of Mode 5. The NRC staff has previously approved for implementation TS Task Force Traveler (TSTF)-422-A which revises TS action end states. However, the proposed end states identified in Table 2-1 are not, in all cases, consistent with TSTF-422-A. Specifically, limiting conditions for operation (LCOs) 3.6.8 and 3.6.10 are not in the scope of TSTF-422-A, and should retain a Mode 5 end state. However, LCO 3.7.11 and 3.7.12 are in the scope of TSTF-422-A with a Mode 4 end state, this end state could be retained.

   Further, a licensee must implement TSTF-422-A to justify the revised end states of this TR for these new TS actions. If a licensee chose to request implementation of this TR without having applied for TSTF-422-A, then the existing end states (i.e., Mode 5) of its TS should be retained for consistency with the end state for the one inoperable train action requirement.

   The TR should be revised to: 1) make the end states consistent with the previously approved TSTF-422-A end states and 2) to identify the requirement to retain plant-specific consistency of the LCO action end states if TSTF-422-A has not been implemented.

Response

The request to modify end states has been removed. This has resulted in changes to Table 2-1, Sections 4.4 and 4.5, and discussions contained in Sections 5.2.1 and 5.2.2.

3. In Table 4.1-1 for the Low Pressure Safety Injection (LPSI) system and power-operated relief valve (PORV), the NRC staff requests the following clarifications:

   a) LPSI is identified as a backup mitigation function for small break Loss-of-Coolant Accidents (LOCAs) and Steam Generator Tube Ruptures (SGTR). Further details describing how the risk analyses credit this system for mitigation of these initiating events are requested.

   b) The pressurizer PORV is identified as not required for mitigation of a SGTR. Typically, a PORV is used for depressurization of the Reactor Coolant System (RCS) and termination of primary-to-secondary leakage.

Response

(a) The LPSI was not credited in the SGTR assessment. The footnote (d) will be changed to (b) "System/Component is not required to avert core damage for this event".

(b) Table 4.1-1 identified the PORV as not being required for mitigation of an SGTR. While the PORVs may be used for depressurization, CEN-152 (Combustion Engineering Owner's Group "Emergency Procedure Guidelines," (EPGs)) directs that they should only be used as a last resort in the unlikely event that main and auxiliary pressurizer spray are not available and it is necessary to lower pressurizer pressure. The EPGs directs a controlled cooldown while staying within post accident Pressure-Temperature limits and cooldown rates. Pressure control by a PORV requires close operator attention, because the resultant pressure decrease, when the PORV(s) is opened, can be dramatic. In addition, the operator must closely monitor RCS inventory control and pressure/temperature conditions in the RDT/containment while utilizing this method. As a result of the low likelihood of the use of the PORV for this event, component was identified with footnote b, "System/Component is not required to avert core damage for this event".

4. Table 4.3-2 identifies the probability of a system challenge during the proposed new action requirement of 24 hours. Please provide an explanation as to why this not then multiplied by the frequency of entering the new action requirement (i.e., 1-in-3 years or 1-in-5 years) as was done with the core damage mitigation systems in Table 4.1-3.

Response

The intent of the table was to demonstrate a bounding likelihood of the system being challenged within the AOT. The expected frequency of this challenge may be estimated by dividing the per entry challenge probability by a factor reflecting the frequency of condition entry (e.g., 1-in 3 years). This would effectively result in challenge frequencies between 4E-08 per year and 2.7 E-07 per year. With the exception of the containment spray, the systems identified in the subject table are cooling, ventilation and /or radiological cleanup systems that do not contribute to core damage frequency or large early release frequency. Thus, regardless of the system challenge frequency the CDF and LERF for these systems is zero. The CS spray CDF and LERF impacts are separately evaluated in Tables 4.1-3 and 4.2-1b respectively.

5. In all of the subparts of Section 5.1 for the core damage mitigating systems, the basis for the proposed change is not adequately described. Specifically, the Incremental Conditional Core Damage Probability (ICCDP) and Incremental Conditional Large Early Release Probability (ICLERP) are typically identified, but the TR in Section 4.0 (Page 9, third paragraph) had previously stated that Regulatory Guide (RG) 1.174 requirements for core-damage frequency (ΔCDF) and large early release frequency (ΔLERF) were the applicable basis for these changes. Tables 4.1-3 and 4.2-1b present these metrics, although there is no comparison to the RG 1.174 guidance. The TR needs to clearly identify the basis for which the NRC staff is being requested to approve these changes.

Response

The appropriate subsections of 5.1 have been modified to specifically compare the incremental risks with both the RG 1.174 and RG 1.177 guidelines. The results of this comparison are summarized below and are captured in Section 4.2.5.

Comparison of Estimated Incremental Risks to RG 1.174 Guidelines for "Permanent" Change				
System	**Change in CDF (1/yr)**	**RG 1.174 Guideline Metric for "Small Change" in Mean CDF**	**Change in LERF**	**RG 1.174 "Small Change" in Mean LERF**
SIT	4.57E-09	1.00E-06	1.37E-11	1.00E-07
LPSI	4.11E-08	1.00E-06	1.23E-10	1.00E-07
CS (no CARC available)	1.67E-07	1.00E-06	2.81E-09	1.00E-07
PORV	1.53E-07	1.00E-06	1.11E-08	1.00E-07
Boration Systems	1.55E-08	1.00E-06	1.12E-09	1.00E-07
Pressurizer Heaters (once in 3 years)	1.00E-07	1.00E-06	3.76E-09	1.00E-07
Pressurizer Heaters (once in 5 years)	6.00E-08	1.00E-06	2.26E-09	1.00E-07

Comparison of Estimated Incremental Risks to RG 1.177 Guidelines for "TS Change"				
System	**ICCDP**	**RG 1.177 "Small Change" in Mean CDF**	**ICLERP**	**RG 1.177 "Small Change" in Mean LERF**
SIT	1.37E-08	5.0 E-07	4.11E-11	5.0 E-08
LPSI	1.23E-07	5.0 E-07	3.70E-10	5.0 E-08
CS (no CARC available)	5.02E-07	5.0 E-07	8.44E-09	5.0 E-08
PORV	4.60E-07	5.0 E-07	3.32E-08	5.0 E-08
Boration Systems	4.66E-08	5.0 E-07	3.36E-09	5.0 E-08
Pressurizer Heaters	3.01E-07	5.0 E-07	1.13E-08	5.0 E-08

For those systems where changes in the unavailability of the system have no direct impact on ICCDP or ICLERP and will not change the plant CDF or LERF (e.g., ECCS PREACS), the following was added to the Section 5 discussions.

"Since extending the AOT on these systems has no impact on the aforementioned risk metrics, the impact of the proposed modification to the TS for this system is below the RG 1.174 incremental risk guidelines and derivative RG 1.177 TS guidelines."

6. Section 5.1.3 for Tier 2 Restrictions refers to the Defense-in-Depth section, which states that the TS will include an action to confirm that LCO 3.7.5 is met. It is not clear to the NRC staff what this means (i.e., are all auxiliary feedwater pumps required to be operable?). In addition, LCO 3.7.5 is not always applicable in Mode 4 in the standard TS. It is not clear whether application of the new TS action in Mode 4 would therefore be acceptable. The TR needs to address these issues.

Response

The defense-in-depth considerations for Specification 3.4.11, PORVs, will require verification that LCO 3.7.5, "AFW System," is met. In Technical Specifications, the term "met" means that the LCO (which requires the AFW system to be Operable) is met. Therefore, all AFW trains must be Operable. LCO 3.4.11 is applicable in Modes 1, 2, and 3. LCO 3.7.5 is also applicable in Modes 1, 2, 3.

7. In Section 4.3.1.2, the TR states that availability of Containment Spray (CS) reduces the need for the additional fission product filtering afforded by the Shield Building Exhaust Air Cleanup System (SBEACS), Penetration Room Exhaust Air Cleanup System (PREACS), and Integrated Control System (ICS). However, in Sections 5.2.1 (SBEACS), 5.2.2 (ICS), and 5.2.6 (PREACS), the availability of the CS system is not discussed, nor is any Tier 2 or Tier 3 requirement identified. The availability of CS would seem to be an appropriate defense-in-depth consideration for these systems. The TR needs to address this concern.

Response

The TR states that the availability of containment sprays is a defense in depth consideration and adds the verification of an Operable train of containment spray to the Required Actions of the applicable Technical Specifications. (See "Defense in Depth" discussions in the associated Section 5 entries). As the system inoperability has no impact on CDF or LERF, this action does not meet the definition of Tier 2 or Tier 3 actions. In some cases, these defense in depth actions were inaccurately identified as being Tier 2 requirements. The TR has been corrected.

8. Section 5 identifies both Tier 2 restrictions and Tier 3 restrictions for each TS LCO. However, there are also recommendations found in other sections such as Defense-in-Depth, which are not reflected in either the Tier 2 or Tier 3 sections. This seems to establish a third layer of restrictions. It is not clear how the NRC staff is intended to consider these items in its Safety Evaluation (SE). Are these requirements for implementation? The TR needs to clarify its intent.

Response

Regulatory Guide 1.174 states that risk-informed changes to the Technical Specifications must be based on five principles, summarized as:

1. The proposed change meets the regulations.
2. The proposed change is consistent with the defense-in-depth philosophy.
3. The proposed change maintains sufficient safety margins
4. When proposed changes result in an increase in core damage frequency or risk, the increases should be small and consistent with the intent of the Commission's Safety Goal Policy Statement.
5. The impact of the proposed change should be monitored using performance measurement strategies.

Tier 2 and Tier 3 considerations fall under the fourth and fifth principle: the changes in risk should be small and should be monitored using performance measurement strategies (e.g., the program in place to satisfy 10 CFR 50.65, the Maintenance Rule).

Defense in depth actions meet the second principle, and therefore, represent a separate set of restrictions. In risk-informed changes that affect the AOT of a single train of a system, the defense-in-depth is provided by the remaining Operable train of the system. In the proposed change, the subject systems are inoperable and in some cases additional defense-in-depth requirements are identified. The TR takes the position that these defense-in-depth requirements are sufficiently important to include in the plant's Technical Specifications and are a condition on the use of the proposed AOTs.

Information on these principles has been added to Section 1.0 of the TR. Table 2-2 has been added to the TR to provide a summary of the actions to be included in plant documentation including administrative guidance, CRMP and Technical Specifications.

9. The subsections in Section 5.2 addressing the radiological control systems, the basis for the proposed changes are not clear. For example, in Section 5.2.1 for the SBEACS, the only basis for the new TS action is that there is no impact on ICCDP and ICLERP. All the other statements in this section are statements of fact about the system function. A clear basis must be established for acceptability of the proposed changes. This will involve not only the zero impact on severe accidents, but should also clearly identify the defense-in-depth afforded by the other operable systems (i.e., CS for the SBEACS), and the calculated frequency of having a "less-than-LERF" release meeting the RG 1.174 acceptance guidelines for LERF.

Response

Section 5.2 discussion of radiological control systems is requesting approval of these changes based on the low risk impact of the change as defined in RG 1.174 and RG 1.77, the defense in depth afforded by an Operable train of containment spray and the low frequency of potential "Less than LERF" releases. Furthermore, repair of these systems "at power" reduces the potential for any additional risks that may be associated with transitioning the plant to shutdown and returning the plant to power. The individual sections have been re-written accordingly.

10. The NRC staff review noted the following items during its review of the TR which may warrant some revisions to the document.

    a) In Section 1.0, Page 1, and in other places throughout the TR, a distinction is drawn between Tier 2 actions and Tier 3 actions, based on the inclusion of Tier 2 actions in the TS action statement. The NRC staff notes that this is not consistent with RG 1.177, Tier 2 and Tier 3 considerations. RG 1.177 defines Tier 2 considerations as any restriction on equipment unavailability which is required to avoid high risk configurations. Tier 3 considerations involve the application of a configuration risk management program to assess emergent equipment unavailabilities. The redefinition of Tier 2 and Tier 3 in this TR may lead to confusion during implementation by licensees and will not be consistent with the NRC staff SE, which will use the RG 1.177 definitions.

Response

As discussed in the previous responses, Tier 2 considerations are not appropriate for inclusion in the Technical Specifications. Technical Specification Required Actions are added to various Specifications to address defense-in-depth. In some cases, the TR inappropriately attributed those actions to Tier 2 considerations. The TR has been corrected and additional explanation has been added to Section 1.0 to clarify the relationship between defense in depth, Tier 2, and Tier 3 considerations.

    b) In Section 4.1.1, sentences five through seven ("In this bounding risk approach,...") and sentence 11 ("In general, it is assumed...") conflict, in that the sentences five through seven states that core damage is assumed based on unavailability of the systems, and sentence 11 says this is true only "in general."

Response

Text has been rewritten to remove this conflict.

    c) In Section 4.1.2, a separate calculation of the ICCDP for a 23-hour period is provided. This is inconsistent with RG 1.177, which defines the ICCDP as the incremental risk over the entire action time, not just for the proposed increase in time.

Response

Calculations have been adjusted to consider the full AOT. This resulted in changes to Tables 4.1-2, 4.1-3, 4.2-3b, 4.2-2c, 4.2.5 and associated discussion in Sections 4 and 5.

    d) In Section 4.2, Category 2 (severe accidents accompanied by loss of containment isolation) is identified as resulting in an early release (last sentence of first paragraph), then it is further clarified that Category 2 encompasses a range of events varying from early to late, and scrubbed and unscrubbed. These two sentences are inconsistent.

Response

While releases occurring during events with an unisolated containment can take many forms, all unisolated containment conditions were considered to be contributors to LERF. This is a conservative position and will increase the expected LERF contribution.

e) In Section 4.2.4, Page 28, under Final Comments, the term ΔLERP is assumed to be a typographical error, and should be ΔLERF.

Response

The term has been changed in the revised TR

f) In Section 4.3, Page 28, second paragraph after the bullet list of systems, it is not clear to what "containment iodine controlled atmospheric cleanup systems" refers, and it is also not clear why this and the control room emergency air temperature control system are specifically described in the second paragraph.

Response

The phrase has been revised to refer to all cleanup systems identified above. The revised phrase reads "atmospheric iodine cleanup systems". The CREATCS was called out in addition since it also controls environmental temperature.

g) In Section 5.0, Page 39, the paragraph after the list of subsections identifies that defense-in-depth is not maintained. The prior discussions which identify that there is significant redundancy of mitigation capability among the various systems being evaluated with regards to their impact on mitigation of offsite dose made a case that there is defense-in-depth maintained for achieving the dose mitigation function.

Response

The statement refers specifically to the fact that the TS is entered based on both trains of a system being inoperable. Other systems provide defense in depth capability. This is clarified in the revised TR as follows:

The TS changes being proposed generally are associated with the inoperability of an entire system (or unavailability of a given function). To address this loss of function, public safety is maintained by ensuring public risk is acceptably low, taking compensatory measures to provide defense in depth using alternate systems, and by providing an opportunity to repair the equipment during power operations thereby potentially avoiding the additional risk of plant transitions.

h) In Section 5.0, Page 39, the statement that the one hour plant shutdown (assumed to refer to TS 3.0.3 action requirement) is based on an inoperable containment being a violation of plant design is without basis.

Response

The TR has been reworded as follows: "The requirement for an immediate (1 hour) shutdown is usually based on the loss of an important safety function. The 1 hour time to initiate a plant shutdown was chosen to provide time to prepare for the plant evolution. The goal is to place the plant in a condition where the health and safety of the public could be better assured, but in a controlled manner."

i) In Section 5.1.2, Page 44, it is stated that the reactor vessel head vents may be cycled during a plant cooldown to eliminate steam voids which may have formed. The NRC staff considers this statement to be inaccurate since the purpose of the system is to vent non-condensable gasses only. Opening the head vents in a saturated RCS will only further increase voiding in the head.

Response

The sentence has been reworded to remove reference to head vents as follows:

"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 and/or adjusting RCS make-up flow rate."

j) In Section 5.1.5,Page 52, in the Basis for Proposed Change subsection, the third and fourth sentences regarding LPSI impact on SGTR events appear to be incomplete:

   i. The clause "as for many systems" needs further explanation as to what is intended to be conveyed.

Response

The associated text has been revised as follows:

"There is no significant impact of the unavailability of LPSI following SGTR events. A short term unavailability of the LPSI subsystems will result in a negligible incremental increase in the plant risk associated with large LOCA events."

References to the SDC function of LPSI are removed from the TR as no change is being requested to the SDC specification requirements.

   ii. The statement that LPSI is "required to be aligned to the SDC [shutdown cooling]" is not understood in the context of a justification for condition where neither train of LPSI is available.

Response

References to the SDC function of LPSI are removed from the TR as no change is being requested to the SDC specification requirements.

iii. The statement regarding the "risk impact of a plant shutdown with availability of SDCS [shutdown cooling system]" " is not understood since neither LPSI is available for shutdown cooling.

Response

References to the SDC function of LPSI are removed from the TR as no change is being requested to the SDC specification requirements.

RESPONSE TO REQUEST #2 FOR ADDITIONAL INFORMATION REGARDING

TOPICAL REPORT (TR) WCAP-16125-NP, REVISION 1

"JUSTIFICATION FOR RISK-INFORMED MODIFICATIONS TO SELECTED

TECHNICAL SPECIFICATIONS FOR CONDITIONS LEADING TO EXIGENT

PLANT SHUTDOWN"

PRESSURIZED WATER REACTOR OWNERS GROUP (PWROG)

NRC Question #11

11. A 72-hour CT is proposed to apply to TS 3.6.6A for the condition with two of two CS trains inoperable. TS 3.6.6A is applicable to plants which credit iodine removal from their CS system. Therefore, the proposed CT is not consistent with the 24-hour CT proposed for inoperabilities of two trains of a system credited for iodine removal (i.e., SBEACS, ICS, PREACS, and CREACS). Further, no defense-in-depth measures are proposed for operability of these other systems, which are identified in the TR as providing a similar function to the CS for iodine removal. The staff would consider a 24-hour CT for two inoperable CS trains, provided the TS also required operability of the other systems for iodine removal (SBEACS, ICS, PREACS, and CREACS) for defense-in-depth. Propose a 24-hour alternative to the 72-hour CT which includes appropriate compensatory measures.

PWROG Response to Question #11

From an iodine removal perspective, operation of fan coolers is expected to provide iodine removal via the impact of condensation of water in the containment atmosphere and the impact of the Containment Air Recirculation Coolers (CARCs). Using advanced source term methods, one CE plant no longer relies on containment sprays for either post-LOCA heat removal or iodine removal. For purposes of consistency Westinghouse Electric Company (WEC) will adjust its request for the CE PWRs to limit the TS 3.6.6.A CT for 2 spray trains out of service to 24 hours. During this condition at least one train of Containment Air Coolers must be operable. Furthermore, as a defense in depth action, a TS requirement will be added to confirm operability of the CREACS. The operable containment air coolers will maintain containment pressure, thus limiting containment leakage. Availability of the CREACS will mitigate the consequences of the reduction in post accident iodine removal to the control room and therefore help maintain functionality of the control room following a severe core damage event and minimize the risk of needing control room countermeasures.

As a result of the short duration of the outage and operability of the containment air coolers, the impact on core damage and large early release risk metrics is negligible. Thus, additional technical specification actions are not considered necessary. However, in addition to the TS requirement, Maintenance Rule risk assessment guidance will identify the importance of other plant iodine removal systems (e.g., SBEACS, ICS, and PREACS, as appropriate for the plant.). This change will be reflected in the final submittal of WCAP-16125-NP-A, Revision 2.

NRC Question #12

12. An 8-hour CT is proposed to apply to TS 3.6.6A and 3.6.6B when 3 or more containment heat removal trains are inoperable. Currently, this condition requires an immediate plant shutdown. As identified in the TR, this condition represents a loss of the containment heat removal function, and also fails the ECCS function during the recirculation phase due to overheating of the sump water. No compensatory measures to provide the containment heat removal function are identified. Because the condition fails functions associated with post-accident decay heat removal with no compensatory measures established to achieve the functions, the change is not consistent with other changes in the TR and not acceptable as proposed. Propose deletion of this change from the TR.

PWROG Response to Question #12

This TS will be removed from the requested CT extensions.

NRC Question #13

13. For TS 3.7.11, the TR proposes a new action requirement to implement mitigating actions to address chemical and toxic gas releases, smoke, and other hazards. The staff considers such actions to be plant-specific, requiring specific staff review to assure acceptability. A generic approval of the action is not considered to be acceptable. Propose that this new action be addressed on a plant-specific basis.

PWROG Response to Question #13

Specification 3.7.11, Control Room Emergency Air Cleanup System (CREACS), currently contains actions for an inoperable barrier that renders all CREACS trains inoperable. However, the Topical Report focused on the radiological impact of the inoperability of this system for a period of 24 hours. While the Condition discussed in the Topical Report is for an equipment inoperability which may partially or fully disable the CREACS, the function of the CREACS is also defeated by a loss of the control room envelope. This later Condition currently requires implementation of mitigating actions and verification that the mitigating actions ensure control room envelope occupant exposures to radiological, chemical, and smoke hazards will not exceed limits. This action was approved on a generic basis by the NRC as TSTF-448, "Control Room Habitability," and published in the Federal Register on January 17, 2007. The mitigating actions are described in the Bases as follows: "During the period that the CRE boundary is considered inoperable, action must 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. Actions must be taken within 24 hours to verify that in the event of a DBA, the mitigating actions will

ensure that CRE occupant radiological exposures will not exceed the calculated dose of the licensing basis analyses of DBA consequences, and that CRE occupants are protected from hazardous chemicals and smoke. These mitigating actions (i.e., actions that are taken to offset the consequences of the inoperable CRE boundary) should be pre-planned for implementation upon entry into the condition, regardless of whether entry is intentional or unintentional. The 24 hour Completion Time is reasonable based on the low probability of a DBA occurring during this time period, and the use of mitigating actions." The specific actions taken under this Required Action are left to licensee control. If the licensee verifies that the control room envelope occupants are protected, 90 days are provided to restore the boundary and the supported CREACS trains to Operable status.

WCAP-16125 proposes the same mitigating actions be implemented immediately when the CREACS trains are inoperable for reasons other than an inoperable boundary. The Topical Report proposes that a 24 hour Completion Time be provided in this condition.

Given that TSTF-448 allows the use of licensee controlled mitigating actions to protect the control room occupants from radiological, chemical, and smoke hazards for a period of 90 days when both CREACS trains are inoperable due to an inoperable barrier, we believe that the generic Completion Time extension of 24 hour for two inoperable CREACS trains, for less hazardous conditions, is reasonable. For plants that have not adopted TSTF 448, additional plant specific justification will be required. The above discussion will be added to the Topical Report to support the CT extension request.

ATTACHMENT: E-Mail Transmittal of RAIs

From: Cruz, Holly
To: Holderbaum, Chad M.
Sent: Mon Jun 08 11:23:17 2009
Subject: REQUESTS FOR ADDITIONAL INFORMATION - TR WCAP-16125 REV. 2 (Nos. 11-13)
Chad,

Please find the additional RAIs noted below, and advise when you think the PWROG will be able to respond.

11. A 72-hour CT is proposed to apply to TS 3.6.6A for the condition with two of two CS trains inoperable. TS 3.6.6A is applicable to plants which credit iodine removal from their CS system. Therefore, the proposed CT is not consistent with the 24-hour CT proposed for inoperabilities of two trains of a system credited for iodine removal (i.e., SBEACS, ICS, PREACS, and CREACS). Further, no defense-in-depth measures are proposed for operability of these other systems, which are identified in the TR as providing a similar function to the CS for iodine removal. The staff would consider a 24-hour CT for two inoperable CS trains, provided the TS also required operability of the other systems for iodine removal (SBEACS, ICS, PREACS, and CREACS) for defense-in-depth. Propose a 24-hour alternative to the 72-hour CT which includes appropriate compensatory measures.

12. An 8-hour CT is proposed to apply to TS 3.6.6A and 3.6.6B when 3 or more containment heat removal trains are inoperable. Currently, this condition requires an immediate plant shutdown. As identified in the TR, this condition represents a loss of the containment heat removal function, and also fails the ECCS function during the recirculation phase due to overheating of the sump water. No compensatory measures to provide the containment heat removal function are identified. Because the condition fails functions associated with post-accident decay heat removal with no compensatory measures established to achieve the functions, the change is not consistent with other changes in the TR and not acceptable as proposed. Propose deletion of this change from the TR.

13. For TS 3.7.11, the TR proposes a new action requirement to implement mitigating actions to address chemical and toxic gas releases, smoke, and other hazards. The staff considers such actions to be plant-specific, requiring specific staff review to assure acceptability. A generic approval of the action is not considered to be acceptable. Propose that this new action be addressed on a plant-specific basis.

Thanks for your help,

Holly

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WCAP-16125-NP, Rev. 2 (Non-Proprietary)
Project No. 694

November 11, 2009

OG-09-440

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

Subject: Pressurized Water Reactor Owners Group
Comments to the NRC Draft Safety Evaluation for Pressurized Water Reactor Owners Group (PWROG) on Topical Report (TR) WCAP-16125-NP, Revision 2, "Justification for Risk-Informed Modifications to Selected Technical Specifications For Conditions Leading to Exigent Plant Shutdown", (PA-LSC-0364, Revision 2)

References:

1. NRC Letter, Stacey L. Rosenberg, Chief, Special Projects Branch of NRR to Mr. Tony Nowinowski of the PWR Owners Group Program Management Office, "Draft Safety Evaluation for Pressurized Water Reactor Owners Group (PWROG) Topical Report (TR) WCAP-16125-NP, Revision 2, "Justification for Risk-Informed Modifications to Selected Technical Specifications For Conditions Leading to Exigent Plant Shutdown", (TAC No. MD8138), October 19, 2009.

This letter provides the PWROG comments on the draft Safety Evaluation (SE) for WCAP-16125-NP, Revision 2. As requested, the comment package includes a markup to the draft SE, enclosure 2, and a Table of the recommended changes, Enclosure 1. The comments include the need to revise the terms of usage of the change to be consistent with the Topical Report justification and the intent of the change. A basis for this recommendation is provided as an attachment to the tabulated comments shown in Enclosure 1.

Following receipt of the Safety Evaluation for WCAP-16125-NP Revision 2, this letter will be incorporated into the approved version and will be issued as WCAP-16125-NP-A, Revision 0.

If you have any questions concerning this matter, please feel free to call Chad Holderbaum at (412) 374-6230.

Sincerely,

Dennis E. Buschbaum, Chairman
Pressurized Water Reactor Owners Group

DEB:CMH:las

Enclosures: (2)

cc: PWROG Management Committee
PWROG Licensing Subcommittee
PWROG Risk Management Subcommittee
PWROG Program Management Office
R. E. Schneider, Westinghouse
P. J. Hijeck, Westinghouse
J. D. Andrachek, Westinghouse
J. A. Gresham, Westinghouse
R .J. Schomaker, AREVA NP
Brian Mann, Excel Services

ENCLOSURE 1

Table 1: PWROG Comments on Draft SE for WCAP-16125-NP, REVISION 2		
Page/Line(s)	Recommended Change	Comment
Page 4, Line 35	Recommend modifying sentence as follows: However, since these proposed changes to CTs are not intended for routine voluntary removal of redundant systems or components from service in lieu of other alternatives that would not result in redundant systems or components being inoperable,	(see Attachment A)
Page 4, Lines 29-30	"application of a particular TS action requirement" should be revised to "risk-informed completion time"	Reg Guides 1.174 and 1.177 are only applicable to risk-informed completion times.
Page 5, Line 23	Revise "result in" to "require"	clarification
Page 5, Line 29	Should "unavailable" be revised to "inoperable"?	clarification
Page 8, Line 42	Should "administrative guidance" be revised to "administrative controls"?	Question
Page 18, Lines 9-10	"plant specific information…gas release mitigation," should be revised to "plant specific information…gas release mitigation, unless TSTF-448 has been implemented,"	clarification
Page 20, Line 17	Revise "uses" to "has implemented either"	clarification
Page 22, Line 43	Revise "prohibited" to "not applicable to"	clarification
Page 23 Lines 17-19	Replace with: The changes are not intended for routine voluntary removal of redundant systems or components from service in lieu of other alternatives that would not result in redundant systems or components being inoperable.	(See Attachment A)

Attachment A

Additional Comments Changes Regarding "Voluntary Entry"

Page 4, lines 35-38 and Page 21, lines 17-20, state that the proposed Technical Specification conditions cannot be entered voluntarily and can only be applied to emergent failures of equipment. This wording is unnecessarily restrictive and not consistent with the Topical Report.

The Topical Report states that the proposed changes "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." The Topical Report also indicates that entry into these Action Statements is expected to be infrequent. These statements have been interpreted in the SE as to preclude voluntary entry into an Action Statement unless an emergent failure forces the plant into such a condition.

The current Technical Specifications require entry into LCO 3.0.3 for the Conditions justified for a limited period of time in the Topical Report. Licensees are allowed to enter LCO 3.0.3 voluntarily under certain circumstances. As stated in the LCO 3.0.3 Bases, "It [LCO 3.0.3] is not intended to be used as an operational convenience that permits routine voluntary removal of redundant systems or components from service in lieu of other alternatives that would not result in redundant systems or components being inoperable."

It is not the intent of WCAP-16125 to be more restrictive than the current Technical Specification requirements. TSTF-426 (the TSTF Traveler implementing WCAP-16125) does not add Technical Specifications prohibitions on voluntary entry of the modified Conditions. The TSTF-426 Bases for each of the Conditions modified in accordance with WCAP-16125 state, "The Required Actions are not intended for routine voluntary removal of redundant systems or components from service in lieu of other alternatives that would not result in redundant systems or components being inoperable." This is consistent with current Technical Specification requirements.

Prohibiting voluntary entry into the modified Conditions can result in increases in plant risk. For example, consider the situation in which Train A of a system is inoperable for corrective maintenance and a Surveillance is due on Train B which will render Train B inoperable for a short period of time (less than one hour). Under the current Technical Specifications, the licensee could - after performing the required Maintenance Rule assessments - determine that the safest course of action is to temporarily enter LCO 3.0.3 to perform the required testing instead of shutting down the plant to perform the test. However, if adoption of WCAP-16125 eliminated the ability to voluntarily enter the condition for two trains inoperable, a plant shutdown would be required. Alternatively, there may be other instances where planned short duration repairs to a degraded but functional component(s) prior to component failure may be more appropriate than allowing the component to fail and address the issue in an emergent manner.

In summary, the two noted locations impose requirements on the adoption of WCAP-16125 that do not appear in the Topical Report and are not consistent with the current or requested Technical Specification requirements. By retaining the existing wording in the TSTF-426 Bases, the expectation of low entry frequency would remain valid.

ENCLOSURE 2

WCAP-16125-NP, REVISION 2

"JUSTIFICATION FOR RISK-INFORMED MODIFICATIONS TO SELECTED TECHNICAL SPECIFICATIONS FOR CONDITIONS LEADING TO EXIGENT PLANT SHUTDOWN"

PRESSURIZED WATER REACTOR OWNERS GROUP

PROJECT NO. 694

1.0 INTRODUCTION AND BACKGROUND

On January 7, 2008, the Pressurized Water Reactor (PWR) Owners Group submitted topical report (TR) WCAP-16125, Revision 1, "Justification for Risk-Informed Modifications to Selected Technical Specifications for Conditions Leading to Exigent Plant Shutdown," (Reference 1) for the U.S. Nuclear Regulatory Commission (NRC) staff review and approval. In response to staff requests for additional information (RAI), TR WCAP-16125, Revision 2 (Reference 2) was submitted on May 8, 2009. Additional clarifications and changes to the scope of the TR were submitted on July 8, 2009 (Reference 3). The TR provides justification for risk-informed technical specifications (TS) initiative 6 for nuclear plants with Combustion Engineering (CE) designed nuclear steam supply systems.

The proposed changes would address conditions involving the inoperability of both redundant trains of selected plant systems, which is a loss of function of the system, leading to an exigent plant shutdown, either due to applicability of limiting condition for operation (LCO) 3.0.3 or a specific short duration shutdown action requirement in the individual TS LCO. The intent of these modifications to the TSs is to provide a risk-informed alternative to the existing shutdown requirements to permit resolution of the loss of function condition while the plant continues to operate. The completion times (CT) associated with the proposed actions are specified. The extended CTs are intended to allow for the potential restoration of the system to an operable condition, thereby avoiding the risk associated with an immediate controlled shutdown.

Table 1 summarizes the proposed changes to the standard TS (NUREG-1432).

Table 1				
TS LCO	SYSTEM/COMPONENT	CONDITION	CURRENT CT	PR(
3.1.9	Boration System	System inoperable	None (3.0.3)	24 ho
3.4.9	Pressurizer Heaters	Two groups of class 1E heaters inoperable	None (3.0.3)	24 ho
3.4.11	Power-operated Relief Valves (PORV)	Inability of two PORVs to open, or inability of both PORVs to close and block valves to be closed		8 hou
3.5.1	Safety Injection Tanks (SIT)	Two or more SITs inoperable	None (3.0.3)	24 ho
3.5.2	Low Pressure Safety Injection (LPSI)	Two LPSI subsystems inoperable	None (3.0.3)	24 ho
3.6.6.A	Containment Spray (CS) System	1) Two CSS trains inoperable, or 2) Two CSS and two Containment Cooling System (CCS) trains inoperable [a]	None (3.0.3)	1) 72 2) 12
3.6.6.B [a]	CS System	Two CSS trains inoperable, or two CSS and two Containment Cooling System (CCS) trains inoperable	None (3.0.3)	12 ho
3.6.8	Shield Building Emergency Air Cleanup System (SBEACS)	Two trains inoperable	None (3.0.3)	24 ho
3.6.10	Iodine Control System (ICS)	Two trains inoperable	None (3.0.3)	24 ho
3.7.11	Control Room Emergency Air Cleanup System (CREACS)	Two trains inoperable for reasons other than an inoperable boundary	Explicit 3.0.3	24 ho
3.7.12	Control Room Emergency Air Temperature Control System (CREATCS)	Two trains inoperable (modes 1 – 4)	Explicit 3.0.3	24 ho

Table 1				
TS LCO	SYSTEM/COMPONENT	CONDITION	CURRENT CT	PR(
3.7.13	Emergency Core Cooling System Pump Room Emergency Air Cleanup System (ECCS PREACS)	Two trains inoperable for reasons other than an inoperable boundary	None (3.0.3)	24 ho
3.7.15	Penetration Room Emergency Air Cleanup System (PREACS)	Two trains inoperable for reasons other than an inoperable boundary	None (3.0.3)	24 ho

[a] This proposed change was withdrawn by Reference 3.
[b] This proposed CT was revised from 72 hours to 24 hours by Reference 3.

2.0 REGULATORY EVALUATION

2.1 Applicable Regulations

In Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.36, "Technical Specifications," the Commission established its regulatory requirements related to the content of TSs. Pursuant to 10 CFR 50.36, TSs are required to include items in the following categories related to station operation: (1) safety limits, limiting safety system settings, and limiting control settings; (2) LCOs; (3) surveillance requirements; (4) design features; and (5) administrative controls. The rule does not specify the particular requirements to be included in a plant's TSs. As stated in 10 CFR 50.36(c)(2), "Limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When a limiting condition for operation of a nuclear reactor is not met, the licensee will shut down the reactor or follow any remedial action permitted by the technical specifications until the condition can be met."

Most TS LCOs provide a fixed time interval, referred to as the allowed outage time (AOT) or CT, during which the LCO may not be met, to permit a licensee to perform required testing or maintenance activities, or to conduct repairs. Upon expiration of the CT, the requirement to shut down the reactor or follow remedial action is imposed. The proposed changes in the TR provide a means for the licensee to make repairs to a system where both trains are inoperable, rather than immediately shutting down the plant under the current TS requirements.

2.2 Applicable Regulatory Criteria/Guidelines

The yearly risk impact is represented by the delta-core damage frequency (ΔCDF) and delta-large early release frequency (ΔLERF) metrics referenced in Regulatory Guide (RG) 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," (Reference 4). The risk of each individual risk informed completion time is represented by the incremental conditional core damage probability (ICCDP) and the incremental conditional large early release probability (ICLERP), referenced in RG 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications," (Reference 5). The guidelines in RG 1.177 for ICCDP and ICLERP are intended to assure that each entry in a TS action results in no more than a very small incremental risk. However, since these proposed changes to CTs are not intended for routine voluntary removal of redundant systems or components from service in lieu of other alternatives that would not result in redundant systems or components being inoperable, the frequency of entry into the TS actions is expected to be very rare, and so the ICCDP and ICLERP metrics are not evaluated to determine the acceptability of the proposed changes.

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General guidance for evaluating the technical basis for proposed risk-informed changes is provided in Section 19.2, "Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance," of the NRC Standard Review Plan (SRP), NUREG-0800 (Reference 6). More specific guidance related to risk-informed TS changes, including changes to TS CTs, is provided in SRP Section 16.1, "Risk-Informed Decisionmaking: Technical Specifications," (Reference 7).

Chapter 19.2 of the SRP states that a risk-informed application should be evaluated to ensure

1 that the proposed changes meet the following key principles:

- The proposed change meets the current regulations, unless it explicitly relates to a requested exemption.

- The proposed change is consistent with the defense-in-depth philosophy.

- The proposed change maintains sufficient safety margins.

- When proposed changes increase core damage frequency or risk, the increase(s) should be small and consistent with the intent of the Commission's Safety Goal Policy Statement.

- The impact of the proposed change should be monitored using performance measurement strategies.

3.0 TECHNICAL EVALUATION

The staff evaluated the TR for conformance with the five key principles of SRP 19.2.

3.1 Compliance with Current Regulations

Regulations in 10 CFR 50.36 permit either a plant shutdown or other remedial actions specified by TSs when an LCO is not met. The proposed changes provide new action requirements for conditions of equipment inoperability which currently require an immediate plant shutdown. Since such remedial actions are permitted per 10 CFR 50.36, the proposed changes in the TR continue to comply with current regulations, and therefore, satisfy this key principle.

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3.2 Defense-in-Depth

The proposed changes address conditions where both trains of a system are inoperable, resulting in a loss of that system's function. Each proposed change addresses available defense-in-depth measures and includes requirements to assure defense-in-depth is adequately maintained. Therefore, this key principle is satisfied by the unique requirements identified for each proposed TS change.

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3.3 Safety Margins

The proposed changes do not have any impact on the use of NRC-approved codes and standards, nor do the changes impact any acceptance criteria used in a plant's licensing basis. Under the current TSs, if an accident occurs during the 6-hour controlled shutdown time of LCO 3.0.3 caused by two trains of these systems being unavailable, it could potentially result in offsite dose limits that do not meet NRC regulatory limits. Since the changes proposed by WCAP-16125 do not modify the design basis of the systems evaluated, extending the AOT to 24 hours would have no quantitative effect on the dose consequence as compared to the existing condition. As such, the proposed changes would not alter a nuclear power plant's available safety margin, and this key principle is satisfied.

3.4 Performance Monitoring

The proposed changes would permit continued plant operation for short periods to address equipment failures. Degradation of equipment performance could lead to excessive use of the new action requirements. This is adequately addressed by equipment performance monitoring required by 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," and therefore, this key principle is satisfied.

3.5 Risk Assessment

The risk of each of the TS LCOs for which action requirements are proposed is evaluated in the TR by three methods, as described below.

1. For those TSs governing systems or components which provide mitigation of core damage and large early releases, λCDF and λLERF metrics are calculated using a simplified generic method, and the results are compared to the acceptance guidelines of RG 1.174. This applies to TS 3.1.9 Boration Systems, 3.4.11 PORVs, 3.5.1 SITs, and 3.5.2 ECCS (LPSI).

   For calculations of λCDF, a bounding approach was applied to evaluate loss of function of a system by identifying the initiating events for which the system provides mitigation, and assuming that the event goes directly to core damage. No credit was taken for alternate mitigation strategies, and the baseline CDF was effectively assumed to be zero. The initiating event frequencies were taken from NUREG/CR-5750 (Reference 8).

   For λLERF, a simplified approach using an event tree was developed to calculate the fraction of core damage events which result in large early releases. The event tree assessed containment isolation status, reactor coolant system (RCS) pressure, secondary side depressurization via the steam generators, thermally-induced steam generator tube rupture (SGTR), and reactor pressure vessel (RPV) lower head failure. Assumptions related to the potential impact on LERF for each of these events, and the associated basis for probabilities used in the analysis, are discussed below:

   Containment Isolated – This event defines containment integrity prior to the core damage event. If containment is not isolated, then a large early release will occur concurrent with core damage. A probability of 3.0E-3 was applied for an unisolated containment, which is identified as the upper end of the range used in the CE PRA models.

   RCS Pressure – High – This event defines the RCS pressure at the time of core damage. If the pressure is low, then large early releases are assumed not to occur (except via an unisolated containment); otherwise, thermally-induced SGTR and high pressure melt ejection events are further evaluated. All core damage events involving loss-of-coolant accidents (LOCAs) are assumed to result in low or intermediate RCS pressure, and all other events result in high RCS pressure.

   Steam Generator Depressurization – This event defines the status of the secondary side, and affects the next event which is the potential for induced SGTR. Depressurization of the secondary occurs either due to prior operator response or due to failure of a safety relief valve. Based on NUREG-1570 (Reference 9), a probability of 0.9 is assigned for secondary

depressurization.

Thermally-induced SGTR Occurs – This event represents a loss of steam generator tube integrity due to thermal stresses during a severe accident, which is assumed to result in a large early release. Two values are used, based on the status of the prior event, steam generator depressurization. A probability of 0.5 is assigned when the steam generators are depressurized, and 0.01 otherwise. These values are conservative, based on the assumptions regarding tube age and integrity and based on neglecting operator actions to depressurize the RCS after core damage.

RPV Lower Head Failure Results in Containment Failure – This event represents a high pressure failure of the lower head, with an energetic discharge of the molten fuel and direct containment heating (DCH), leading to failure of containment. Based on NUREG/CR-6338 (Reference 10), the conditional containment failure probability given the event for CE-designed plants is 0.01, which is considered to be a bounding value.

None of the assessed initiating events include either SGTRs or other containment bypass events because the systems being evaluated do not mitigate these events. The staff concludes that the simplified LERF event tree is reasonable and acceptable to support the evaluation of LERF for the scope of the TR.

2. For TS 3.4.9 Pressurizer Heaters, an evaluation of the increased likelihood of a plant trip due to degraded pressure control is made in order to calculate λCDF. The λLERF calculation for this TS is the same simplified approach described above for case 1.

3. For all remaining systems associated with mitigation of radiological releases with magnitudes less than those associated with LERF, there is no impact to either CDF or LERF, as the systems are provided to meet design basis dose limits. An evaluation of the frequency of events which challenge the systems is made and compared to the acceptance guidelines of RG 1.174 applicable to λLERF in order to characterize the risk of these lesser releases. Additional justification is provided based on the availability of other systems which provide a degree of defense-in-depth for prevention of these releases. The following systems are addressed in this manner: TS 3.6.10 ICS, TS 3.6.8 SBEACS, TS 3.7.15 PREACS, TS 3.7.13 ECCS PREACS, TS 3.7.11 CREACS, and TS 3.7.12 CREATCS.

The primary purpose of these systems is to ensure that accident radiation exposures meet the limits as defined in General Design Criteria 19, "Control Room," 10 CFR 100.11, "Determination of Exclusion Area, Low Population Zone, and Population Center Distance," or 10 CFR 50.67, "Accident Source Term," for plants that have adopted the alternative source term. The design basis dose consequence analyses that credit these systems are based on the assumption of an intact containment. In addition, since these systems are designed only to be used in the event of an accident condition, the operability of these systems has no impact on normal releases of radioactive material as governed by 10 CFR Part 20, "Standards for Protection Against Radiation."

To assess the impact of the unavailability of these systems, the TR examined the expected iodine releases for three categories of events:

- Beyond design basis scenarios that lead to large early releases,
- Maximum Hypothetical Accident (MHA), and
- LOCA and Non-LOCA Design Basis Accidents (DBA).

The purpose of this assessment is to show that, using worst case assumptions, the potential accident releases anticipated under the short term operational conditions proposed by the increased CT for the ICS, SBEACS, PREACS and ECCS PREACS will be well below and bounded by a large early release. For clarity, the evaluation was limited to the release of iodine. For each category, iodine releases were estimated assuming various combinations of system availability. The results of this assessment are shown in Table 4.3-1 of the TR, supplemented by additional information in Reference 2. The NRC staff reviewed the assumptions and methodology used to determine the bounding iodine release quantities and resulting dose consequences and found that in all cases appropriately conservative assumptions were used. In many cases the assumptions used were more conservative than the applicable regulatory guidance. For instance, the Table 4.3-1 MHA analyses assume a containment leak rate of 0.5% per day which is significantly higher than the TS allowed leak rate normally used in dose consequence analyses. The evaluation of the iodine release contribution from ECCS leakage was based on a conservative iodine partition factor of 0.01. This is considered to be a conservative assumption as suggested in the applicable regulatory guidance; however, in various licensing actions, the NRC staff has accepted more realistic values when supported by a sound technical basis.

In most of these cases evaluated, the offsite doses for the MHA and DBA with various combinations of systems inoperable are near or below regulatory limits. If more realistic yet still conservative assumptions were incorporated into the dose consequence analysis, it could easily be shown that offsite doses for the MHA and DBA accidents would be below regulatory limits for all the cases evaluated. Further, when realistic meteorological conditions are assumed and containment sprays are available, calculated doses for all the evaluated accident conditions including the MHA are shown to be within regulatory limits.

To reduce the impact of an increased CT for the CREACS, the TR proposes to add conditions to verify that RCS specific activity is within limits and to verify that dose mitigating actions are available in the control room. For limited durations, such as the short term operational conditions proposed by the increased CT for the CREACS, the NRC has accepted credit for the use of respirators and potassium iodide on an interim basis to demonstrate that control room dose limits can be met.

Similarly, the TR proposes to add pre-planned actions to ensure that impact of loss of post accident temperature control associated with an increased CT for the CREATCS is mitigated. Actions can include use of portable fans, temporary opening of doors or use of normal heating, ventilation, and air conditioning systems. To support this change, administrative guidance will be provided to monitor the control room temperature to ensure control room habitability and operability of TS equipment. If compensatory measures impact the CR envelope, the operability of containment and auxiliary building post accident air cleanup systems will be verified. The 24 hour CT proposed in the TR for the CREACS and the CREATCS is consistent with the allowed 24-hour period for the evaluation of a breach of the control room envelope provided in TSTF-448.

Based on an evaluation of the methods and assumptions used to produce the results shown in Table 4.3-1, the NRC staff has reasonable assurance that that the postulated accident releases calculated for the short term operational conditions proposed by the increased CT for the ICS, SBEACS, PREACS and ECCS PREACS will be well below the LERF releases. In addition, the NRC staff has reviewed the bases for the increased CT for the CREACS and the CREATCS and has determined that the proposed conditions and compensatory measures provide reasonable assurance that control room habitability will be adequately maintained during the proposed 24-hour CT.

External events, including internal fires and floods, were not evaluated in the TR. None of the systems being evaluated provide in a primary mitigating function for external events, and therefore these events are not significant to the risk-informed decision.

The TR also evaluated sensitivity studies for key areas of uncertainty in the analyses. Specifically, the TR considered uncertainties in the initiating event frequencies which are the input to the CDF calculations and showed that even assuming a 95% upper bound frequency would not result in excessive risk. These were also propagated into the LERF calculations with similar results. The TR also addressed uncertainties in the thermally-induced SGTR assumptions and steam generator depressurization assumptions, and demonstrated that the LERF results are not significantly impacted. These sensitivity studies performed to evaluate the key sources of uncertainty in the risk analyses adequately demonstrate the robustness of the results to support the proposed TS changes.

Each individual proposed change to TS is discussed in detail in Sections 3.5.1 – 3.5.10.

3.5.1 LCO 3.1.9 – Boration Systems – Operating

The boration systems are required to ensure that adequate shutdown margin exists to bring the plant to cold shutdown with the most reactive control element assembly not fully inserted into the reactor core and the decay of all xenon poison. The systems also mitigate main steam line breaks and reactor coolant pump restart by preventing a return to power scenario (due to cold water injection), and also mitigate anticipated transients without scram (ATWS) events. (Only the ATWS mitigation function is typically included in PRA models because the other functions are not directly related to core damage events.) These systems are not included in the current standard TSs (NUREG-1432) because they do not satisfy any of the criteria of 10 CFR 50.36.

The plant-specific TSs in applicable CE plants do not provide any action requirements for two inoperable boration paths, and therefore TS 3.0.3 applies, which requires an immediate plant shutdown. The proposed change provides for a 24-hour CT to restore at least one boration path to operable status, to permit continued operation under an existing action requirement.

A risk assessment of the proposed 24-hour CT was provided which conservatively assumed that all ATWS events would result in core damage, with the following results:

λCDF	RG 1.174 Guidance	λLERF	RG 1.174 Guidance
1.55E-8/yr	<1.0E-6/yr	1.12E-9/yr	<1.0E-7/yr

The λCDF and λLERF were assessed based on an assumed once per three years entry into the proposed action requirement and assumed that the entire 24-hour duration of the CT is used. The risk metrics are well below the acceptance guidelines of RG 1.174 as noted in the table.

Defense-in-depth is provided by the operability of the high head safety injection system, which can provide boration of the reactor; operability is already required by TS 3.5.2 and 3.5.3, which assure operability of at least one high-pressure safety injection (HPSI) train.

Based on the conservatively calculated risk impact being within the acceptance guidelines of RG 1.174, and on defense-in-depth provided by an operable high head safety injection train, the staff finds the proposed new action requirement and 24-hour CT are acceptable.

3.5.2 LCO 3.4.9 – Pressurizer Heaters

The pressurizer and the class 1E electrical heaters maintain a liquid-to-vapor interface to permit RCS pressure control during normal operations and in response to anticipated design basis transients. The class 1E heaters, with their power provided by emergency AC power busses, are used to maintain RCS subcooling during a natural circulation cooldown, and the unavailability of the heaters will extend the time to reach entry conditions for the shutdown cooling system. The unavailability of the class 1E heaters may complicate steady-state RCS pressure control and may increase the potential of an unplanned reactor trip. However, the availability of additional heaters beyond the two required by this TS LCO permit continued RCS pressure control.

The standard TSs do not provide any action requirements for two inoperable pressurizer heaters, and therefore TS 3.0.3 applies which requires an immediate plant shutdown. The proposed change provides for a 24-hour CT to restore at least one pressurizer heater to operable status, to permit continued operation under an existing action requirement.

The unavailability of the class 1E pressurizer heaters would not have any significant impact on plant transient response, and so there is no quantifiable impact to CDF or LERF. While mitigation of a SGTR is enhanced by the availability of pressurizer heaters, the non-class 1E heaters can also function if offsite power is available, and plant procedures provide for mitigation of a SGTR without pressurizer heaters, if necessary.

Conservatively, the risk impact due to increased likelihood of a reactor trip was calculated by assuming an order-of-magnitude increase in the reactor trip frequency when both class 1E heaters are inoperable. The risk impact is then calculated based on the conditional core damage probability given a reactor trip with no other complications:

λCDF	RG 1.174 Guidance	λLERF	RG 1.174 Guidance
1.0E-7/yr	<1.0E-6/yr	3.8E-9/yr	<1.0E-7/yr

The λCDF and λLERF were assessed based on an assumed once per three years entry into the proposed action requirement and assumed that the entire 24-hour duration of the CT is used. The risk metrics are well below the acceptance guidelines of RG 1.174 as noted in the table. Defense-in-depth is provided by the normal availability of non-class 1E heaters for normal plant pressure control, and the availability of plant procedures which permit orderly plant shutdown

and cooldown even with no pressurizer heaters available. If any heaters are available, normal plant operations can continue. Because unavailability of non-class 1E heaters would physically result in plant shutdown, the staff does not consider it necessary to specify additional TS or administrative requirements for the non-class 1E heater availability.

Based on the conservatively calculated risk impact being within the acceptance guidelines of RG 1.174, and the limited impact of plant shutdown and cooldown without pressurizer heaters, the staff finds the proposed new action requirement and 24-hour CT are acceptable.

3.5.3 LCO 3.4.11 – Pressurizer PORVs and Associated Block Valves

The pressurizer PORVs and block valves are required to be operable to minimize the potential for a small break LOCA through a PORV pathway. The PORVs also automatically open for RCS pressure control to avoid challenging the primary safety relief valves, and may be manually opened by the operator to control pressure. In the event of a total loss of feedwater to the steam generators, one or more PORVs may be opened manually to provide for feed-and-bleed cooling of the reactor using once-through cooling from high pressure injection to the RCS. The PORVs may also be used for low temperature overpressure protection during heatup and cooldown.

The standard TS is proposed to be revised to provide an 8-hour CT to restore at least one PORV or one block valve to operable status, to permit continued operation under an existing action requirement. This action may only be applied provided the PORV is isolable by its block valve.

This constraint is already provided for in the TS action requirements to close the associated block valve, and if this cannot be accomplished, an immediate plant shutdown is still required.

The risk impact of unavailable PORVs or block valves is primarily attributable to the non design basis function of providing for feed-and-bleed cooling:

λCDF	RG 1.174 Guidance	λLERF	RG 1.174 Guidance
1.5E-7/yr	<1.0E-6/yr	1.1E-8/yr	<1.0E-7/yr

The λCDF and λLERF were assessed based on an assumed once per three years entry into the proposed action requirement and assumed that the entire 8-hour duration of the CT is used. The risk metrics are well below the acceptance guidelines of RG 1.174 as noted in the table.

The primary safety relief valves provide the design basis pressure control function, controlled by TS requirements. The non design basis feed-and-bleed function is considered to be risk-significant, and the proposed change includes a TS requirement to confirm that the LCO for auxiliary feedwater (AFW) is met, which requires both trains to be operable.

In addition, the new 8-hour CT does not apply in the standard TSs to PORVs which are leaking and unisolable. Plants implementing the proposed change must verify that their TS changes are consistent with the standard TS with regards to leaking and unisolable PORVs.

Based on the risk impact being within the acceptance guidelines of RG 1.174, and the additional restriction on operability of both AFW trains in the TS action, the staff finds the proposed new action requirement and 8-hour CT are acceptable.

3.5.4 LCO 3.5.1 – Safety Injection Tanks (SIT)

The SITs provide for rapid refill of the RCS following a large break LOCA. The SITs are passive devices; no component actuation or operator control is required for the components to perform their safety function. The standard TSs explicitly require entry into LCO 3.0.3 when two or more SITs are inoperable. The proposed change would provide a 24-hour CT to restore at least all but one SIT to operable status, to permit continued operation under an existing action requirement.

The risk impact of two or more inoperable SITs was conservatively evaluated by assuming that all large break LOCAs proceed to core damage:

λCDF	RG 1.174 Guidance	λLERF	RG 1.174 Guidance
4.57E-9/yr	<1.0E-6/yr	1.37E-11/yr	<1.0E-7/yr

The λCDF and λLERF were assessed based on an assumed once per three years entry into the proposed action requirement and assumed that the entire 24-hour duration of the CT is used. The risk metrics are well below the acceptance guidelines of RG 1.174 as noted in the table.

The availability of both trains of the ECCS limits the impact of unavailable SITs, and provides additional defense-in-depth. The TS action requirement will therefore include verification that both ECCS trains are operable when more than one SIT is inoperable.

Based on the risk impact being well below the acceptance guidelines of RG 1.174, and the additional restriction on operability of both ECCS trains in the TS action, the staff finds the proposed new action requirement and 24-hour CT are acceptable.

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

The LPSI system refloods the RCS following a large break LOCA during the injection phase of ECCS operation. The system is normally shut down after the refueling water storage tank is depleted, at which point the recirculation phase of ECCS operation is entered, using the HPSI system only.

The standard TSs explicitly require entry into LCO 3.0.3 when both LPSI trains are inoperable. The proposed change would provide a 24-hour CT to restore at least one LPSI train to operable status, to permit continued operation under an existing action requirement.

The risk impact of having both LPSI trains inoperable was conservatively evaluated by assuming that all large and medium break LOCAs proceed to core damage:

λCDF	RG 1.174 Guidance	λLERF	RG 1.174 Guidance
4.11E-8/yr	<1.0E-6/yr	1.23E-10/yr	<1.0E-7/yr

The λCDF and λLERF were assessed based on an assumed once per three years entry into the proposed action requirement and assumed that the entire 24-hour duration of the CT is used. The risk metrics are well below the acceptance guidelines of RG 1.174 as noted in the table.

The availability of the SITs and both trains of the HPSI limits the impact of having no operable LPSI system by providing mitigation capability for some LOCA scenarios normally mitigated by the LPSI system. The TR stated that best-estimate analysis using the CEFLASH-4AS code on a representative CE PWR showed that LOCAs with break sizes up to the size of the largest branch line pipe break can be mitigated using one HPSI pump and the SITs. The existing structure of the standard TS would not permit applicability of the new proposed action unless both HPSI trains were operable, and therefore no additional TS requirements are needed to address operability of the HPSI system. The TS action requirement will include verification that all SITs are operable when both LPSI trains are inoperable. Therefore, defense-in-depth is enhanced by assuring all available mitigating equipment for LOCAs is operable while both LPSI trains are inoperable.

Based on the risk impact being well below the acceptance guidelines of RG 1.174, and the additional restriction on operability of the SITs and both HPSI trains in the TS, the staff finds the proposed new action requirement and 24-hour CT are acceptable.

3.5.6 LCO 3.6.8 – Shield Building Exhaust Air Cleanup System
LCO 3.6.10 – Iodine Cleanup System
LCO 3.7.15 – Penetration Room Exhaust Air Cleanup System

The SBEACS (also called the Shield Building Ventilation System), ICS, and PREACS function to assure radioactive material released from containment leakage following a design basis accident is filtered prior to being exhausted to the environment. Each system includes two redundant trains with high efficiency particulate air filters, moisture absorbers, and charcoal adsorbers in the flowpath. The SBEACS filters leakage from containment into the shield building for dual containment facilities. The PREACS filters leakage from containment into the penetration room between the containment and the auxiliary building. The ICS removes elemental iodine directly from the containment atmosphere. The design basis for these systems is a postulated MHA involving a LOCA with a short duration uncovery of fuel, resulting from a temporary interruption, or significant degradation, of the ECCS flow. The event is assumed to result in significant iodine releases (40 - 50% of core inventory) from the fuel into the containment. The containment remains intact, with no more than the design basis leakage permitted by TSs. Releases associated with the MHA are significantly below the release which would occur for a postulated large early release (at least two orders of magnitude lower). None of these systems provide any mitigation capability for preventing either core damage or large early releases.

The current standard TSs do not address the condition of two inoperable trains of these systems; therefore, a default LCO 3.0.3 entry is required, resulting in an immediate plant shutdown. The proposed change would provide a 24-hour CT to restore at least one train of the affected system to operable status, to permit continued operation under an existing action requirement.

As noted above, these systems do not provide any core damage or large early release mitigation. Therefore, the risk metrics are zero for these systems. However, it may be conservatively assumed that if any of these systems are unavailable following a postulated core damage event, then some radioactive release above design limits, but well below the large early release level, would occur. A bounding estimate for CDF of CE plants was identified as 1E-4/year, so that over a 24-hour period the probability of a significant core damage event which would require the unavailable system would be:

(1E-4/year) x (24 hours) x (year/8760 hours) = 2.7E-7

Assuming a once per three year entry into the new TS would result in a frequency of a "less than LERF" release of about 9.0E-8/year. This frequency is within the acceptance guidance of RG 1.174 applicable to large early releases, and therefore provides a context for consideration of the risk impact for smaller releases.

As noted in the TR, there are also higher frequency design basis accidents (e.g., rod ejection and reactor coolant pump locked rotor) which are assumed to result in fuel damage and therefore rely upon these systems to filter any containment leakage. These accidents are associated with releases from the fuel into containment two or more orders of magnitude below those associated with the MHA described above, and four or more orders of magnitude below large early releases.

Containment spray can effectively scrub the post-accident containment atmosphere of fission products and therefore reduce reliance upon the downstream air cleanup systems. In order to assure additional defense-in-depth protection for the spectrum of accidents for which these systems provide mitigation, the TS action will include a verification of operability of at least one train of the CS system.

Based on the zero risk impact for severe accidents being well below the acceptance guidelines of RG 1.174, and the additional restriction on operability of at least one CS train in the TS, the staff finds the proposed new action requirements and 24-hour CTs are acceptable.

3.5.7 LCO 3.7.13 – ECCS PREACS

The ECCS PREACS filters air from the area of the active ECCS components during the recirculation phase of a LOCA to remove volatized iodine resulting from leakage of post-accident recirculation fluid from the ECCS systems. As for the SBEACS, ICS, and PREACS discussed in Section 3.5.6, the design basis for the system is a postulated MHA involving a LOCA with a short duration uncovery of fuel, resulting from a temporary interruption, or significant degradation, of the ECCS flow. During the recirculation phase of ECCS operation following a LOCA, radioactive water from containment flows through the active portions of the ECCS. The ECCS PREACS assures that any nominal leakage from the ECCS (e.g., pump seals, valve flanges, etc.) is filtered before reaching the environment. Releases associated with the MHA are significantly below the release which would occur for a postulated large early release (at least two orders of magnitude lower). Therefore, this system does not provide any mitigation capability for preventing either core damage or large early releases.

The current standard TSs do not address the condition of two inoperable trains of the ECCS PREACS; therefore, a default LCO 3.0.3 entry is required, resulting in an immediate plant

shutdown. The proposed change would provide a 24-hour CT to restore at least one train of the affected system to operable status, to permit continued operation under an existing action requirement. (Note that the current standard TS do provide a 24-hour CT when both trains are inoperable due specifically to ECCS pump room boundary inoperability.)

As noted above, the ECCS PREACS does not provide any core damage or large early release mitigation. Therefore, the risk metrics are zero for this system. However, it may be conservatively assumed that if both ECCS PREACS trains are unavailable following a postulated core damage event, then some radioactive release above design limits, but well below the large early release level, would occur. Since ECCS recirculation is only used for large and medium LOCAs, a bounding estimate for a core damage event during recirculation is made by assuming all large and medium LOCAs go to core damage; the total frequency of these LOCAs was identified as 4E-5/year, so that over a 24-hour period the probability of a significant core damage event which would require the ECCS PREACS would be:

(4E-5/year) x (24 hours) x (year/8760 hours) = 1.2E-7

Assuming a once per three year entry into the new TS would result in a frequency of a "less than LERF" release of about 4.0E-8/year. This frequency is within the acceptance guidance of RG 1.174 applicable to large early releases, and therefore provides a context for consideration of the risk impact for smaller releases.

As noted in the TR, there are also higher frequency design basis accidents (e.g., rod ejection and reactor coolant pump locked rotor) which are assumed to result in fuel damage and therefore rely upon these systems to filter any containment leakage. These accidents are associated with releases from the fuel into containment two or more orders of magnitude below those associated with the MHA described above, and four or more orders of magnitude below large early releases.

Unlike for the other radiological filtration systems, the operation of the CS system does not have any mitigating impact on the unavailability of the ECCS PREACS, because the sprays transport the fission products from the containment atmosphere into the sumps, which is the source of recirculation water which is then assumed to leak into the ECCS pump room. Therefore, there is no requirement for operability of the CS system when the ECCS PREACS is inoperable.

Some designs of the ECCS PREACS rely upon the system to provide temperature control of the ECCS pump rooms, which may therefore impact the operability of the ECCS pumps and equipment. The staff has not evaluated the unavailability of the ECCS pump room cooling function, and therefore this TS change is prohibited if the ECCS PREACS is required by the plant design basis for ECCS pump room cooling.

Based on the zero risk impact for severe accidents being well below the acceptance guidelines of RG 1.174, and the prohibition for applicability of this TS change for plant designs which credit the room cooling function of the ECCS PREACS, the staff finds the proposed new action requirement and 24-hour CT is acceptable.

3.5.8 LCO 3.7.11 – Control Room Emergency Air Cleanup System

The CREACS provides for filtration of outside air delivered to the control room by the ventilation system in the event of radioactive releases of particulates or iodine from containment following an accident involving fuel failures. This is to assure that control room personnel are protected from potential radiation exposures in excess of regulatory limits. The system may also provide protection of control room personnel from chemical or toxic gas releases by isolating the control room air intakes.

The current standard TSs do not address the condition of two inoperable trains of these systems; therefore, a default LCO 3.0.3 entry is required, resulting in an immediate plant shutdown. The proposed change would provide a 24-hour CT to restore at least one train of the CREACS to operable status, to permit continued operation under an existing action requirement. (Note that the current standard TS do provide a 24-hour CT when both trains are inoperable due specifically to control room pressure boundary inoperability.)

In the event of an accident involving radioactive releases without the availability of the CREACS, there would be no direct impact on the capability of the control room staff to perform any actions required to mitigate severe core damage or large early releases, because alternative protective measures would be implemented to reduce the dose impacts. If the accident did not involve severe core damage, control room doses even without the CREACS would be minimal, and therefore the CREACS has no direct role in preventing core damage (i.e., λCDF = 0). If a core damage accident did occur with CREACS unavailable, then the bounding impact would be to simply assume the event proceeded to a large early release based on the unavailability of the control room personnel to perform any mitigating actions. This assumption would be very conservative, since large releases occur primarily due to containment bypass accidents, and control room actions following core damage do not prevent the release from occurring.

A bounding estimate for CDF of CE plants was identified as 1E-4/year, so that over a 24-hour period the probability of a significant core damage event, which with the CREACS unavailable is assumed to proceed to a large early release, would be:

(1E-4/year) x (24 hours) x (year/8760 hours) = 2.7E-7

Assuming a once per three year entry into the new TS, and assuming the entire 24-hour duration of the CT is used, the conservatively calculated λLERF is about 9.0E-8/year. This λLERF, and the zero λCDF, are below the acceptance guidelines of RG 1.174.

A significant contributor to control room radiological hazards was identified in the TR from the release of radioactive RCS fluid from a SGTR event. A required TS action to verify LCO 3.4.16, "RCS Specific Activity" is met will be included in the new proposed action to provide additional defense-in-depth.

The TR also addressed a TS action to require initiation of mitigating actions to lessen the effects of potential hazards of smoke, chemical, radiological or toxic gas releases. The staff considers the specific hazards and compensatory measures to be plant-specific, and did not find sufficient information to conclude that the proposed changes are acceptable for these events without a

plant-specific evaluation. In response to staff RAI, Reference 3 identifies that these mitigating actions were previously reviewed and approved by the staff for TS Task Force (TSTF) traveler TSTF-448 (Reference 11). TSTF-448 authorizes a generic TS change to permit a 24-hour CT when the control room boundary is inoperable, and includes the same mitigating actions to assure protection of the control room staff from non-radiological hazards. Therefore, plants which have previously adopted TSTF-448 would not require any further plant-specific reviews in order to adopt the proposed change to TS 3.7.11.

Based on the risk impact being below the acceptance guidelines of RG 1.174, the additional restriction on meeting RCS specific activity limits in the TSs and the plant-specific information regarding smoke, chemical, radiological and toxic gas release mitigation, unless TSTF-448 has been implemented, the staff finds the proposed new action requirement and 24-hour CT acceptable.

3.5.9 LCO 3.7.12 – Control Room Emergency Air Temperature Control System

The CREATCS provides for temperature control of the control room when it is isolated during accident conditions. (For some CE plants, the cooling function is integrated with the system used for air cleanup in LCO 3.7.11.) This assures control room temperature will not exceed equipment operability requirements.

The standard TSs explicitly require entry into LCO 3.0.3 when both CREATCS trains are inoperable. The proposed change would provide a 24-hour CT to restore at least one CREATCS train to operable status, to permit continued operation under an existing action requirement.

The TR stated that the unavailability of the CREATCS has a negligible impact on severe accident risk, based on long room heatup times, availability of alternate cooling strategies, and alternate means to control emergency systems locally. The staff reviewed the basis for this conclusion and considered the potential plant impacts if an accident occurred which isolated the control room while the CREATCS was inoperable.

If an accident occurred which isolated the control room without cooling, and core cooling was being maintained by the ECCS, then there would be negligible radiological consequences and the operators could simply unisolate and realign the normal control room ventilation system to provide continued cooling of the control room. Therefore, there would be no impact on CDF (i.e., λCDF = 0).

If core damage occurred after the accident and the control room needed to remain isolated without cooling, the bounding impact would be to simply assume the event proceeded to a large early release based on the unavailability of the control room personnel to perform any mitigating actions. This assumption would be very conservative, since large releases occur primarily due to containment bypass accidents, and control room actions following core damage do not prevent the release from occurring.

A bounding estimate for CDF of CE plants was identified as 1E-4/year, so that over a 24-hour period the probability of a significant core damage event, which with the CREATCS unavailable is assumed to proceed to a large early release, would be:

1 (1E-4/year) x (24 hours) x (year/8760 hours) = 2.7E-7

Assuming a once per three year entry into the new TS, and assuming the entire 24-hour duration of the CT is used, the conservatively calculated λLERF is about 9.0E-8/year. This λLERF, and the zero λCDF, are below the acceptance guidelines of RG 1.174.

Defense-in-depth is provided by alternative control room cooling actions and by the capability for local operation of equipment, if necessary. These actions are typically found in plant procedures, and are not required to be implemented by TS controls.

Based on the risk impact being below the acceptance guidelines of RG 1.174, the staff finds the proposed new action requirement and 24-hour CT acceptable.

3.5.10 LCO 3.6.6A – Containment Spray and Cooling Systems (credit taken for iodine removal)
LCO 3.6.6B – Containment Spray/Coolers (credit not taken for iodine removal)

The CS system and the containment coolers provide containment heat removal following accidents which release high energy steam to the containment. In addition to the heat removal function, the CS system enhances post accident fission product removal. Each CE plant has implemented either LCO 3.6.6A or 3.6.6B, based on whether the plant design credits the fission product removal function of the CS system. Each train of the CS system provides a nominal 50% of the cooling function, and similarly each train of the containment coolers provides 50% of the cooling function; thus the combined capacity of both systems is 200%.

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The standard TSs for LCO 3.6.6A and 3.6.6.B provide for an explicit LCO 3.0.3 entry when less than 100% containment cooling capacity is available (i.e., any combination of three or more trains inoperable). The proposed change would provide a 12-hour CT to restore at least any two trains to operable status, to permit continued operation under an existing action requirement.

For LCO 3.6.6A, when both CS trains are inoperable, and the fission product removal function is therefore not available, an explicit LCO 3.0.3 entry is required. The proposed change would provide a 72-hour CT to restore at least one CS train to operable status, to permit continued operation under an existing action requirement.

For the first proposed change regarding the loss of 100% capacity in the containment cooling system, the condition as stated in the TR represents total loss of any heat removal capability for the containment atmosphere (i.e., no CS or cooler train operable). No compensatory measures are identified to provide defense-in-depth for the heat removal function. The TR also states that unavailability of containment heat removal will result in failure of ECCS sump recirculation mode of operation, which could lead to core damage when the refueling water storage tank is depleted. The staff therefore finds that this proposed change is not acceptable, because it results in a loss of safety functions for containment heat removal and core cooling by ECCS recirculation, with no mitigation available from other systems. In response to a staff RAI, Reference 3 proposed to withdraw these proposed changes to TS 3.6.6A and TS 3.6.6B for less than 100% of containment cooling capacity available.

For the second proposed change regarding the loss of all CS trains when the system is credited for fission product removal, the proposed CT of 72 hours is inconsistent with the 24-hour CT applied for all other systems related to fission product removal, and therefore not consistent with

the intent of the TR to provide a short time period to restore one train to operable status. In addition, no compensatory measures were identified as was done for the other fission product removal systems. Therefore, the staff finds that this change, as proposed, is not acceptable, because it results in a loss of safety function (fission product removal) for an unacceptable period of time without compensatory measures.

In response to a staff RAI, Reference 3 proposed to revise the 72-hour CT for TS 3.6.6A to a 24-hour CT, consistent with the other iodine removal TS changes. Reference 3 also identified that the TS-required operability of the containment coolers would provide a similar iodine removal function such that additional TS requirements for operability of other iodine removal systems would not be required. A TS action for operability of the CREACS was proposed to assure additional defense-in-depth for control room functionality when both CS trains are inoperable during the 24-hour CT.

Based on the information in the TR, the challenge frequency of the CS system for fission product removal is identical to the challenge frequency described for the SBEACS, ICS, and PREACS, described in Section 3.5.6. Similar to those analyses, it may be conservatively assumed that if both CS trains are unavailable following a postulated core damage event, then some radioactive release above design limits, but well below the large early release level, would occur. A bounding estimate for CDF of CE plants was identified as 1E-4/year, so that over a 24-hour period the probability of a significant core damage event which would require the unavailable system would be:

(1E-4/year) x (24 hours) x (year/8760 hours) = 2.7E-7

Assuming a once per three year entry into the new TS would result in a frequency of a "less than LERF" release of about 9.0E-8/year. This frequency is within the acceptance guidance of RG 1.174 applicable to large early releases, and therefore provides a context for consideration of the risk impact for smaller releases.

When the function of the CS for fission product removal is unavailable, then the operability of the CREACS, which provides for filtration to protect control room habitability, should be verified as a defense-in-depth measure.

Based on the zero risk impact for severe accidents being well below the acceptance guidelines of RG 1.174, and the verification of operability of the CREACS, the staff finds a new action requirement with a 24-hour CT would be acceptable for the case of both CS trains inoperable.

4.0 CONCLUSION

The staff has reviewed WCAP-16125 Rev. 2 using the five key principles of risk-informed decision making, and concludes that the proposed TS changes are acceptable within the limits and conditions identified in this safety evaluation and provided below:

LCO 3.1.9 – Boration Systems – Operating: Provide a new condition for both boration flowpaths inoperable and an associated action requirement to restore at least one path to operable with a 24-hour CT.

LCO 3.4.9 – Pressurizer Heaters: Provide a new condition for both class 1E heaters inoperable and an associated action requirement to restore at least one class 1E heater to operable with a 24-hour CT.

LCO 3.4.11 – Pressurizer PORVs and Associated Block Valves: Provide a new condition for both PORVs or both block valves inoperable and an associated action requirement to restore at least one PORV or block valve to operable with a 8-hour CT. The proposed change must include a requirement controlled in the TS action to confirm that the LCO for auxiliary feedwater is met, which requires both trains to be operable. In addition, the new 8-hour CT does not apply in the standard TS to PORVs which are leaking and unisolable. Plants implementing the proposed change must verify that their TS changes are consistent with the standard TS with regards to leaking and unisolable PORVs.

LCO 3.5.1 – Safety Injection Tanks: Provide a new condition for two or more SITs inoperable and an associated action requirement to restore at least all but one SIT to operable with a 24-hour CT. The proposed change must include verification that both ECCS trains are operable when more than one SIT is inoperable.

LCO 3.5.2 – ECCS – Operating (Low Pressure Safety Injection System): Provide a new condition for both LPSI trains inoperable and an associated action requirement to restore at least one LPSI train to operable with a 24-hour CT. The proposed change must include verification that all SITs are operable when both LPSI trains are inoperable. The TS structure already assures that both high-head safety injection trains are operable when this condition is applicable.

LCO 3.6.8 – Shield Building Exhaust Air Cleanup System: Provide a new condition for both SBEACS trains inoperable and an associated action requirement to restore at least one SBEACS train to operable with a 24-hour CT. The proposed change must include verification of operability of at least one train of the CS system when both SBEACS trains are inoperable.

LCO 3.6.10 – Iodine Cleanup System: Provide a new condition for both ICS trains inoperable and an associated action requirement to restore at least one ICS train to operable with a 24-hour CT. The proposed change must include verification of operability of at least one train of the CS system when both ICS trains are inoperable.

LCO 3.7.15 – Penetration Room Exhaust Air Cleanup System: Provide a new condition for both PREACS trains inoperable and an associated action requirement to restore at least one PREACS train to operable with a 24-hour CT. The proposed change must include verification of operability of at least one train of the CS system when both PREACS trains are inoperable.

LCO 3.7.13 – ECCS PREACS: Provide a new condition for both ECCS PREACS trains inoperable and an associated action requirement to restore at least one ECCS PREACS train to operable with a 24-hour CT. The proposed change is not applicable, if the ECCS PREACS is required by the plant design basis for ECCS pump room cooling.

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LCO 3.7.11 – Control Room Emergency Air Cleanup System: Provide a new condition for both CREACS trains inoperable and an associated action requirement to restore at least one CREACS train to operable with a 24-hour CT. The proposed change must include verification that LCO 3.4.16, "RCS Specific Activity," is met, and must provide plant-specific information

regarding smoke, chemical, radiological and toxic gas release mitigation unless TSTF-448 has previously been implemented.

LCO 3.7.12 – Control Room Emergency Air Temperature Control System: Provide a new condition for both CREATCS trains inoperable and an associated action requirement to restore at least one CREATCS train to operable with a 24-hour CT.

LCO 3.6.6A – Containment Spray and Cooling Systems (credit taken for iodine removal): Provide a new condition for both CS trains inoperable and an associated action requirement to restore at least one CS train to operable with a 24-hour CT. The proposed change must include verification that the LCO for CREACS is met. These proposed changes were identified in Reference 3 and supersede the proposed changes in the TR (Reference 2).

Additional changes proposed in the TR to address loss of containment cooling capability for LCO 3.6.6A and LCO 3.6.6B are not acceptable, and were withdrawn by Reference 3.

The changes are not intended for routine voluntary removal of redundant systems or components from service in lieu of other alternatives that would not result in redundant systems or components being inoperable.

All the changes are acceptable.

Deleted: only assuming that appropriate TS prohibitions on voluntary entry into these action requirements are also implemented, to assure that the new action requirements are only applied to address emergent failures of equipment, consistent with the stated purpose of the TR

5.0 REFERENCES

1. Letter from F. P. Schiffley II to U. S. NRC, "Transmittal of WCAP-16125, Rev. 1, "Justification for Risk-Informed Modifications to Selected Technical Specifications for Conditions Leading to Exigent Plant Shutdown, (PA-LSC-0364)", January 7, 2008 (ADAMS Accession Number ML080390521).

2. Letter from D. Buschbaum to U. S. NRC, "Responses to the NRC Request for Additional Information (RAI) on Topical Report (TR) WCAP-16125-NP, Revision 1, "Justification for Risk-Informed Modifications to Selected Technical Specifications for Conditions Leading to Exigent Plant Shutdown, (PA-LSC-0364 Revision 2)", August 10, 2009 (ADAMS Accession Number ML092260399).

3. Letter from D. Buschbaum to U. S. NRC, "Responses to the NRC Request #2 for Additional Information (RAI) on Topical Report (TR) WCAP-16125-NP, Revision 1, "Justification for Risk-Informed Modifications to Selected Technical Specifications for Conditions Leading to Exigent Plant Shutdown, (PA-LSC-0364 Revision 2)", July 8, 2009 (ADAMS Accession Number ML091940063).

4. Regulatory Guide 1.174 Rev. 1, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," USNRC, November 2002.

5. Regulatory Guide 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications," USNRC, August 1998.

6. NUREG-0800, Standard Review Plan 19.2, "Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance," June 2007.
7. NUREG-0800, Standard Review Plan 16.1, "Risk-Informed Decisionmaking: Technical Specifications," Revision 1, March 2007.

8. NUREG/CR-5750, "Rates of Initiating Events at U. S. Nuclear Power Plants: 1987 – 1995," February 1999.

9. NUREG/CR-1570, "Risk Assessment of Severe Accident Induced Steam Generator Tube Rupture," March 1998.

10. NUREG/CR-6338, "Resolution of Direct Containment Heating Issue for all Westinghouse Plants with Large Dry Containments or Subatmospheric Containments," January 1996.

11. TSTF-448-A, "Control Room Habitability," Rev. 3, August 8, 2006 (ADAMS Accession Number ML062210095).

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June 15, 2010

OG-10-203

To: PWROG Licensing Subcommittee Representatives in PA-LSC-0364 Revision 2

Subject: PWR Owners Group
Final Safety Evaluation for PWR Owners Group (PWROG) Report WCAP-16125-NP, Revision 2, "Justification for Risk-Informed Modifications to Selected Technical Specifications for Conditions Leading to Exigent Plant Shutdown" (TAC No. MD8138). PA-LSC-0364

Enclosed for your review is the Final Safety Evaluation (SE) for WCAP-16125-NP, Revision 2, "Justification for Risk-Informed Modifications to Selected Technical Specifications for Conditions Leading to Exigent Plant Shutdown".

If you have any questions, please feel free to contact me at (412) 374-6230 or Ray Schneider at (860) 731-6461.

Regards,

Chad Holderbaum, Project Manager
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CMH:rfn

Enclosure (1)

Electronically Approved Records are Authenticated in the Electronic Document Management System

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