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).

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<br>2) Two CSS and two Containment Cooling System (CCS) trains inoperable [a] | None (3.0.3) | 1) 72 hours [b]<br>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 $\Delta$CDF, 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."

Attachment

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.