September 28, 2006

Mr. Rick A. Muench
President and Chief Executive Officer
Wolf Creek Nuclear Operating Corporation
Post Office Box 411
Burlington, KS 66839

SUBJECT: WOLF CREEK GENERATING STATION - ISSUANCE OF AMENDMENT RE: EXTENDED CONTAINMENT ISOLATION VALVE COMPLETION TIMES (TAC NO. MC3944)

Dear Mr. Muench:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 167 to Facility Operating License No. NPF-42 for the Wolf Creek Generating Station. The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated July 23, 2004 (WO 04-0030), as supplemented by letters dated August 11 (ET 06-0028) and September 22 (ET 06-0040), 2006.

The amendment revises TS 3.6.3, "Containment Isolation Valves," by (1) adding the abbreviation "(CIV)" for containment isolation valve in Condition A of the Actions for the Limiting Condition for Operation; (2) deleting the note and revising Condition A to be for only one penetration flow path with one CIV inoperable; (3) revising the completion time for Required Condition A.1 from 4 hours to as much as 7 days depending on the category of the inoperable CIV; and (4) revising Condition C to be for two or more penetration flow paths with one CIV inoperable. The amendment is based on Topical Report WCAP-15791-P, "Risk-Informed Evaluation of Extensions to Containment Isolation Valve Completion Times."

A copy of our related Safety Evaluation is enclosed. The Notice of Issuance will be included in the Commission's next biweekly *Federal Register* notice.

Sincerely,

/RA/

Jack Donohew, Senior Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-482

Enclosures: 1. Amendment No. 167 to NPF-42
2. Safety Evaluation

cc w/encls: See next page

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DATE	9/28/06	9/28/06	8/21/06	9/5/06	9/27/06	9/27/06

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WOLF CREEK NUCLEAR OPERATING CORPORATION

WOLF CREEK GENERATING STATION

DOCKET NO. 50-482

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 167
License No. NPF-42

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment to the Wolf Creek Generating Station (the facility) Facility Operating License No. NPF-42 filed by the Wolf Creek Nuclear Operating Corporation (the Corporation), dated July 23, 2004, as supplemented by letters dated August 11 and September 22, 2006, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this license amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and Paragraphs 2.C.(2) and 2.C.(15) of Facility Operating License No. NPF-42 are hereby amended to read as follows:

   2. <u>Technical Specifications</u>

      The Technical Specifications contained in Appendix A, as revised through Amendment No. 167, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated in the license. The Corporation shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

   15. <u>Additional Conditions</u>

       The Additional Conditions contained in Appendix D, as revised through Amendment No. 167, which are attached hereto, are hereby incorporated in the license. Wolf Creek Nuclear Operating Corporation shall operate the facility in accordance with the Additional Conditions.

3. The license amendment is effective as of its date of issuance and shall be implemented prior to the start of Refueling Outage 18, which is scheduled to start in spring 2008.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

David Terao, Chief
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: September 28, 2006

ATTACHMENT TO LICENSE AMENDMENT NO. 167

FACILITY OPERATING LICENSE NO. NPF-42

DOCKET NO. 50-482

Replace the following page of Appendix D with the attached page. The revised page is identified by an amendment number and contains marginal lines indicating the areas of change. The corresponding overleaf page is provided to maintain document completeness.

REMOVE	INSERT
Page 4	Page 4

Replace the following pages of the Appendix A Technical Specifications with the attached pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change. The corresponding overleaf pages are provided to maintain document completeness.

REMOVE	INSERT
ii	ii
3.6-7 to 3.6-20	3.6-7 to 3.6-20
3.6-21	------

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 167 TO FACILITY OPERATING LICENSE NO. NPF-42

WOLF CREEK NUCLEAR OPERATING CORPORATION

WOLF CREEK GENERATING STATION

DOCKET NO. 50-482

1.0 INTRODUCTION

By application dated July 23, 2004, as supplemented by letters dated August 11 and September 22, 2006 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML042160031, ML062290238, and MLxxxxxxxxx[1], respectively), Wolf Creek Nuclear Operating Corporation (the licensee) requested changes to the Technical Specifications (TSs, Appendix A to Facility Operating License No. NPF-42) for the Wolf Creek Generating Station (WCGS). The proposed amendment would revise TS 3.6.3, "Containment Isolation Valves," by (1) adding the abbreviation "(CIV)" for containment isolation valve in Condition A of the Actions for the Limiting Condition for Operation (LCO); (2) deleting the note and revising Condition A to be for only one penetration flow path with one CIV inoperable; (3) revising the completion time (CT) for Required Condition A.1 from 4 hours to as much as 7 days depending on the category of the inoperable CIV; and (4) revising Condition C to be for two or more penetration flow paths with one CIV inoperable.

The proposed CT changes are based on the Westinghouse Topical Report, WCAP-15791-P, Revision 1, "Risk-Informed Evaluation of Extensions to Containment Isolation Time Valve Completion Times," (WCAP-15791). The topical report was submitted to the Nuclear Regulatory Commission (NRC) for its review and approval by the Westinghouse Owners Group (WOG) by a letter dated May 6, 2004 (WOG-04-0234). The NRC approved the use on WCAP-15791 for plant licensing actions in its final safety evaluation (SE) issued March 10, 2006 (ADAMS Accession No. ML060330330), which contained regulatory commitments by, and conditions on, licensees adopting the topical report. These commitments and conditions will be addressed in this SE.

The supplemental letters dated August 11 and September 22, 2006, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination published in the *Federal Register* on December 7, 2004 (69 FR 70724).

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[1] As of the date of this SE, the September 22, 2006, letter had not been entered into ADAMS.

2.0 REGULATORY EVALUATION

In Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.36, "Technical specifications," the NRC issued a rule and established its regulatory requirements related to the content of TSs. In doing so, the NRC emphasized those matters related to the prevention of accidents and mitigation of consequences of such accidents. As stated in the Statements of Consideration, Technical Specifications for Facility Licenses: Safety Analysis Reports (33 FR 18610, December 17, 1968), the NRC noted that licensees are expected to incorporate into their plant TSs those items that are directly related to maintaining the integrity of the physical barriers designed to contain radioactivity. Pursuant to 10 CFR 50.36, TSs are required to include items in five specific categories related to station operation. Specifically, those categories include: (1) safety limits, limiting safety system settings (LSSSs), and limiting control settings; (2) LCOs; (3) surveillance requirements (SRs); (4) design features; and (5) administrative controls. However, 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)(i), the "Limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When [an LCO] of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the technical specification..." The remedial actions in the TSs are specified in terms of LCO conditions, required actions, and CTs to complete the required actions. When an LCO is not being met, the CTs specified in the TSs are the time allowed in the TSs for completing the specified required actions. The conditions and required actions specified in the TSs must be acceptable remedial actions for the LCO not being met, and the CTs must be a reasonable time for completing the required actions.

The Maintenance Rule 10 CFR 50.65, “Requirements for monitoring the effectiveness of maintenance at nuclear power plants,” requires that a licensee shall monitor the performance or condition of structure, systems, or components (SSCs) against licensee-established goals, in a manner sufficient to provide reasonable assurance that SSCs are capable of fulfilling their intended functions as applicable to the Implementation and Monitoring Program guidance of Regulatory Guide (RG) 1.174, Section 2.3, and RG 1.177, Section 3. In addition,10 CFR 50.65(a)(4) requires the assessment and management of the increase in risk that may result from the proposed maintenance activity.

General Design Criterion (GDC 35), "Emergency core cooling" of Appendix A to 10 CFR Part 50, requires suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities to assure that the system safety function can be accomplished assuming a single failure.

GDC 54, "Piping systems penetrating containment," requires those piping systems penetrating primary containment shall be provided with leak detection, isolation, containment capabilities having redundancy, reliability, and performance capabilities which reflect the importance to safety of isolating these piping systems.

GDC 55, "Reactor coolant pressure boundary penetrating containment," requires that each line that is part of the reactor coolant pressure boundary and that penetrates primary containment shall be provided with CIVs.

GDC 56, "Primary containment isolation," requires that each line that connects directly to the containment atmosphere and penetrates primary reactor containment shall be provided with CIVs.

GDC 57, "Closed system isolation valves," requires that each line that penetrates primary reactor containment and is neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere shall have at least one CIV which shall be either automatic, or locked closed, or capable of remote manual operation.

General guidance for evaluating the technical basis for proposed risk-informed changes is provided in Chapter 19, “Use of Probabilistic Risk Assessment [PRA] in Plant-Specific, Risk-Informed Decisionmaking: General Guidance,” of the NRC Standard Review Plan (SRP), NUREG-0800. More specific guidance related to risk-informed TS changes is provided in SRP Section 16.1, “Risk-Informed Decisionmaking: Technical Specifications,” which includes CT changes as part of risk-informed decisionmaking. Chapter 19 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 or rule change.
- 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 (CDF) 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.

RG 1.174, “An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis,” dated November 2002, describes a risk-informed approach, acceptable to the NRC, for licensees to assess the nature and impact of proposed permanent licensing basis changes by considering engineering issues and applying risk insights. In addition, RG 1.174 provides risk acceptance guidelines applicable to risk-informed decisionmaking.

RG 1.177, “An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications,” dated August 1998, identifies an acceptable risk-informed approach including additional guidance geared toward the assessment of proposed TS CT changes. Specifically, RG 1.177 identifies a three-tiered approach for the licensee's evaluation of the risk associated with a proposed CT TS change as shown below:

- Tier 1 assesses the risk impact of the proposed change in accordance with acceptance guidelines consistent with the Commission's Safety Goal Policy Statement, as documented in RG 1.174 and RG 1.177. The first tier assesses the impact on operational plant risk based on the change in CDF (ΔCDF) and change in large early release frequency (ΔLERF). It also evaluates plant risk while equipment covered by the proposed CT is out-of-service, as represented by incremental conditional core damage probability (ICCDP) and incremental conditional large early release probability (ICLERP). Tier 1 also addresses PRA quality, including the technical adequacy of the licensee's plant-specific PRA for the subject application. Cumulative risk of the present TS change in light of past (related) applications or additional applications under review are also considered along with uncertainty/sensitivity analysis with respect to the assumptions related to the proposed TS change.

- Tier 2 identifies and evaluates any potential risk-significant plant equipment outage configurations that could result if equipment, in addition to that associated with the proposed license amendment, are taken out of service simultaneously or if other risk-significant operational factors, such as concurrent system or equipment testing, are also involved. The purpose of this evaluation is to ensure that there are appropriate restrictions in place such that risk-significant plant equipment outage configurations will not occur when equipment associated with the proposed CT is implemented.

- Tier 3 addresses the licensee's overall configuration risk management program (CRMP) to ensure that adequate programs and procedures are in place for identifying risk-significant plant configurations resulting from maintenance or other operational activities and for identifying appropriate compensatory measures to avoid entering configurations that may not have been considered when the Tier 2 evaluation was performed. Compared with Tier 2, Tier 3 provides additional coverage to ensure risk-significant plant equipment outage configurations are identified in a timely manner and that the risk impact of out-of-service equipment is appropriately evaluated prior to performing any maintenance activity over extended periods of plant operation. Tier 3 guidance can be satisfied by the Maintenance Rule (10 CFR 50.65(a)(4)), which requires a licensee to assess and manage the increase in risk that may result from activities such as surveillance testing, corrective and preventive maintenance, subject to the guidance provided in RG 1.177, Section 2.3.7.1, and the adequacy of the licensee's program and PRA model for this application. The CRMP ensures that equipment removed from service prior to or during the proposed extended CT will be appropriately assessed from a risk perspective.

More specific methods and guidelines acceptable to the staff are also outlined in RG 1.177 for assessing risk-informed TS changes. Specifically, RG 1.177 provides recommendations for utilizing risk information to evaluate changes to TS CTs with respect to the impact of the proposed change on the risk associated with plant operation.

RG 1.174 and RG 1.177 also describe acceptable implementation strategies and performance monitoring plans to help ensure that the assumptions and analysis used to support the proposed TS changes will remain valid. The monitoring program should include means to adequately track the performance of equipment that, when degraded, can affect the conclusions of the licensee's evaluation for the proposed licensing basis change. RG 1.174 states that monitoring performed in accordance with the Maintenance Rule, 10 CFR 50.65, can be used when the monitoring performed under the Maintenance Rule is sufficient for the SSCs affected by the risk-informed application.

3.0 TECHNICAL EVALUATION

3.1 Proposed License Amendment

In its application and supplemental letter, the licensee proposed the following changes to TS 3.6.3:

1. Revise Condition A for one or more penetration flow paths with one CIV inoperable except for purge valve leakage not within limit to the following:

    a. Remove the note stating "Not applicable to penetration flow paths with two containment isolation valves."

    b. Add "(CIV)" as the abbreviation for "containment isolation valves."

2. Revise the completion time (CT) for Required Condition A.1 from 4 hours to as much as 7 days depending on the category of the inoperable CIV.

3. Replace existing Condition C, which is for one or more penetration flow paths with one CIV inoperable, with a new Condition C, which is for two or more penetration flow paths with one CIV inoperable.

The licensee provided a risk-informed justification for extending the CIV CTs from 4 hours up to 168 hours using the methodology presented in WCAP-15791. WCAP-15791 includes plant specific WCGS evaluations to justify CIV CT extensions of up to 168 hours. For WCGS CIVs that did not demonstrate acceptable results for a 168 hours extended CT, WCAP-15791 provides plant-specific WCGS CIV evaluations for CIV CTs of less than 168 hours.

3.2 Background

The safety function of CIVs is to help ensure that the containment will be isolated from the environment, within the time limits assumed in the safety analysis, should a fission product release into containment occur as a result of a design-basis accident (DBA). The DBAs that result in a release of radioactive material within containment are a loss-of-coolant accident (LOCA) and a rod ejection accident. The CIVs form part of the containment pressure boundary and provide a means to isolate containment penetration flow paths not serving accident consequence limiting functions with two isolation barriers that close on a containment isolation signal. The containment barrier can be passive or active. Passive barriers include manual valves, secured automatic valves, blind flanges, and closed systems. Check valves or automatic valves designed to close without operator action following an accident are considered active devices. Two barriers are provided in series for each penetration flow path to satisfy the single failure criteria such that no malfunction of an active barrier results in leakage exceeding the limits assumed in the safety analysis. Penetration flow paths may be open or closed. An open system can be directly connected to the containment atmosphere inside containment or directly connected to the outside environment. A closed system is one that is not directly connected to containment atmosphere inside containment or is not directly connected to the environment outside containment. The operability requirements for CIVs help ensure that containment is isolated within the time limits assumed in the safety analysis.

Various barriers collectively make up the containment isolation system. Isolation signals are comprised of phase signals initiated during accidents identified as phase “A” isolation and phase “B” isolation conditions. Phase signals are generated upon receiving a safety injection signal and isolates nonessential process lines to minimize fission product release. Phase “B” signals are generated on a containment High-3 signal and isolate the remaining process lines except for accident consequence limiting systems. In addition to the above, the purge and exhaust valves isolate upon receipt of a containment high radiation signal.

The licensee stated in its application that the current CIV CTs are generally insufficient to respond to CIV inoperability and perform preventive maintenance activities at power. The licensee stated that the proposed CIV CT extensions were requested primarily to provide an improvement to operational safety, reduce unnecessary burden, and provide a more consistent risk basis in regulatory requirements based on WCAP-15791. The reasons for the proposed change are consistent with RG 1.177, Section 1.1. In addition, the licensee stated that extended CIV CTs may avert risks associated with shutdown or transition risk.

3.3 Evaluation of Proposed Extended CIV CTs

In its justification of its proposed changes to the CIV CTs, the licensee presented what it called the traditional engineering or deterministic considerations and the impacts on plant risk. The NRC staff has addressed (1) the deterministic considerations of this amendment in Section 3.3.1 of this SE and (2) the impacts on plant risk in Section 3.3.2 of this SE.

3.3.1 Deterministic Review

In its deterministic review, the NRC staff evaluates the effect of the proposed amendment on the regulations related to SSCs affected by the amendment and, if the amendment is risk-informed, evaluates whether the amendment would compromise the fundamental safety principles on which the plant design is based. These are addressed separately below.

3.3.1.1 Effect on the Appropriate Regulations

In its application, the licensee is only proposing to change the conditions, required actions, and CTs in TS 3.6.3 for inoperable CIVs. The licensee is not proposing to change the design or function of any of the CIVs at WCGS.

The WCGS CIVs are addressed in Section 6.2.4 of the Updated Safety Analysis Report (USAR). Table 6.2.4-1 lists the containment piping penetrations and Figure 6.2.4-2 shows the containment pressure boundary with respect to the steam generator and associated systems. The licensee stated in its supplemental letter dated August 11, 2006, that TS 3.6.3 does not apply to the main steam safety valves, main steam isolation valves (MSIVs), main feedwater isolation valves (MFIVs), the associated by-pass valves, and steam generator atmospheric relief valves. This is because these valves are outside the containment pressure boundary for WCGS. The containment pressure boundary for WCGS against fission product release to the environment, shown in USAR Figure 6.2.4-2, is the inside of the steam generator tubes, the outside of the steam generator, and the outside of the lines from the steam generator shell side. This containment pressure boundary was established as the WCGS licensing basis when WCGS received its full power license on June 4, 1985, and the original TSs issued by the NRC did not list the MSIVs and MFIVs as CIVs. Therefore, as the licensee stated, the MSIVs and MFIVs are not included in TS 3.6.3.

3.3.1.2 Maintaining Safety Principles

In Section 2.2.1 of RG 1.174, the NRC staff stated that one aspect of its engineering evaluation of risk-informed changes to a plant's licensing basis is to show that the fundamental safety principles on which the plant design and operation is based are not compromised. In that section, the NRC staff listed defense-in-depth and safety margins as two fundamental safety principles. In its application, the licensee addressed the impact of the amendment on defense-in-depth and safety margins in the following manner.

3.3.1.2.1 Impact on Defense-in-Depth

The licensee addressed the following six factors from RG 1.174:

1. A reasonable balance among prevention of core damage, prevention of containment failure, and consequence mitigation is preserved.

The licensee stated that CIVs are part of the plant design to ensure containment integrity following an accident. The proposed longer CT for an inoperable CIV does not significantly degrade the ability of the containment barrier. The balance between prevention of core damage and prevention of containment and consequence mitigation is maintained. No new accident or transient is introduced with the amendment and the likelihood of an accident is not changed. The containment barrier is only one of the three barriers in plant design that protects against the release of radioactivity material to the public; the others being the fuel cladding and the RCS pressure boundary. The other two barriers are not affected by the amendment.

2. Over-reliance on programmatic activities to compensate for weaknesses in plant design.

The licensee stated that the WCGS plant design will not be changed by the amendment. All safety systems, including the CIVs, will still function in the same manner and with the same reliability. There will be no additional reliance on other systems, procedures, operator actions, or control processes. There are no compensating administrative controls.

3. System redundancy, independence, and diversity are maintained commensurate with the expected frequency and consequences of challenges to the system.

The licensee stated that the amendment has no impact on either the redundancy, independence, or diversity of the CIVs or on the ability of the plant to isolate containment penetrations with diverse systems. The redundant and diverse containment isolation designs will not be changed by the amendment. The CIVs are reliable components in the containment isolation design and will remain reliable after the amendment is approved. The CIVs affected by the amendment are not being replaced by new CIVs.

4. Defenses against potential common cause failures are maintained and the potential introduction of new common cause failure mechanisms is assessed.

The licensee stated that the defenses against common cause failures are maintained. The operating environment for these components will remain the same. The number, design, and types of CIVs will remain the same. Neither the CIVs nor the containment isolation design will be changed by the amendment. The licensee concluded that new common cause failure modes should not be expected. The isolation of all containment penetrations will remain single failure proof.

5. Independence of barriers is not degraded.

The licensee stated that the independence of the fuel cladding, RCS pressure boundary, and containment barriers that protect the public from radiological releases will be maintained by the amendment. The fuel cladding and RCS pressure boundary barriers are not affected by the amendment. Also, for the containment barrier, neither the CIVs nor the containment isolation design will be changed by the amendment. It is not expected that multiple systems will be out of service simultaneously during a time when only one CIV could be in an extended CT that could lead to degradation of these barriers and an increased risk to the public.

6. Defenses against human errors are maintained.

The licensee stated that no new operator actions are introduced by this amendment to maintain plant safety. There are no changes to current operating, maintenance, or test procedures because of the amendment. The increase in the CT for one inoperable CIV provides additional time to complete troubleshooting, test, and repair activities on the inoperable CIV to improve operator and maintenance personnel performance, and should result in reduce system re-alignment and restoration errors.

3.3.1.2.2 Impact on Safety Margins

In addressing the impact on safety margins, the licensee stated that the USAR safety margins are not impacted by the amendment. The amendment will not allow plant operation in a new configuration and isolation of all containment penetrations will remain single failure proof. CIV operation and testing requirements, and containment leakage requirements will not change by the amendment. In its application, the licensee addressed the no significant hazards consideration for the amendment and stated the following about the amendment not causing a significant reduction in the margin of safety: the proposed changes do not alter the manner in which safety limits, LSSSs, or LCOs are determined; the safety analysis acceptance criteria are not impacted by the changes; and the proposed changes will not result in plant operation in a configuration outside the design basis.

Also, as addressed in Section 3.3.2.4.2.4 of this SE, on Tier 3 and risk-informed configuration risk management, risk management actions are evaluated by the licensee for maintenance activities determined to be risk significant and these should act to maintain safety margins at the plant.

3.3.1.3 Deterministic Review Conclusions

The licensee will not be replacing any CIVs or changing the containment isolation design at WCGS with the amendment. As discussed above, the NRC staff concludes that the licensee has acceptably demonstrated that the amendment does not significantly impact the existing defense-in-depth and safety margins at WCGS. Based on this, the NRC staff concludes that because the amendment only changes the CIV CTs, the plant continues to meet the design criteria in GDC 35, 54, 55, 56, and 57.

3.3.2 Risk-Informed Review

The NRC staff has reviewed the licensee's analysis in support of its proposed license amendment described in the application dated July 23, 2004, and its response dated August 11, 2006, to the NRC staff's request for additional information (RAI). The NRC staff's evaluation in this section of this SE provides a detailed description of the licensee's proposed changes, the review methodology used by the NRC staff, the key information used in the NRC staff's review, the applicability of the proposed changes to the regulatory acceptance guidelines, and the NRC staff's findings.

3.3.2.1 Detailed Description of the Proposed Changes

WCAP-15791 provides a justification for extending the CIV CT from 4 hours to up to 168 hours. For CIVs that cannot demonstrate acceptable results for 168 hours, shorter times are considered in the WCGS WCAP-15791 analysis. The proposed CIV CT extension would revise TS 3.6.3, as follows.

- In WCAP-15791, TS 3.6.3, Condition A was split into two conditions, Condition A for a CIV pressure boundary intact and a new Condition B for a CIV pressure boundary not intact. WCGS modified Condition A and B to eliminate the need to differentiate between pressure boundary intact and not intact thereby eliminating the need for the new pressure boundary intact Condition B. The licensee identified 18 CIVs that had different CTs based on the integrity of the pressure boundary. The licensee chose to accommodate CIVs with different CTs in the TS by eliminating the TS pressure boundary determination and implementing a single more restrictive CIV CT applicable to either pressure boundary condition and, therefore, minimize burden on the control room staff.

- TS 3.6.3, Condition A was also revised to eliminate the note stating that Condition A is only applicable to penetration flow paths with two CIVs.

- The CT for TS 3.6.3, Condition A is revised by adding CIV categories 1 through 7, as listed in Tables D-1, D-2, and D3 of Appendix D to WCAP-15791.

  1. 4 hours for Category 1 CIVs
  2. 8 hours for Category 2 CIVs
  3. 12 hours for Category 3 CIVs
  4. 24 hours for Category 4 CIVs
  5. 48 hours for Category 5 CIVs
  6. 72 hours for Category 6 CIVs
  7. 168 hours for Category 7 CIVs

- TS 3.6.3 was also modified from the TS proposed by WCAP-15791 (draft Technical Specification Task Force Traveler 446 (TSTF-446)) in that the phrase "or more" is reinstated for TS 3.6.3 Condition A. This change provides consistency with the standard TS.

- TS 3.6.3, Condition C is deleted to be consistent with WCAP-15791, which evaluated the CT for each CIV individually in the penetration flow path. The licensee has added a new Condition C for 2 or more penetration flow paths with one CIV inoperable in each flow path. Condition C limits the CIVs in an extended CT consistent with the single extended CIV analysis of WCAP-15791. This new condition limits the extended CT of an inoperable CIV in more than one penetration flow path, as allowed by Note 2 to the TS 3.6.3 Actions table.

- Condition B remains "as is" with a CT of 1 hour.

  Of the conditions identified in LCO 3.6.3, the risk-impact of two CIVs inoperable in one or more penetration flow paths, was not evaluated by WCAP-15791. The CT for this configuration is generally limited by NUREG-1431 LCO 3.6.3 Condition B to a CT of 1 hour. This remains unchanged by WCAP-15791 in that the WCGS LAR does not propose to change the CT for this condition.

- In addition, the licensee also provides, for information, the following discussion to be added to the Required Action C.1 bases:

  > For subsequent containment isolation valve inoperabilities, the Required Action and Completion Time continue to apply to each additional containment isolation valve inoperability, with the Completion Time, based on each subsequent entry into the Condition consistent with Note 2 to the ACTIONS Table (e.g., for each entry into the Condition), the containment isolation valves(s) inoperable as a result of that entry shall meet the Required Action and Completion Time.

Systems used for accident mitigation that contain CIVs that also function as containment pressure boundaries were evaluated by WCAP-15791 only with regard to the valve impact on loss of containment isolation. CT limitations, with respect to accident mitigation system function, remain unchanged. In response to the NRC staff's RAI to WCAP-15791, the WOG evaluated the potential impact of the CT extensions on the availability of other mitigative functions and the corresponding impact on risk. The WOG results show that this impact is very small.

In support of WCAP-15791, the Nuclear Energy Institute (NEI) submitted Technical Specification Task Force (TSTF) 446, Revision 1, "Risk-Informed Evaluation of Extensions to Containment Isolation Valve Completion Times (WCAP-15791)," by letter dated January 31, 2005. TSTF-446 is under review by the staff but has not been approved. The licensee has proposed TS revisions similar to the TSTF but the TSTF is not referenced in the licensee's LAR. The acceptability of the TSs for the proposed CIV CTs was evaluated plant-specifically by the staff using the TS format suggested by WCAP-15791, the proposed TSTF, and associated WCAP-15791 and WCGS RAI responses.

3.3.2.2 Review of Methodology

In accordance with NRC SRP Chapter 19 and Section 16.1, the NRC staff reviewed the WCGS application incorporating WCAP-15791 using the three-tiered approach and the five key principles of risk-informed decisionmaking presented in RG 1.174 and RG 1.177 and the NRC staff's final SE conditions for WCAP-15791.

3.3.2.3 Key Information Used in the Review

Key information used in the NRC staff's review is contained in the NRC staff final SE on WCAP-15791, and Attachments II, III, and V of the licensee's application dated July 23, 2004, as modified by the licensee's RAI response submitted by letter dated August 11, 2006. The NRC staff also reviewed the licensee's individual plant examination (IPE) and individual plant examination of external events (IPEEE) and the associated staff safety evaluation reports (SERs) for the IPE and IPEEE.

3.3.2.4 Comparison Against Regulatory Criteria/Guidelines

The staff's evaluation of the licensee's proposed amendment to extend the CIV CTs up to 168 hours used the three-tier approach and the five key principles outlined in RGs 1.174 and RG 1.177, and are presented in the following sections.

3.3.2.4.1 Traditional Engineering Evaluation

The traditional engineering evaluation addresses key principles 1, 2, 3, and 5 of the NRC staff's philosophy of risk-informed decisionmaking, which concerns compliance with current regulations, evaluation of defense-in-depth, and evaluation of safety margins, and performance measurement strategies.

Key Principle 1: Compliance with Current Regulations

This is addressed in Section 3.3.1 of this SE.

Key Principle 2: Evaluations of Defense-in-Depth

This is addressed in Section 3.3.1 of this SE.

Key Principle 3: Evaluation of Safety Margins

This is addressed in Section 3.3.1 of this SE.

Key Principle 5: Performance Measurement Strategies - Implementation and Monitoring Program

RG 1.174 and RG 1.177 establish the need for an implementation and monitoring program to ensure that extensions to TS CIV CTs do not degrade operational safety over time and that no adverse degradation occurs due to unanticipated degradation or common cause mechanisms. An implementation and monitoring program is intended to ensure that the impact of the proposed TS change continues to reflect the reliability and availability of SSCs impacted by the change.

RG 1.174 states that monitoring performed in conformance with the Maintenance Rule, 10 CFR 50.65, can be used when the monitoring performed is sufficient for the SSCs affected by the risk-informed application.

The licensee stated that the containment isolation function is within the scope of the WCGS maintenance rule program. The licensee's RAI response stated that availability is defined as loss of isolation function based on TS 3.6.3, "Containment Isolation Valves." The maintenance rule reliability criterion is less than one functional failure per valve per an 18-month period. This criterion is monitored in accordance with plant procedure AP 29E-001 as part of the containment leakage measurement program. With respect to WCAP-15791, the key parameter to be monitored is the time the CIV is unavailable consistent with RG 1.177. No credit was taken for improved reliability based on the extended CT. The licensee established CIV maintenance unavailability criteria to ensure CIV unavailability remains consistent with the analysis in WCAP-15791. The yearly unavailability criterion is established based on the CT for each CIV category. The licensee will revise its procedures to incorporate CIV unavailability monitoring by CIV category. This is identified as a regulatory commitment (its second) by the licensee, and is addressed in Section 3.5 of this SE.

In addition, the application of the three-tiered approach in evaluating a TS-allowed CT change provides additional assurance that the change will not significantly impact the key principle of defense-in-depth.

The NRC staff finds that the licensee's implementation and monitoring program includes CIV unavailability monitoring and evaluation. As such, the licensee's program meets the intent of

RG 1.174 and RG 1.177 for an implementation and monitoring program and, therefore, satisfies additional information Item 6 of the NRC staff's final SE on WCAP-15791.

3.3.2.4.2 NRC Staff Technical Evaluation (PRA) - Key Principle 4

The licensee's proposed change to extend CIV CTs employs a risk-informed approach using risk insights to justify changes to the TS CIV CT. The risk metrics ΔCDF, ΔLERF, ICCDP, and ICLERP were used by the licensee to evaluate the impact of the proposed changes and are consistent with those presented in RGs 1.174 and 1.177.

The risk evaluation presented below addresses the staff's philosophy of risk-informed decisionmaking, that if the proposed changes result in a change in risk, then the increase should be small and consistent with the intent of the Commission's Safety Goal Policy Statement.

3.3.2.4.2.1 Applicability of WCAP-15791 to WCGS

In Sections 3.4, 3.5, and 3.6 of the NRC staff's final SE on WCAP-15791, the NRC staff identified regulatory commitments, conditions, and additional information that needed to be addressed in licensees' plant-specific application adopting WCAP-15791. These are addressed below.

<u>NRC Staff's Final SE on WCAP-15791 Regulatory Commitments</u>

To determine that WCAP-15791 is applicable to WCGS, the licensee addressed the issue of regulatory commitments in Section 3.4 of the NRC staff's final SE on WCAP-15791, as discussed below.

The RG 1.177 Tier 3 program ensures that while a CIV is in an LCO condition, additional activities will not be performed that could further degrade the capabilities of the plant to respond to a condition for which the inoperable CIV or system was designed to mitigate, and as a result, increase plant risk beyond that assumed by the TR analysis. A licensee's implementation of RG 1.177 Tier 3 guidelines generally implies the assessment of risk with respect to CDF. However, the proposed CIV CT impacts containment isolation and consequently LERF and ICLERP, as well as CDF. Because the equations in WCAP-15791 to determine the extended CIV CTs are based on the LERF and ICLERP metrics, the management of risk in accordance with 10 CFR 50.65(a)(4) for these extended CIV CTs must assess LERF and ICLERP. Therefore, a licensee's CRMP, including those implemented under the maintenance rule of 10 CFR 50.65(a)(4), must be addressed in the plant-specific submittal to explain how LERF/ICLERP is assessed and must be documented in the plant-specific applications as a regulatory commitment (i.e., included in the licensee's commitment tracking system in accordance with NEI 99-04, Revision 0, "Guidelines for Managing NRC Commitment Changes") in the licensees' plant-specific applications referencing WCAP-15791, as well as demonstrating PRA quality as part of the licensee's Tier 3 assessment. Since NUMARC 93-01 implements ILERP as the quantitative risk metric (i.e., based on a zero maintenance model) and RG 1.177 utilizes ICLERP (i.e., based on an average maintenance model), the licensees, in their implementation of WCAP-15791, will need to demonstrate the equivalence for Tier 3 decisionmaking.

This issue is addressed in Section 3.3.2.4.2.2 on Tier 1 and the PRA technical adequacy, Section 3.3.2.4.2.4 on Tier 3 considerations, and Section 3.5 on regulatory commitments of this SE. Based on these evaluations, the NRC staff concludes that the licensee has adequately addressed for WCGS the regulatory commitment identified in the NRC staff's final SE for WCAP-15791.

<u>NRC Staff's Final SE on WCAP-15791 Conditions</u>

Additionally, to determine that WCAP-15791 is applicable to WCGS, the licensee addressed Conditions 1 and 2 in Section 3.4 of the NRC staff's final SE on WCAP-15791, as discussed below.

Condition 1 of NRC staff final SE for WCAP-15791:

> WCAP-15791 is based on only one CIV that is in maintenance at any time. WCAP-15791states that it is not expected that multiple systems will be out of service simultaneously during extended CTs, but does not preclude the practice. Although TS LCO 3.6.3 Note 2 allows separate condition entry for each penetration flow path, proposed Condition D addresses an inoperable CIV in more than one penetration flow path and limits the CT to 4 hours. If the licensees' proposed TS change does not include this Condition D, then the licensee's application must verify that the potential for any cumulative risk-impact of failed CIVs and multiple CIV LCO entries has been evaluated and is acceptable. The licensee must confirm that its Tier 3 configuration risk management program, which is in accordance with 10 CFR 50.65(a)(4), will address the possibility of simultaneous LCO entries of inoperable CIVs in separate penetrations, such that this combination will not exceed the RG 1.174 and RG 1.177 acceptance guidelines confirmed by the analysis presented in WCAP-15791 and that defense-in-depth for safety systems is maintained.

Based on its review of the licensee's application and RAI responses, the NRC staff concludes that the licensee has incorporated proposed TS Condition D (as plant-specific Condition C), as discussed further below, and provided Tier 2 and 3 information as evaluated in Section 3.3.2.4.2.3 and 3.3.2.4.2.4 of this SE. Based on this, the NRC staff concludes that the licensee has adequately addressed Condition 1 of the final SE for WCAP-15791.

Condition 2 of NRC staff final SE for WCAP-15791:

> The existing and proposed TS 3.6.3 does not allow multiple simultaneous extended CIV CTs to occur for more than 4 hours, which is the existing CT for an inoperable CIV in the Standard Technical Specifications (STS) LCO 3.6.3 in NUREG-1431 for Westinghouse plants like WCGS and in the WCGS LCO 3.6.3. This is to meet the WCAP-15791 assumption that only one valve within a single penetration can be in maintenance at a time for more than the 4 hours allowed by the current STS LCO 3.6.3 Condition A (i.e., in an extended CIV CT). The existing STS LCO 3.6.3 Condition B, and the proposed

STS 3.6.3 Conditions A and D, assure that this assumption is being met. If the TSs do not prevent this case, then this case must be evaluated in the plant-specific application to demonstrate that the risk impact assumptions of ΔCDF, ΔLERF, ICCDP and ICLERP remain less than the RGs 1.174 and 1.177 acceptance guidelines, which is shown in WCAP-15791. Also, the plant-specific application must address if the position and operability of the remaining CIVs in the affected penetration flow path, or another penetration flow path, are confirmed before entering the extended CT for the inoperable CIV.

The licensee has addressed this condition by adopting proposed TS Conditions A and D and the incorporation of a regulatory commitment (the licensee's third commitment which was proposed as a license condition and is addressed in Sections 3.5 and 3.6 of this SE) to ensure that CIVs are in their correct positions prior to performing maintenance on a CIV. Therefore, the NRC staff concludes that the licensee has adequately addressed Condition 2 of the SER to WCAP-15791.

<u>NRC Staff's Final SE on WCAP-15791 Additional Information Needed</u>

Finally, Section 3.6 of the NRC staff final SE on WCAP-15791 included additional information items that must be addressed in the licensee's plant-specific application adopting WCAP-15791. This additional information is identified below, and in most cases will be addressed in other sections of this SE:

1. As stated in the NRC staff final SE on WCAP-15791: "Address how the general assumptions of WCAP-15791, which are listed in Section 3.2 in the [WCAP-15791] SE, are incorporated in the specific plant practices, procedures, TSs, and PRA."

   The licensee in its RAI response letter dated August 11, 2006, addressed each assumption identified in WCAP-15791. The licensee categorized the 13 general assumptions as either a statement of the analysis approach or an analysis assumption. Of the 13 items presented in WCAP-15791, the licensee determined that items 1, 3, and 6 were analysis assumptions to be confirmed on a plant-specific basis as shown below. This assessment is acceptable to staff.

   Assumption 1: Only one CIV is in maintenance with an extended CT at any one time.

   This assumption is addressed by the licensee by the proposed changes to TS 3.6.3 Conditions A and C, which limits a CIV in an extended CT to a single valve. Required Action C.1, which requires the isolation of all but one penetration flow path by the use of at least one closed and deactivated automatic valve, closed manual valve, or blind flange and has a stated CT of 4 hours. In addition, Note 2 allows separate condition entry with each inoperable CIV such that each entry shall meet the Required Action and CT. The licensee has also provided a basis revision consistent with the proposed TS change for information. Based on the above, the NRC staff concludes that the licensee has acceptably addressed Assumption 1.

Assumption 3: Before maintenance or corrective maintenance (repair) is performed on a CIV, the topical report evaluation assumes that the other CIV(s) in the penetration flow path have been checked to ensure they are in the proper position.

This assumption is addressed by the licensee by implementing a procedural requirement to confirm that the remaining CIV(s) are in their correct positions before maintenance is performed. This is identified by the licensee in a RAI response as a regulatory commitment (its third commitment which was proposed as a license condition and is addressed in Sections 3.5 and 3.6 of this SE). Based on this, the NRC staff concludes that the licensee has acceptably accounted for Assumption 3.

Assumption 6: Multiple systems are not expected to be out of service simultaneously during extended CTs.

To address this, a Tier 2 and Tier 3 evaluation is to be performed by the licensee to ensure that any risk significant plant configurations will be prevented. Based on the licensee's Tier 2 and Tier 3 evaluations, the NRC staff concludes that the licensee has acceptably addressed Assumption 6. See Sections 3.3.2.4.2.3 (Tier 2) and 3.3.2.4.2.4 (Tier 3) below of this SE.

Based on the above, the NRC staff concludes that the licensee in its application and RAI response letter has satisfied additional information Item 1 of the NRC staff final SE for WCAP-15791.

2. As stated in the NRC staff final SE on WCAP-15791: "Because not all penetrations have the same impact on ΔCDF, ΔLERF, ICCDP, or ICLERP, verify the applicability of WCAP-15791 to the specific plant, including verification that (a) the CIV configurations for the specific plant match the configurations in the topical report (TR) and (b) the risk-parameter values used in the TR are bounding for the specific plant. Any additional CIV configurations, CT extensions, or non-bounding risk parameter values not evaluated by the TR should be addressed in the plant-specific analyses. Note that CIV configurations and extended CTs not specifically evaluated by the TR, or non-bounding risk parameter values outside the scope of the TR will require NRC staff review of the specific penetrations and related justifications for the proposed CTs."

   The licensee's analysis is based on the plant-specific methodology presented in WCAP-15791 for WCGS. The methodology uses plant-specific input data and, therefore, only requires WCGS to confirm that the WCAP-15791 CIV configurations match those of WCGS but not that the plant-specific input parameters are bounded by the alternate generic analysis data of WCAP-15791. The licensee confirmed that the CIV penetration flow path configurations for WCGS are representative of those developed for WCAP-15791. For the plant-specific approach, all the penetration flow path configurations evaluated in the generic analysis of WCAP-15791 are re-evaluated using plant-specific information. All WCGS plant-specific penetration flow paths were matched to the corresponding configurations in WCAP-15791 and the CT evaluated.

Based on the above, the NRC staff concludes that the licensee has satisfied the additional information Item 2 of the NRC staff final SER for WCAP-15791.

3. As stated in the NRC staff final SE on WCAP-15791: "Confirm that the Tier 2 conclusion of the TR [WCAP-15791] (i.e., no Tier 2 requirements are needed) is applicable to the specific plant, or provide the plant-specific Tier 2 requirements needed for the plant." See Section 3.3.2.4.2.3 (Tier 2) of this SE.

4. As stated in the NRC staff final SE on WCAP-15791: "Because WCAP-15791 does not address Tier 3, each plant-specific application must address Tier 3 for the specific plant. The plant-specific application must discuss conformance to the requirements of the Maintenance Rule (i.e.,10 CFR 50.65(a)(4)), as the requirements relate to the proposed CIV CTs and the guidance contained in NUMARC 93.01, Section 11, as endorsed by RG 1.182, including verification that the licensee's maintenance rule program, with respect to CIVs, includes a LERF and ICLERP (i.e., ILERP as defined in NUMARC 93-01) assessment as part of the maintenance rule process." See Section 3.3.2.4.2.4 (Tier 3) of this SE.

5. As stated in the NRC staff final SE on WCAP-15791: "Verify that the technical adequacy of the plant-specific PRA for Tier 2 and 3 assessments is acceptable for this application in accordance with the guidelines given in RGs 1.174 and 1.177, which are identified in the 6 items listed in Section 3.3.1.1 of the SE to WCAP-15791. This includes a verification that external event risk, including seismic and fires, is bounded by the WCAP-15791 assumptions and will not have an adverse impact on the conclusions of the plant-specific analysis for extending the CIV CTs." See section 3.3.2.4.2.2 (PRA technical adequacy) of this SE.

6. As stated in the NRC staff final SE on WCAP-15791: "Address how plant-specific CIV reliability and availability are monitored and assessed at the plant under the Maintenance Rule (i.e., 10 CFR 50.65(a)(4)) to confirm that performance continues to be consistent with the analysis assumptions used to justify extended CIV CTs, including the assumptions in WCAP-15791." See Section 3.3.2.4.1 (Key Principle 5) and Section 3.3.2.4.2.4 (Tier 3) of this SE.

7. As stated in the NRC staff final SE on WCAP-15791: "The cumulative risk impact of the proposed CIV CT extensions must be addressed in the plant-specific application in accordance with the acceptance guidelines in RG 1.174. The cumulative risk impact must include both previous plant license changes and additional plant applications still under review." See Section 3.3.2.4.2.2 (PRA technical adequacy) of this SE.

8. As stated in the NRC staff final SE on WCAP-15791: "Because uncertainty due to plant PRA models is not addressed in WCAP-15791, the plant-specific applications must discuss uncertainties in the risk assessment." See Section 3.3.2.4.2.2 (PRA technical adequacy) of this SE.

9. As stated in the NRC staff final SE on WCAP-15791: "Address the plant-specific CRMP, including the Maintenance Rule program implemented under

10 CFR 50.65(a)(4), and explain how the LERF/ICLERP [are] assessed in the program. This assessment is to be documented in a regulatory commitment [by the licensee] in the plant-specific application." See Section 3.3.2.4.2.2 (PRA technical adequacy) of this SE.

3.3.2.4.2.2 Tier 1: PRA Capability and Insights

Tier 1 evaluates the impact of the proposed changes on plant operational risk based on the methodology presented in WCAP-15791, as applicable to WCGS. The NRC staff Tier 1 review involves the following two aspects: (1) evaluation of the validity of the WCAP-15791 methodology and WCGS PRA and its application to the proposed changes and (2) evaluation of the PRA results and insights based on the licensee's proposed application.

PRA Technical Adequacy

The objective of the PRA technical adequacy review by the NRC staff is to determine whether the WCAP-15791 methodology used in evaluating the proposed extended CIV CTs is of sufficient quality, scope, and level of detail for the application to WCGS. The NRC staff final SE for WCAP-15791 concluded that the risk-impact of the proposed 168 hour CIV CT, as estimated by ΔCDF, ΔLERF, ICCDP, and ICLERP, was consistent with the acceptance guidelines specified in RG 1.174, RG 1.177, and NRC staff guidance outlined in SRP Section 16.1 and Chapter 19. The NRC staff final SE also noted that to be within these guidelines, some CIV CTs, as evaluated by WCAP-15791, had to be less than 168 hours. WCAP-15791 showed calculations whereby shorter than 168-hour CTs were justified for certain groups of the CIVs, as listed in WCAP-15791 Tables D-1 and D-2. The NRC staff final SE found that the risk-analysis methodology and approach used by the WOG to estimate the risk impacts were reasonable and of sufficient quality.

WCAP-15791 Methodology

The methodology presented in WCAP-15791 is performed in two parts, the first part utilizes a deterministic review that determines the minimum hole size that results in a large release. Based on NRC staff discussions with the WOG and the WOG's RAI responses, WCAP-15791 reduced the containment hole size determined by the original evaluation to two inches. Penetration flow paths connected to the containment atmosphere (which excludes all reactor cooling system and steam generator connections) smaller than the minimum hole size are screened out of the total list of penetration flow paths (i.e., no further evaluation is made), and the screened penetrations are assigned the maximum CT of 168 hours. The second part evaluates penetrations larger than the minimum hole size using a probabilistic evaluation to verify the CT (i.e., 168 hour or shorter CT) is justified by the evaluation, based on the methodology presented in WCAP-15791.

Two analyses were performed for WCGS in WCAP-15791. The first analysis was a generic assessment to demonstrate the methodology is applicable to Westinghouse plants using generic data. The second analysis used the same methodology, but substituted WCGS plant-specific information and is the approach used by the licensee in the application for WCGS. In either case, the implementation and methodology are the same; the only difference being the data used in the evaluation. The steps in the evaluation are summarized below.

1. CIV penetration data collection.

2. Confirm analysis input parameters - This step confirms that the input parameters are conservative with respect to the WCGS plant-specific data.

3. Grouping - Each CIV penetration is grouped based on the penetration type (i.e., closed loop or open loop system), system connections, and whether it is connected to the reactor coolant system (RCS) or containment atmosphere. Based on the WCGS CIV penetration definition, connections to the steam generator were not considered.

4. Penetrations that are less than the 2-inch containment hole size are identified. These lines are considered of insufficient size to provide a large release, based on the WCAP-15791 methodology. CIV penetration line sizes that fit this category that are connected to the containment atmosphere (closed or open inside containment) are considered “small lines” and receive a CT of 168 hours.

5. CIV penetrations that do not screen out of the analysis are matched to the WCAP-15791 penetration groups. WCAP-15791 provides guidelines for performing the generic groupings. Each penetration group is evaluated based on the penetration configurations provided in WCAP-15791, which consider valve types, an open or closed CIV, and the application of common cause as required by WCAP-15791 methodology.

The licensee stated in its application that the applicability of WCAP-15791 to WCGS is provided in Section 9 of WCAP-15791 and a plant-specific analysis for WCGS is documented in Section 10 of the WCAP. The licensee’s plant-specific evaluations used WCGS specific values incorporated into the WCAP-15791 methodology to obtain plant-specific results for WCGS. Containment isolation valves with pipe diameters less than 2 inches were assigned a CT of 168 hours, consistent with the guidance of the WCAP-15791. For penetrations that did not screen out based on penetration hole size, the licensee calculated ICLERP and ΔLERF using WCGS plant-specific parameters. The CT was determined by either ICLERP or ΔLERF, whichever was the most limiting. For the proposed CIV CTs, ICLERP and ΔLERF are the controlling metrics.

WCAP-15791 did not evaluate the impact on CDF and ICCDP for extended CIV CTs. WCAP-15791 considered the containment isolation a function of containment response to an event and not the ability of the plant design to prevent or mitigate core damage. The staff requested the WOG in an RAI to evaluate the following conditions: (1) an evaluation if the impact on CDF for containment isolation configurations and systems associated with an accident mitigation function (2) the impact of an open system during maintenance activities, and (3) ICCDP assessment associated with valves that also have a safety function (in addition to primary containment isolation) that are in a closed position during maintenance. The WOG, in their RAI response, provided an evaluation for each of the above conditions that demonstrated that the acceptance guidelines of RG 1.174 and RG 1.177 (ΔCDF and ICCDP) would continue to be met.

The WCAP-15791 methodology originally assumed that the total CDF used in the analysis was based on plant-specific internal events CDF data bases, with the worst case value used in the analysis. Based on RAI responses, the WOG revised the total CDF used in the generic analysis to 1E-4/yr, thereby bounding external event contributions, as well. For WCGS, the licensee demonstrated that the total plant CDF remains less than 1E-04/yr.

The licensee also evaluated the CIVs and confirmed that there are no CIV isolation signals that will isolate accident mitigation systems (i.e., emergency core cooling system (ECCS), decay heat removal system) or support systems that would compromise the ECCS or decay heat removal accident mitigation functions.

WCGS PRA

The WCGS IPE was submitted to the NRC on September 28, 1992. The NRC staff SE concluded that the WCGS IPE was complete with regard to the information requested by Generic Letter (GL) 88-20, "Individual Plant Examination for Severe Accident Vulnerabilities," and that the IPE results were reasonable. The NRC staff SE dated November 18, 1996, identified the following two concerns:

1. A limited set (5) of human reliability analyses (HRAs) of calibration actions. Including refueling water storage tank level, which other IPEs have identified as a potentially significant event. However, the basis why these were the only events identified for analysis was not provided.

The licensee in its RAI response in its letter dated August 11, 2006, stated that a systematic analysis was performed to identify pre-initiators. The licensee (1) reviewed calibration and surveillance test procedures, and plant historical data, and (2) assessed the impact during plant operation. The licensee stated that the current WCGS PRA incorporates the pre-initiator events from this analysis.

2. The modeling errors associated with the actions that have to be performed within a very short time (e.g., times in the range of seconds to one minute).

The licensee provided a discussion in its RAI response on the treatment of time critical actions in the WCGS PRA addressing the concern identified in the IPE. The NRC staff in its IPE SE stated that it did not consider whether this shortcoming would have prevented the licensee from identifying a vulnerability.

The NRC staff also reviewed the WCGS IPEEE. The WCGS IPEEE was developed in response to Supplement 4 to GL 88-20. In its SE dated February 29, 2000, the NRC staff concluded that the WCGS IPEEE process was capable of identifying the most likely severe accidents and severe accident vulnerabilities and, therefore, met the intent of GL 88-20, Supplement 4. The NRC staff did not identify any findings associated with the IPEEE.

The licensee confirmed that all category A facts and observations (F&Os) from the industry peer review have been dispositioned and incorporated into the licensee’s PRA as required. The licensee stated that most of the category B F&Os have also been dispositioned and incorporated into the WCGS PRA. The licensee confirmed that the remaining category B F&Os do not impact the proposed CIV CT extension analysis results or the applicability of WCAP-15791 to WCGS. The WCAP-15791 methodology did not use the WCGS PRA model, but, did use plant-specific data from WCGS. The data used for the plant-specific analysis were taken from the WCGS PRA that underwent the industry peer review and is used for the Maintenance Rule, 10 CFR 50.65(a)(4), assessments.

The licensee’s process for PRA configuration and revision control is contained in procedure SADI-001, “Maintenance of the Wolf Creek PSA [probabilistic safety assessment] Model.” This procedure requires independent review of the model and associated documentation and approval.

The licensee stated that the current plant-specific WCGS PRA reflects the as-built, as-operated plant. The licensee identified one additional change not reflected in the WCGS PRA, which is the use of the Sharpe Station to support an extension of the WCGS emergency diesel generators (EDG) CT that was approved in Amendment No. 163 issued April 26, 2006. The licensee is currently incorporating the Sharpe Station into the WCGS PRA and safety monitor. In its letters submitted in support of Amendment No. 163, the licensee stated that the Shift Manager has several options available to evaluate the risk associated with changing conditions, including performing a qualitative assessment based on his/her knowledge and experience, using the safety monitor to assess the risk or requesting assistance from the PSA group.

The addition of the Sharpe Station has a beneficial impact on the PRA results for WCGS and, based on the licensee's RAI response, should provide a reduction in CDF and a minor reduction in LERF. In addition, the WCGS PRA will be modified to consider WCGS plant-specific CIV penetration flow path configurations, as evaluated in accordance with WCAP-15791. By the WCAP-15791 RAI response, dated February 13, 2004, the WOG demonstrated that WCGS is representative of Westinghouse plants with no unique features that would cause WCGS to be a risk outlier. A WOG cross-comparison with other similar Westinghouse plants indicates that the risk-significant parameters are typical.

A concern of the NRC staff in the implementation of WCAP-15791 is the licensee’s methodology for determining risk with respect to LERF and ICLERP under Tier 2 and Tier 3. Because the extended CIV CT influences LERF, a licensee’s CRMP needs to include a LERF/ICLERP assessment to ensure that risk significant configurations can be identified and evaluated during an extended CIV CT. For WCGS, the licensee, in its RAI response dated August 11, 2006, stated that although a number of CIVs are specifically modeled in the WCGS PRA, not all of the CIVs evaluated in WCAP-15791 are considered in the WCGS PRA. To address this PRA technical adequacy concern, the licensee stated, as a regulatory commitment (the licensee's first commitment which was proposed as a license condition and is addressed in Sections 3.5 and 3.6 of this SE), that it will revise the containment isolation fault tree to include: (1) at least one CIV from each penetration type greater than 2 inches, as applicable to WCGS, which will be used in the Tier 3 evaluation as a surrogate for any un-modeled CIV of that type or (2) include all CIVs greater than 2 inches, applicable to the implementation of the topical report to WCGS. The NRC staff considers the revision to the WCGS PRA as an upgrade and a significant change in the models capability as referenced in the American Society of Mechanical Engineers (ASME) PRA Standard, “Standard for Probabilistic Risk Assessment for Nuclear

Power Plant Applications," ASME RA-S-2002, dated April 2002, NEI 00-02, "Probabilistic Risk Assessment (PRA) Peer Review Process Guidance," Section 1.1, RG 1.200 for trial use, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Table B-1, Section 1.1, and SRP Section 19.1, "Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities."

Based on this, the licensee needs to initiate a peer review following the appropriate guidance once the upgrade to the WCGS PRA model is completed. In its letter dated September 22, 2006, the licensee identified the revision to the WCGS PRA and the performance of a peer review as a licensee regulatory commitment (the licensee's first commitment which was proposed as a license condition and is addressed in Sections 3.5 and 3.6 of this SE). Based on its review of the methodology of WCAP-15791 in conjunction with the modification of the WCGS PRA to include the CIV penetration flow path configurations applicable to WCGS for this LAR, the NRC staff concludes that the modification of the PRA provides an adequate means to evaluate LERF and ICLERP.

External Events

The licensee evaluated the proposed CIV CT extensions for their potential impact on external events including fire events, seismic events, and high winds, floods, and other (HWFO) events. To generically address external events, WCAP-15791 assumed that the total internal and external CDF was 1E-4/yr. WCAP-15791 also used a default value of 1E-4/yr for an estimated internal and external CDF for the WCGS plant-specific analysis.

Seismic Events

The licensee performed a reduced-scope seismic margins assessment (SMA) and, therefore, did not quantify a seismic CDF. No significant issues were identified by the NRC staff SE for the licensee's IPEEE; however, the IPEEE did identify improvements to be implemented. The licensee confirmed the improvements were completed as referenced by the NRC staff SE dated February 29, 2000, for the IPEEE, and the licensee's e-mail dated February 28, 2000 (ADAMS Accession No. ML003686707).

The IPEEE identified that all equipment and structures were screened out with a high confidence of low probability of failure (HPCLF) capacity of more than the 0.3g review level earthquake (RLE), except for a few SSCs (including the refueling water storage tank and turbine building), which were assigned an HCLPF of 0.2g. Although a few items could not be screened out at the RLE, all SSCs were screened out during the walkdown at the safe shutdown earthquake level (i.e., 0.12g for the plant site and 0.2g for the power block). The IPEEE assigned an HCLPF capacity of 0.2g for the primary and alternate safe shutdown paths with the caveat that a more rigorous evaluation of the identified lower-capacity SSCs could allow the HCLPF to be increased to the RLE.

To confirm that the seismic risk at WCGS is sufficiently small such that the WCGS total CDF is less than 1.0E-4/year, the NRC staff performed an independent simplistic calculation to estimate the magnitude of the seismic risk. The staff used the approximation method provided in a paper by Robert P. Kennedy entitled, "Overview of Methods for Seismic PRA and Margin

Analysis Including Recent Innovations." This approach uses the plant's HCLPF value that is determined by the licensee's SMA and the sites seismic hazard curve that is based on NUREG-1488, "Revised Livermore Seismic Hazard Estimate for Sixty-Nine Nuclear Power Plant Sites East of the Rocky Mountains," to derive an approximation of the magnitude of the risk associated with seismic events. The NRC staff's independent simplistic calculation, using a plant HCLPF value of 0.2g peak ground acceleration (PGA), estimated a seismic CDF of about 7E-6/ year, which is less than 1.0E-04/year.

Fire

The licensee used the Electric Power Research Institute (EPRI) Fire-Induced Vulnerability Evaluation methodology to quantify fire events. The fire CDF, as referenced in the IPEEE, was estimated at 7.5E-6 /year. The licensee stated that in 1998 the fire risk evaluation was updated using the same methodology, resulting in a fire CDF of 8.14E-6/year. The NRC staff's IPEEE SE concluded that fires both inside and outside containment were of little significance to containment performance.

HWFO Events, and Other External Events

The HWFO events evaluation confirmed that the plant conforms to the 1975 SRP criteria, including the assessment of changes since the issuance of the operating license. HWFO events were, therefore, screened as low risk significance. No vulnerabilities were identified in the IPEEE review.

Cumulative Risk

As discussed in RG 1.174, the evaluation of risk should include the risk impacts of the proposed license amendment application in light of past applications. Because the proposed CIV CT extensions were based on a generic methodology using plant-specific data, the licensee's PRA was not used to develop the individual CIV CTs and, therefore, the methodology did not specifically consider any past risk-informed applications implemented by the licensee or present applications under review.

The licensee identified additional changes to the current WCGS PRA model based on the implementation of an extended EDG CT that was approved by the NRC staff in Amendment No. 163, which was issued on April 26, 2006. As part of this amendment, the licensee now considers the Sharpe Station generator sets in support of the extended EDG CT. The licensee stated that it is in the process of modifying the WCGS PRA and safety monitor software to reflect this change. The licensee has stated that the incorporation of the Sharpe Station and extended EDG CT will provide a net reduction in the WCGS CDF and a minor reduction in LERF, with an insignificant impact on the proposed CIV CTs.

The licensee also evaluated previous risk-informed submittals that extended accumulator allowed outage time, reactor trip system (RTS), and engineered safety feature actuation system (ESFAS) CTs, surveillance test intervals and bypass test times, and risk-informed inservice inspections. The increase in ΔCDF for the accumulator CT was estimated generically to be 3.6E-8/year and, although not estimated, the corresponding ΔLERF was considered minimal.

The TS changes to RTS and ESFAS instrumentation ΔCDF and ΔLERF were 8.0E-7/year and 3.09E-8/year. The licensee also submitted a relief request dated February 15, 2001, and this inservice inspection submittal showed a very small impact on ΔCDF.

Therefore, based on the above, the NRC staff concludes that the cumulative ΔCDF and ΔLERF risk for WCGS (1) does not adversely impact the proposed CIV CT implementation and is consistent with the guidance of RG 1.174 and (2) meets the additional information Item 7 of the NRC staff final SE on WCAP-15791.

PRA Uncertainty

Based on the RAI responses for the NRC staff review of WCAP-15791, the WOG stated that the parameters used (e.g., valve failure rates and common-cause failure (CCF) values) were based on generic WOG plant PRA values. The estimates used were stated in WCAP-15791 to be the most conservative values obtained from the WOG plant-specific PRA models. Because of this, the WOG stated that the values used in the analysis are bounding and no data uncertainty analysis was required. Therefore, WCAP-15791 did not provide sensitivity studies with respect to the CT extension risk analysis. However, based on the use of bounding values for input parameters in WCAP-15791, the NRC staff concludes that a sensitivity analysis using an upper bound value should be inherent in the results.

Additional uncertainty due to plant PRA models is not addressed in WCAP-15791. Based on RAI responses for WCAP-15791, the WOG generic analysis assumes a total CDF of 1.0E-4/year to bound internal and external events. The NRC staff expects that the use of this total CDF will bound any impacts of modeling uncertainty with respect to any plant-specific application of WCAP-15791.

For the WCGS application, the licensee provided an evaluation in its RAI response dated August 11, 2006, and confirmed that the plant-specific evaluation for WCGS and the data used remain consistent with that used by the WCAP-15791 plant-specific analysis for WCGS.

Based on the above evaluation, the NRC staff concludes that the licensee has adequately addressed PRA uncertainty for its application of WCAP-15791 to WCGS.

Transition Risk (for CIV extended CT)

Transistion risk is the risk associated with the change in plant modes. The licensee did not provide a specific assessment of transition risk although WCAP-15791 qualitatively discusses transition risk as a potential reason to extend a CIV CT. The NRC staff notes that the additional benefit to transition risk would only occur when unscheduled corrective maintenance could not be completed within the proposed TS CT. For failures occurring during surveillance, transition risk should be considered, but this should have a limited impact on the analysis. With respect to the proposed extended CIV CT, the transition risk averted may provide a qualitative risk benefit, but is not credited or quantified in the risk evaluation performed by the licensee or by the WOG in WCAP-15791.

In addition, the extended CTs provide additional flexibility in the performance of preventive and corrective maintenance during power operation and with a reduced potential for plant shutdown and possible plant transients introduced by this reactor mode change.

3.3.2.4.2.3 Tier 2 - Avoidance of Risk-Significant Plant Configurations

A licensee should provide reasonable assurance that risk-significant plant equipment outage configurations will not occur when specific plant equipment is taken out of service in accordance with the proposed TS change. The licensee confirmed that no Tier 2 plant risk configurations exist that should be avoided when implementing an extended CIV CT.

In response to the NRC staff's RAI, dated February 13, 2004, on WCAP-15791, the WOG stated that the plant TSs should be revised to be consistent with the single inoperable extended CIV assumed in WCAP-15791. The proposed WCGS TSs, in the licensee's application, limits the TS condition entry to a single extended CIV CT such that multiple simultaneous inoperable CIVs, each with an extended CT, would not be allowed. This is consistent with the WOG RAI response.

The TS limitation to allow only one CIV in an extended CT provides additional means to limit the configuration risk and is consistent with the WCAP-15791 analysis and assumptions. The 4-hour CT of TS Condition C also makes it unlikely that more than one CIV would be in scheduled maintenance at any one time. The licensee stated that plant staff attempt to avoid conditions where scheduled activities would require a CIV isolation in the 4-hour CT. The CT limitation also limits Tier 2 to identifying CIV risk-significant configurations consistent with the assumptions of WCAP-15791.

In addition, the WCAP-15791 methodology includes assumptions that a licensee in a plant-specific application is required to meet when implementing WCAP-15791. These assumptions impact configuration risk and are the following:

- Only one CIV is in maintenance with an extended CT at any time.

- Before preventive maintenance or corrective maintenance (repair) is performed on a CIV, the TR evaluation assumes that the other CIV(s) in the penetration flow path have been checked to ensure they are in their proper position. This is a regulatory commitment, as identified by the licensee in its RAI response of August 11, 2006, and later expanded to include a peer review and proposed as a license condition in the supplemental letter dated September 22, 2006. See Sections 3.5 and 3.6 of this SE.

- Multiple systems are not expected to be out of service simultaneously during the extended CTs.

With the licensee implementing the above items through its license condition in its September 22, 2006, letter, the NRC staff does not have a disagreement with the licensee not identifying any Tier 2 risk significant outage configurations with regard to CIV extended CTs when a CIV is out of service consistent with the proposed TS change. The regulatory commitments and license conditions are addressed in Sections 3.5 and 3.6 of this SE.

Based on the above, and the WCGS incorporation of the proposed TS change, the NRC staff finds the licensee's Tier 2 analysis satisfies the guidance of RG 1.177 regarding Tier 2 considerations and, therefore, acceptably supports the implementation of the proposed amendment for extended CIV CTs up to 168 hours.

3.3.2.4.2.4 Tier 3 - Risk-Informed Configuration Risk Management

The licensee addressed its Tier 3 CRMP in its application and its supplemental letter dated August 11, 2006. The CRMP is implemented by the WCGS Operational Risk Assessment Program, Administrative Procedure (AP) 22C-003, consistent with the guidance of RG 1.177, Section 2.3.7.2, "Key Components of the CRMP," and is integrated into the plant programs implementing the Maintenance Rule, 10 CFR 50.65(a)(4). The assessment and management of the risk of various plant configurations is controlled and implemented through AP 22C-003, and follows the requirements of 10 CFR 50.65(a)(4) to assess and manage the increase in risk prior to performing maintenance and testing activities. Operational risk assessment of CDF and LERF are performed for activities within a weekly schedule and are evaluated using safety monitor software. Plant configuration ICCDP and ICLERP are evaluated using the guidance of NUMARC 93-01. Compensatory measures (i.e., risk management actions) are evaluated for maintenance activities determined to be risk significant. The risk assessment also considers added or emergent work or work activities that have experienced schedule delays. The schedule and the associated risk assessment are reviewed by the WCGS PRA group and approved by the plant manager.

Because an extended CIV CT influences LERF, a licensee's CRMP needs to include a LERF/ICLERP assessment to ensure that risk significant configurations can be identified and evaluated during an extended CIV CT. The licensee proposed to revise the containment isolation fault tree prior to implementation of WCAP-15791 to include the CIV configurations applicable to WCGS. The licensee has identified the revision to the WCGS PRA as a regulatory commitment. Using applicable penetration type surrogates in the Tier 3 evaluation or modifying the WCGS PRA to include the CIVs applicable to WCGS for this LAR provides a means to evaluate LERF and ICLERP. The licensee has identified the revision to the WCGS PRA and the performance of a peer review on WCGS updated PRA as a license condition, which is addressed in Section 3.6 of this SE.

Based on the licensee's conformance to the requirements of the Maintenance Rule, 10 CFR 50.65, meeting the guidelines of RG 1.177, including the assessment of LERF, and the above license condition, the NRC staff concludes that the licensee's Tier 3 program satisfies the guidance of RG 1.177 regarding Tier 3 considerations and, therefore, acceptably supports the implementation of the proposed amendment for extended CIV CTs up to 168 hours.

3.3.2.5 Comparison With Regulatory Guidance

Based on its review of the licensee's application and supplemental letter dated August 11, 2006, as discussed above, the NRC staff concludes that the licensee's proposed TS changes to provide an extended CIV CT meets the acceptance guidance of RG 1.174 and 1.177, and the guidance outlined in Chapter 19 and Section 16.1 of the NRC's SRP, NUREG-0800. As stated by the licensee, neither the design or the function of the CIVs that have CTs being changed by the proposed amendment is being changed in any way. Therefore, the proposed CIV CTs do not affect the design or function of these valves and compliance with the referenced GDCs is not changed by the proposed CTs. Because the basis for extending the CTs is shown to be acceptable, the NRC staff also concludes that 10 CFR 50.36 is also met.

3.3.2.6 NRC Staff Findings

Based on its evaluation, the NRC staff finds that the licensee has satisfied the intent of RG 1.174 (Section 2.2.3 and 2.5), and RG 1.177 (Sections 2.3.1, 2.3.2, and 2.3.3), and that the technical adequacy of the WCGS PRA, with consideration of the licensee's regulatory commitments, is sufficient to support the risk evaluation and the performance of the Tier 2 and Tier 3 assessments by the licensee that are needed for this amendment. Also, the NRC staff concludes the licensee's application and supplemental letter have addressed the conditions and additional information required by NRC staff's final SE for WCAP-15791.

The proposed TS change would have only a limited impact on the risk from external events. The current internal events CDF is estimated by the licensee to be 3.24E-5/year, the fire risk is estimated by the licensee to be 8.14E-6/year, and the staff independent estimate of the seismic CDF is about 7E-6/year. Based on the WCAP-15791 analysis assumption of 1E-4/year, the total internal and external CDF for WCGS is expected to be well within that assumed in WCAP-15791.

The cumulative risk for WCGS was evaluated by the licensee and was not found to be adversely impacted by the proposed CIV CT implementation. The licensee addressed uncertainty associated with the proposed CIV CT by confirming that the plant-specific evaluation for WCGS and the data used remain consistent with the WCAP-15791 analysis for WCGS.

Based on the above evaluation, the NRC staff concludes that (1) the risk impacts as estimated by the licensee are within the acceptance guidelines for RG 1.174 and 1.177 for the proposed extended CIV CTs and (2) the licensee has acceptably demonstrated that the regulatory commitments, conditions, and nine additional information items identified in the staff SE to WCAP-15791 have been satisfied.

3.4 Evaluation Conclusions

Therefore, based on its above evaluation, the NRC staff concludes that the licensee has shown that (1) the proposed changes to TS 3.6.3 meet the risk criteria in RGs 1.174 and 1.177, meet the appropriate regulations, and do not significantly impact the existing defense-in-depth and safety margins, and (2) it has met the conditions and information items specified in the NRC staff's final SE for WCAP-15791. Based on this, the NRC staff finds that the proposed changes are acceptable. Based on this, the NRC staff further concludes that the proposed amendment meets 10 CFR 50.36 and is, therefore, acceptable, except that the regulatory commitments need to be reviewed to determine if they should be license conditions. This acceptance of the amendment includes the page changes to page ii of the TS table of contents where the pages numbers for TSs 3.6.4 through 3.6.8 are being changed to account for the reduction in the number of pages for TS 3.6.3 because of the changes to TS 3.6.3. The changes to the table of contents is solely administrative and does not in any way change any requirements in the TSs. The review of the regulatory commitments and the determination of whether they need to be license conditions is addressed in Section 3.5 below of this SE.

3.5 Regulatory Commitments

The following regulatory commitments were identified by the licensee in Attachment III to its supplemental letters dated August 11, 2006:

1. The licensee stated that, prior to the start of refueling outage (RO) 16, it will revise the containment isolation fault tree model prior to utilization of the requested CIV CT extensions by either: (1) including CIV for at least one of each WCAP-15791 penetration type applicable to WCGS greater than 2 inches in diameter or (2) including all CIVs in the PRA and safety monitor associated with this LAR for penetrations greater than 2 inches in diameter. A peer review would be conducted of the containment isolation fault tree model with resolution of any Category A and Category B findings from the review completed during the peer review.

2. The licensee stated that, prior to the start of RO 16, it will revise the appropriate procedures such that CIV unavailability will be monitored by CIV category with the category defined by a common CT. Each CIV will be assigned to a Category 2 through 7 (Category 1 represents the original 4-hour CIV CT) consistent with TS 3.6.3 and the monitoring of unavailability time completed on a category basis.

3. The licensee stated that, prior to the start of RO 16, it will implement in its procedures the requirement to confirm that the remaining CIV(s) in the affected penetration(s) are in their correct position(s) prior to performing maintenance on a CIV.

The above three commitments are referred to in Sections 3.3.2.4.1, 3.3.2.4.2, and 3.3.2.4.2.2 of this SE.

Regulatory commitments are not regulatory requirements in that regulatory commitments are outside the license and cannot be enforced by the license. However, this does not mean that regulatory commitments addressed in an SE must be made license conditions such that they become requirements within the license. A license condition is a well defined and re-occurring future action (i.e., occurring after the amendment is approved) to be performed by the licensee within a well-defined period of time.

Where the regulatory commitment is not sufficiently important to the approval of the amendment, the NRC staff finds that reasonable controls for the licensee's implementation and subsequent evaluation of any changes to the above regulatory commitments are provided by the licensee's administrative processes, including its commitment management program. The NRC staff has agreed that NEI 99-04, Revision 0, provides reasonable guidance for the control of regulatory commitments made to the NRC staff. See Regulatory Issue Summary 2000-17, "Managing Regulatory Commitments Made by Power Reactor Licensees to the NRC Staff," dated September 21, 2000. The commitments will be controlled in accordance with the licensee's commitment management program in accordance with NEI 99-04. Any change to the regulatory commitments is subject to licensee management approval and subject to the procedural controls established at the plant for commitment management in accordance with NEI 99-04, which include notification of the NRC. Also, the NRC staff may choose to verify the implementation and maintenance of these commitments in a future inspection or audit.

In its review of the above three commitments, the NRC staff concluded that the second regulatory commitment listed above for this amendment is acceptable, but that the first and third commitments were more important and should be license conditions. This is addressed in Section 3.6 below of this SE.

3.6 License Conditions and Amendment Conclusion

Because of the importance of the first and third regulatory commitments to the basis for the NRC staff approving the amendment, the NRC staff requested that the licensee propose these two commitments as license conditions. In its supplemental letter dated September 22, 2006, letter, the licensee proposed the first and third commitments as license conditions to be added to Appendix D, "Other Conditions," of the license. Because the license conditions encompass the first and third regulatory commitments discussed in Section 3.5 of this SE, the NRC staff concludes that the license conditions and the amendment are acceptable.

The licensee identified changes to the TS 3.6.3 Bases in Attachment V to its application for the amendment. The NRC staff has reviewed these changes and does not disagree with the identified changes.

3.7 Amendment Implementation Date

The licensee has proposed to revise its implementation date for the one regulatory commitment, two license conditions, and amendment from 90 days after the date of issuance given in its application to prior to the start of RO 16 in its supplemental letter dated September 22, 2006. Since RO 15 is scheduled to begin in October 2006, RO 16 would begin in the spring of 2008. In its letter, the licensee stated that this implementation date is based on (1) the resources necessary to support the fall 2006 refueling outage, (2) the resources necessary to revise the containment isolation fault tree model and address any Category A and B findings from the peer review, and (3) the resources necessary to support currently scheduled projects during 2007. Because the licensee may not use the requested CIV CT extensions until the above regulatory commitments and the amendment are implemented, the NRC staff considers the licensee's proposed implementation date of prior to the start of RO 16 as an acceptable date.

4.0 <u>STATE CONSULTATION</u>

In accordance with the Commission's regulations, the Kansas State official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding (69 FR 70724, published December 7, 2004). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

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