WOLF CREEK
NUCLEAR OPERATING CORPORATION

Stephen E. Hedges
Vice President Oversight

March 4, 2008
WM 08-0003

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Subject: Docket No. 50-482: Revision 21 of the Wolf Creek Updated Safety Analysis Report

Gentlemen:

Pursuant to the updating requirements of 10 CFR 50.71(e), Wolf Creek Nuclear Operating Corporation (WCNOC) is providing its Updated Safety Analysis Report (USAR), Revision 21. This submittal satisfies the Final Safety Analysis Report (FSAR) updating requirements of the aforementioned regulation.

Attachment I to this letter provides information relative to changes in regulatory commitments. This information is provided in accordance with the guidance of Nuclear Energy Institute (NEI) 99-04, "Guidelines for Managing NRC Commitments," Revision 0, July 1999.

Attachment II to this letter describes specific technical changes that have been processed since issuance of the Updated Safety Analysis Report (USAR), Revision 20. In addition to these technical changes, several editorial changes have been made and are included in Revision 21.

Attachment III to this letter provides a discussion of changes made in Revisions 30 through 33 of the Technical Requirements Manual (TRM).

Enclosure I to this letter provides the CD-ROM submittal of the Wolf Creek Updated Safety Analysis Report (USAR), Revision 21. This submittal satisfies the Final Safety Analysis Report updating requirement of 10 CFR 50.71(e)(4).

Enclosure II to this letter provides a CD-ROM containing the station-controlled drawings that are considered incorporated by reference into the USAR. Per the guidance of Nuclear Energy Institute (NEI) document NEI 98-03, Revision 1, "Guidelines for Updating FSARs," the USAR figures that are identical to controlled drawings were relocated from the USAR in Revision 17. Enclosure II is considered sensitive unclassified information and therefore warrants withholding under 10 CFR 2.390.

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Enclosure III to this letter provides a CD-ROM containing the Fire Hazards Analysis and the Quality Program Manual, both of which are incorporated by reference into the USAR. Per the guidance of Nuclear Energy Institute (NEI) document NEI 98-03, Revision 1, "Guidelines for Updating FSARs", Chapter 17.2, Quality Assurance During the Operation Phase, was relocated from the USAR into the Quality Program Manual in Revision 21.

Enclosure IV to this letter provides those changes made to the Wolf Creek Generating Station (WCGS), Unit 1 Technical Requirements Manual (Revisions 30 through 33) and includes a List of Effective Pages. The WCGS TRM is incorporated by reference into the USAR.

WCNOC has historically submitted updates to the USAR on March 11 of each year to coincide with the date of issuance of the WCGS operating license and to comply with the requirements of 10 CFR 50.71(e)(4). WCNOC considers that submittals made prior to or on March 11 satisfy the requirements of 10 CFR 50.71(e)(4).

There are no commitments contained in this letter.

If you have any questions concerning this matter, please contact me at (620) 364-4190, or Mr. Richard Flannigan, Manager Regulatory Affairs at (620) 364-4117.

Sincerely,

Stephen E. Hedges

SEH/rlt

Attachment I – Commitment Changes
Attachment II – USAR Change Requests
Attachment III – Revisions to the Technical Requirements Manual (TRM)
Enclosure I – Updated Safety Analysis Report
Enclosure II – Updated Safety Analysis Report Controlled Drawings
Enclosure III – Updated Safety Analysis Report Fire Hazards Analysis and Quality Program Manual
Enclosure IV – TRM Replacement Pages

cc: E. E. Collins (NRC), w/a, w/e
V. G. Gaddy (NRC, w/a, w/e
B. K. Singal (NRC), w/a, w/e
Senior Resident Inspector (NRC), w/a, w/e

STATE OF KANSAS)
) SS
COUNTY OF COFFEY)

Stephen E. Hedges, of lawful age, being first duly sworn upon oath says that he is Vice President Oversight of Wolf Creek Nuclear Operating Corporation; that he has read the foregoing document and knows the contents thereof; that he has executed the same for and on behalf of said Corporation with full power and authority to do so; and that the facts therein stated are true and correct to the best of his knowledge, information and belief.

By____________________
Stephen E. Hedges
Vice President Oversight

SUBSCRIBED and sworn to before me this 4th day of March, 2008.

RHONDA L. TIEMEYER
MY COMMISSION EXPIRES
January 11, 2010

Rhonda L. Tiemeyer
Notary Public

Expiration Date January 11, 2010

COMMITMENT CHANGES

Commitment No.: 1997-113

Commitment Description: KG&E intends to continue the Quality First program. As in the past, all Quality First files will be retained and will be fully available for review by NRC investigators. In addition, KG&E will keep the commission aware of concerns the Quality First program receives on "wrong-doing" issues (such as intimidation/harassment, drug or alcohol abuse) where the initial review indicates that the concern may have some merit.

Change to Commitment: This commitment was archived in accordance with NEI 99-04, "Regulatory Commitment Management Guidance."

Reason for Change: At the time the commitment was made the Quality First program was a process used by KG&E that the NRC had indicated they thought was a good program. Since that time the program and guidance in this area have matured and now have other industry standards that, when met, will achieve the objectives of the original program. Programs in accordance with these guidelines will meet the intent of the commitment with appropriate access to information by the NRC.

The commitment is actually in three portions: record retention; availability to NRC review; and keeping the Commission aware of concerns brought to the employee concerns program. Regarding the records retention portion of the commitment:

1. Retaining all Quality First (Q-1) or ECP files indefinitely is not necessary based on regulatory requirements. No regulatory requirements apply to the retention of these confidential files.

2. NEI 97-05, "Nuclear Power Plant Personnel - Employee Concerns Program (ECP)-Process Tools in a Safety Conscious Work Environment", Revision 2, December 2003, provides guidelines on record retention.

   - Retention periods should be the minimum necessary until the concern is resolved and then as considered appropriate until all potential litigation on which the information has a bearing has been concluded.

   - The retention period for each type of records should be established and records destroyed as called for in the policy.

   - ECP practice at many licensees calls for records retention of between five and seven years.

3. NRC Directive 8.8, "Management of Allegations", Section (OAC) (B) (4) (f), page 17 states: "Closed allegation files should be held for 2 years, then retired to the NRC Archives. Files may be destroyed 10 years after cases are closed. The OAC should contact the Records Management Branch, Office of the Chief Information Officer, for transfer of files to the archives facility."

Regarding the second portion of the commitment, records availability to NRC review, the NRC is currently able to review any records as appropriate to their jurisdictional authority. Deleting this commitment will not change WCNOC's commitment to continue to work cooperatively with the NRC on any issue, including an employee concern.

The third portion of the commitment, keeping the Commission aware of concerns brought to the employee concerns program, has only been done, as applicable, when a concern may have been relevant to a NRC requirement. Employee concerns have been, and will continue to be, kept confidential by the company. Deleting this commitment will not decrease WCNOC's dedication to thoroughly investigate these employee concerns.

Review of the commitment in accordance with NEI 99-04 concluded that it was acceptable to change the commitment as described. However because the issue of addressing employee concerns is clearly an area that the NRC has an interest in the commitment change is included in this report.

USAR CHANGE REQUEST	DESCRIPTION
07-003	**REVISE THE USAR TO DELETE STATEMENT, "THERE IS NO DRAINAGE BETWEEN MAIN STEAM LINE COMPARTMENTS" FROM THE LAST PARAGRAPH ON PAGE 3B-9.**

Page: 3B-9

07-005	**REVISE USAR SECTION 15.7.4 REGARDING THE RELEASE PATHWAY OF RADIOLOGICAL CONSEQUENCES RESULTING FROM A POSTULATED FUEL HANDLING ACCIDENT INSIDE CONTAINMENT. THIS CHANGE REFLECTS THE CHANGES MADE BY TECHNICAL SPECIFICATION AMENDMENT 146.**

Page: 15.7-15 Page: 15.7-14 Page: 15.7-11 Page: 15.7-10

07-006	**REVISE THE USAR TO INDICATE THAT THE SPECIFIC ACTIVITY USED FOR ACCIDENT ANALYSIS RELEASES IS BASED ON OPERATING WITH 1% FUEL DEFECTS WHICH RESULTS IN A REACTOR COOLANT SYSTEM ACTIVITY LIMIT MORE LIMITING THAN TS 3.4.16. CHANGE DUE TO LICENSE AMENDMENT NO. 170. REVISION IS TO SECTION 11.1.3.**

Page: 11.1-2 Page: 11.1-1

Table: 11.1-5

07-007	**REVISE THE USAR TO CHANGE 3A-15 TO REFLECT THAT THE ALTERNATIVE PROGRAM TO ANSI N18.7-1976 DOES NOT APPLY TO THE EMERGENCY, ALARM RESPONSE, OFF-NORMAL, SEVERE ACCIDENT MANAGEMENT GUIDELINES, AND EMERGENCY PLAN IMPLEMENTING PROCEDURES. THESE PROCEDURES WILL CONTINUE TO BE REVIEWED EVERY TWO YEARS.**

Page: 3A-15

07-008	**REVISE THE USAR TO REFLECT FUEL CONFIGURATION CHANGES FOR REGION 18 (CYCLE 16 FRESH) FUEL ASSEMBLIES CONSISTENT WITH CONFIGURATION CHANGE PACKAGE (CCP) 011840, "CYCLE 16 RELOAD DESIGN CHANGES."**

Page: 4.2-16 Page: 4.2-15 Page: 4.2-13 Page: 4.1-3
Page: 4.2-20 Page: 4.2-14 Page: 4.2-12 Page: 4.2-11
Page: 4.1-4 Page: 4.1-2

Table: 4.3-1 Sheet: 1 Table: 4.1-1 Sheet: 4
Table: 4.1-1 Sheet: 3

Figure: 4.2-2E

USAR CHANGE REQUEST **DESCRIPTION**

07-009 **REVISE THE USAR TO INSERT "BEING" INTO SECTION 10.4.7.2.3. "BEING" SHOULD HAVE BEEN INCLUDED WITH THE CHANGES MADE IN USAR CHANGE REQUEST 99-051, BUT THE WORD WAS OMITTED.**

Page: 10.4-29

07-010 **REVISE THE USAR TO REMOVE REFERENCES TO THE SUPERINTENDENT OPERATIONS SUPPORT (TRAINING) POSITION BECAUSE THE POSITION HAS BEEN DISSOLVED. REMOVING A SHIFT MANAGER DUE TO HIS RETIREMENT.**

Page: 13.1-9 Page: 13.1-19 Page: 13.1-17

Figure: 13.1-2b

07-012 **REVISE THE USAR TO CHANGE THE CONTAINMENT RECIRCULATION SUMP LEVEL TRANSMITTERS TO DIFFERENTIAL PRESSURE TRANSMITTERS, CHANGE THE RANGE OF THE CONTAINMENT WATER LEVEL INDICATION, AND ADD THE ACCURACY OF THE RESIDUAL HEAT REMOVAL SUMP LEVEL INDICATOR DURING ACCIDENT CONDITIONS.**

Page: 18.2-58

Table: 7A-3	Sheet: 6.2	Table: 3.11(B)-10	
Table: 3.11(B)-3	Sheet: 34	Table: 3.11(B)-3	Sheet: 33

07-013 **REVISE THE USAR TO INCORPORATE EDITORIAL CORRECTIONS.**

Page: 3.0-xxi	Page: 6.2-77	Page: 9.0-x	Page: 9.0-vii
Page: 7.0-x	Page: 6.0-x	Page: 6.0-v	Page: 6.0-i
Page: 5.0-vi	Page: 9.0-xiii	Page: 3.0-xxiv	Page: 11.0-ii
Page: 3.0-xix	Page: 3.0-xvi	Page: 3.0-xv	Page: 3.0-xiv
Page: 3.0-xxi	Page: 3.0-xix	Page: 11.0-vi	Page: 10.4-40
Page: 3.0-xii	Page: 3.0-xi	Page: 3.0-viii	Page: 3.0-iii
Page: 6.2-81	Page: 5.0-ii	Page: 6.2-72	Page: 18.0-i
Page: 9.2-25	Page: 3A-61	Page: 3A-5	Page: 3A-4
Page: 9.1-47 to 70	Page: 13.2-21	Page: 1.2-6	Page: 9.0-xi
Page: 6.2-75	Page: 11.0-vii	Page: 18.0-iv	Page: 18.0-ii
Page: 15.0-xii	Page: 15.0-xi	Page: 15.0-vii	Page: 13.0-ii
Page: 13.0-iii	Page: 12.0-i	Page: 1.0-v	Page: 4.0-i

Table: 1.7-3	Sheet: 5	Table: 1.7-2	
Table: 1.7-2	Sheet: 2	Table: 1.7-2	Sheet: 6
Table: 1.7-2	Sheet: 7	Table: 1.7-3	
Table: 1.6-3	Sheet: 3	Table: 1.7-3	Sheet: 4
Table: 7.4-6		Table: 3.5-3	
Table: 6.2.1-56		Table: 13.1-1	Sheet: 3

USAR CHANGE REQUEST **DESCRIPTION**

Table: 13.1-1 Sheet: 4 Table: 1.7-3 Sheet: 2

Figure: 2.4-1 Figure: 15.6-38

07-014 **REVISE THE USAR TO CORRECT THE QUALIFICATION AND TRAINING REQUIREMENTS FOR THE RADIATION PROTECTION MANAGER IN TABLE 13.1-1. THE WORD "ADDITIONAL" IS BEING ADDED TO FIVE YEARS OF PROFESSIONAL EXPERIENCE.**

Table: 13.1-1 Sheet: 7

07-015 **REVISE THE USAR TO CHANGE THE AVAILABLE NET PRESSURE SUCTION HEAD FOR THE RESIDUAL HEAT REMOVAL AND CONTAINMENT SPRAY PUMPS AT THE BEGINNING OF THE RECIRCULATION MODE AS A RESULT OF THE MEASURED HEAD LOSS ACROSS THE CLEAN CONTAINMENT RECIRCULATION SUMP STRAINERS.**

Table: 6.2.2-7 Table: 6.3-1 Sheet: 2

07-016 **REVISE THE USAR TO ABANDON IN PLACE TWO INCORE THERMOCOUPLES (BBT/C0011 AND BBT/C0035). ALSO CHANGE THE TOTAL NUMBER OF THERMOCOUPLES FROM 50 TO 47 DUE TO PREVIOUSLY REMOVED THERMOCOUPLES.**

Page: 18.2-85 Page: 18.2-72

Table: 3.11(B)-3 Sheet: 19 Table: 3.11(B)-3 Sheet: 17
Table: 7A-3 Sheet: 1.3

07-018 **REVISE THE USAR TO INCORPORATE REFERENCE TO NRC LETTER 89-00605, TMI ACTION PLAN ITEM II.K.3.5, REACTOR COOLANT PUMP TRIP, ON PAGE 18.2-102.**

Page: 18.2-102

07-019 **REVISE THE USAR TO CHANGE PAGES 13.1-5, 13.1-6 AND 18.1-16. PAGE 13.1-5, SECTION 13.1.1.2.1, CHANGE "THROUGH SITE MANAGEMENT" TO "THROUGH VICE PRESIDENT ENGINEERING AND VICE PRESIDENT OVERSIGHT." PAGE 13.1-5, SECTION 13.1.1.2.3, ADD ON ISEG FUNCTION AS A RESPONSIBILITY. PAGE 13.1-6, SECTION 13.1.1.2.5, ADD ISEG FUNCTIONS AS A RESPONSIBILITY. PAGE 18.1-16, SECTION 18.1.7.2, CHANGE "SITE MANAGEMENT IS" TO "THE VICE PRESIDENT ENGINEERING AND VICE PRESIDENT ARE..."**

Page: 18.1-16 Page: 13.1-6 Page: 13.1-5

USAR CHANGE REQUEST **DESCRIPTION**

07-020 **REVISE THE USAR TO REFLECT ORGANIZATIONAL CHANGES. THE MANAGER CHEMISTRY/RADIATION PROTECTION POSITION WAS VACATED BY THE INDIVIDUAL TAKING THE INPO ASSIGNMENT. THE CURRENT MANAGER OPERATIONS RELOCATED TO THE MANAGER CHEMISTRY/RADIATION PROTECTION POSITION. THE CURRENT SUPERINTENDENT OPERATIONS RELOCATED TO THE MANAGER OPERATIONS POSITION. ONE OF THE CURRENT SHIFT MANAGER'S RELOCATED TO THE SUPERINTENDENT OPERATIONS POSITION.**

Page: 13.1-19 Page: 13.1-20 Page: 13.1-18 Page: 13.1-17
Page: 13.1-16

07-021 **REVISE THE USAR TO REFLECT THE FOLLOWING ORGANIZATIONAL CHANGE:THE CURRENT VICE PRESIDENT OPERATIONS/PLANT MANAGER IS NOW THE VICE PRESIDENT OVERSIGHT AND THE CURRENT VICE PRESIDENT OVERSIGHT IS NOW THE VICE PRESIDENT OPERATIONS/PLANT MANAGER.**

Page: 13.1-7

07-023 **REVISE THE USAR TO ADD BRAD NORTON AS THE MANAGER INTEGRATED PLANT SCHEDULING.**

Page: 13.1-21

07-024 **REVISE THE USAR TO CHANGE THE STEAM GENERATOR FEEDWATER HYDRAZINE LOW SETPOINT IN TABLE 9.3-5 FROM 100 PPB TO 60 PPB TO BE IN AGREEMENT WITH THE CURRENT CHEMISTRY SPECIFICATION AS GIVEN IN AP 02-003, CHEMISTRY SPECIFICATION MANUAL.**

Table: 9.3-5

07-025 **REVISE THE USAR TO REFLECT THE 2007 INSTALLATION OF THE UPGRADED PLANT PROCESS COMPUTER.**

Page: 8.3-36 Page: 12.3-30 Page: 9.4-53 Page: 9.4-48
Page: 8.2-6 Page: 8.1-15 Page: 7.7-15 Page: 7.7-11
Page: 7.7-35 Page: 7.5-8 Page: 7.3-10 Page: 5.2-42
Page: 5.2-41 Page: 2.3-40 Page: 1.2-15

Table: 7A-3 Sheet: 17.5(CONT) Table: 7A-3 Sheet: 17.5
Table: 7A-3 Sheet: 12.3 Table: 7A-3 Sheet: 4.3
Table: 7A-3 Sheet: 2.8 Table: 7A-3 Sheet: 1.1
Table: 7.5-2
Table: 7.5-1

USAR CHANGE REQUEST	DESCRIPTION
07-026	REVISE THE USAR TO CHANGE TABLE 12.5-1 TO ELIMINATE INSTRUMENTATION AND TO UPDATE AVAILABLE QUANTITIES OF INSTRUMENTS AND EQUIPMENT TO REFLECT CURRENT NUMBERS AND PRACTICES.

Table: 12.5-2 Sheet: 2 Table: 12.5-2
Table: 12.5-1

07-027	REVISE THE USAR TO UPDATE TABLE 3.11(B)-3, SHEET 35, AND TABLE 9.5.5-1, SHEET 2, TO REFLECT EMERGENCY DIESEL INTERCOOLER RE-TUBING OF HEAT EXCHANGERS EKJ03A AND EKJ03B.

Table: 9.5.5-1 Sheet: 3 Table: 7.4-6 Sheet: 23
Table: 9.5.5-1 Sheet: 2 Table: 3.11(B)-3 Sheet: 35

07-028	REVISE THE USAR TO REFLECT THE INSTALLATION OF NEW WASTE GAS ANALYZER RACKS (HA162). THE RACKS ANALYZE THE OXYGEN AND HYDROGEN CONTENT OF THE GAS ENTERING AND EXITING HYDROGEN RECOMBINER SHA01B.

Page: 11.3-10

Figure: 11.3-4

07-029	REVISE THE USAR TO ALLOW THE USE OF CODE CASES N-411, N-318-3, N-391, N-392, AND 1606-1. THE CODE CASES ARE APPLICABLE TO WOLF CREEK AND ARE ALLOWED UNDER REG GUIDE 1.84, REVISION 31.

Page: 5.2-45 Page: 5.2-2

Table: 3.9(B)-7

07-030	REVISE THE USAR TO REMOVE INCORRECT SENTENCES REGARDING FIGURE 8.3-6. ONE SENTENCE IS UNASSOCIATED WITH ITS REFERRAL IN THE USAR AND ONE SENTENCE REFERS TO FIGURE 8.3-6, SHEET 1 THAT WAS INCORPORATED BY REFERENCE IN REVISION 17.

Page: 8.1-3 Page: 8.3-38 Page: 8.3-22

Table: 8.3-4

USAR CHANGE REQUEST **DESCRIPTION**

07-031 **REVISE THE USAR TO REFLECT TOM DOUGAN REPLACING JOHN FLETCHER AS THE SUPERVISOR QUALITY ASSURANCE.**

Page: 13.1-16

07-032 **REVISE THE USAR TO ADD PRESTON LAWSON AS A NEW SHIFT MANAGER IN SECTION 13.1.**

Page: 13.1-19

07-033 **REVISE THE USAR TO UPDATE TEXT IN SECTION 15.6.3.1.2 TO REFLECT ASSUMPTIONS USED IN THE STEAM GENERATOR TUBE RUPTURE ACCIDENT ANALYSIS DOCUMENTED IN CALCULATION AN-99-1025, REV. 1.**

Page: 15.6-11 Page: 15.6-10

07-034 **REVISE THE USAR TO REFLECT THE INSTALLATION OF WELD OVERLAYS ON THE PRESSURIZER (TBB03) SPRAY, RELIEF, SAFETY AND SURGE NOZZLES.**

Page: 5.2-45 Page: 5.4-60 Page: 5.4-49

Table: 5.2-2 Sheet: 2

08-001 **REVISE THE USAR TO REMOVE CHAPTER 17.2 FROM THE USAR AND TO INCORPORATE IT BY REFERENCE. INCLUDE A CROSS-REFERENCE TO SHOW WHERE THE SECTIONS OF 17.2 ARE LOCATED IN THE NEW QUALITY PROGRAM MANUAL. REMOVE REFERENCES TO CHAPTER 17.2 THROUGHOUT THE USAR AND REPLACE WITH REFERENCES TO QUALITY PROGRAM MANUAL.**

Page: 17.2-0	Page: 17.0-i	Page: 17.0-0	
Page: 14.2-16	Page: 14.2-12	Page: 14.2-8	Page: 13.4-2
Page: 13.4-1	Page: 13.1-5	Page: 7.1-9	Page: 3A-15
Page: 3A-14	Page: 1.6-1	Page: 17.2-1 to 55	

Table: 17.2-3 Table: 17.2-2
Table: 17.2-1

Figure: 17.2-1

REVISIONS TO THE TECHNICAL REQUIREMENTS MANUAL (TRM)

1. Technical Requirement (TR) Bases, TRB 3.4.17, Required Actions A.1 and A.2, were revised to clarify the determination of component or system OPERABILITY when a through-wall flaw is identified. Regulatory Issue Summary (RIS) 2005-020, "Revision to Guidance Formerly Contained in NRC Generic Letter 91-18, Information to Licensees Regarding Two NRC Inspection Manual Sections on Resolution of Degraded and Nonconforming Conditions and on Operability," was issued September 26, 2005. WCNOC incorporated the guidance of RIS 2005-020 into the Technical Requirements Manual (TRM) with the issuance of Revision 27 (DRR 06-0799) on May 18, 2006. An event on October 7, 2006 and a second event on January 3, 2007 (PIR 2007-000022), identified an inconsistency in the RIS 2005-020 guidance of Section C.11, Flaw Evaluation, and Section C.12, Operational Leakage From Code Class 1, 2, and 3 Components, and this guidance was incorporated into the TR 3.4.17 Bases.

   The change specifies that only the component is initially required to be declared inoperable when pressure boundary leakage is occurring. If a reasonable expectation of OPERABILITY does not exist for the system, the system should also be declared inoperable. Further flaw characterization and evaluation is necessary and if the evaluation determines the flaw does not meet structural integrity requirements, the system is declared inoperable. The system may be considered OPERABLE with the component isolated as long as the system is capable of performing its specified safety function. The proposed change is consistent with the intent of the guidance in RIS 2005-020.

2. Technical Surveillance Requirement (TSR) 3.4.3.3 Bases is revised to include a discussion of the Note to the TSR. During the development of the TRM for consistency with the improved Technical Specification format, the TRM format was also revised and detailed Bases were developed for the TRM as well. In the Improved Standard Technical Specification Writers Guide, there is no specific discussion about ensuring that the Notes in the TSs are discussed in the TS Bases (WCNOC utilizes the same format/content guidance for the TRM). However, in general, it is believed that it was the intent to include in the TR Bases discussions on Notes in the TR. TSRs 3.4.3.1 and 3.4.3.2 have similar Notes in the TSR and the Notes are discussed in the TR Bases. Adding a discussion regarding the use of the Note is considered an administrative change as the TR Bases should contain sufficient information to ensure an adequate basis for an understanding of the requirements.

3. TSR 3.3.16.3 Bases are revised to incorporate changes based on the implementation of Design Change Package (DCP) 10399, "Replace HA161 & HA162 Gas Analyzer Racks." The Waste Gas Hydrogen and Oxygen Analyzers that are part of the Gaseous Radwaste System (GRWS), located in the Radwaste Building, have had maintenance and design issues that are causing system operational problems. The analyzers were not designed to compensate for the presence of Helium (a by product of the neutron absorption of Boron 10) in the sample and consequently will not read correctly. The helium increase over the original design is due to the extended 18-month cycle and the increase in the reactor coolant system (RCS) boron concentration to support the cycle. As a result, the analyzers read conservatively high. This conservatively high reading makes the system hard to operate during a Volume Control Tank (VCT) purge and during Waste Gas System Hydrogen recombination. Operations has had to maintain personnel in constant

attendance when recombining gas from the VCT. To correct this problem, new flow panels and new analyzers that automatically compensate for helium have been installed in Gas Analyzer Racks HA161 and HA162.

4. TR 3.7.20 and associated Bases is being revised to incorporate changes for the addition of new Technical Specification Limiting Condition for Operation (LCO) 3.0.8 approved in Amendment No. 173. Amendment No. 173 modifies the TS requirements for inoperable snubbers by adopting the NRC-approved Revision 4 to Technical Specification Task Force (TSTF) Standard Technical Specification (STS) Change Traveler TSTF-372, "Addition of LCO 3.0.8, Inoperability of Snubbers." The amendment adds (1) a new LCO 3.0.8 addressing situations where one or more required snubbers are unable to perform their associated support function(s) (i.e., the snubber is inoperable) and (2) a reference to LCO 3.0.8 in LCO 3.0.1, which describes when LCOs shall be met. Changes to the TR and TR Bases are being made consistent with the approval of Amendment No. 173.

5. Technical Requirement Manual (TRM) Revision 27 revised TR 3.4.17, "Structural Integrity," and associated TR Bases based on the guidance in Regulatory Issue Summary (RIS) 2005-20: Revision to Guidance Formerly Contained in NRC Generic Letter 91-18, "Information to Licensees Regarding Two NRC Inspection Manual Sections on Resolution of Degraded and Nonconforming Conditions and on Operability." RIS 2005-020 was issued to inform licensees that it has revised the guidance in NRC Inspection Manual, Part 9900, Technical Guidance – this guidance supersedes the guidance previously provided in Generic Letter 91-18 and Revision 1 to Generic Letter 91-18. Appendix C, Section C.11 (Flaw Evaluation) and Section C.12 (Operational Leakage From Code Class 1, 2, and 3 Components) of Part 9900 primarily impact TR 3.4.17. TR 3.4.17 and the associated TR Bases were revised to reflect the revised guidance provided in Part 9900. Specifically, this revision required, in part, declaring the component inoperable upon identification of a through wall leak. Revision 30 to the TRB 3.4.17, Required Actions A.1 and A.2, clarified the determination of component or system OPERABILITY when a through-wall flaw is identified.

   Subsequently, on June 22, 2007, an NRC memorandum entitled, "Inspection Manual Part 9900 Operability Guidance Involving Structural Integrity of ASME Code Class 2 and 3 Piping," was issued providing guidance that indicates that the NRC agrees with the past position (pre-RIS 2005-20) on assessing OPERABILITY of ASME Code Class 2 and 3 piping with through wall leakage. TR 3.4.17 and associated Bases were revised based on the guidance in the NRC memorandum.

6. TR 3.4.16, "RCS Chemistry," and associated Bases were revised to provide an exception to allow TSR 3.4.16.1 to not be performed until a time period after entry into MODE 6 from defueled. A Note is added to TSR 3.4.16.1 indicating that the surveillance is not required to be met for Chloride and Fluoride until 72 hours after entry into MODE 6 from a defueled state and footnote (b) is added to Table TR 3.4.16-1 to indicate that the limits are not applicable with no fuel in the reactor vessel. A 72-hour exception is reasonable based on the Frequency of the TSR being 72 hours. The word "Applicable" in Table TR 3.4.16-1 was misspelled and is being corrected.

   Additionally, the Frequency for TSR 3.4.16.1 is revised from "Once per 72 hours" to "72 hours" per Section 1.4, "Frequency." The use of "once" implies a one-time performance Frequency and does not qualify for the 25% extension allowed by SR 3.0.2 as discussed in Example 1.4-2. The Frequency specifies an interval (72 hours) during which the associated TSR must be performed at least once.

The RCS water chemistry is selected to minimize corrosion. A periodic analysis of the coolant chemical composition is performed to verify that the reactor coolant quality meets the requirements of TR 3.4.16. This TR places limits on the dissolved oxygen, chloride and fluoride content of the RCS to minimize corrosion. Both chlorides and fluorides have been shown to cause stress corrosion if present in the RCS in sufficiently high concentrations at high pressure and temperature conditions. Stress corrosion can lead to either localized leakage or catastrophic failure of the RCS. The associated effects of exceeding the dissolved oxygen, chloride, and fluoride limits are time and temperature dependent. Corrosion studies show that operation may be continued with contaminant concentration levels in excess of the steady state limits, up to the transient limits, for the specified limited time intervals without having a significant effect on the structural integrity of the RCS. When the RCS is drained to a reduced inventory condition with all fuel removed from the reactor vessel and all loops drained, a representative sample of the RCS for chlorides and fluorides is not possible. Sampling is not possible since sample flow through the RCS loop sampling taps or instrument taps is not possible. Fluoride and chloride concentration limits are based on corrosion at high temperature (> 500°F), and are not a concern from a materials standpoint at refueling temperatures of less than 150°F.

Subject
Enclosed is the CD-ROM submittal of the Wolf Creek Updated Safety Analysis Report (USAR), Revision 21.

Contact

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Document Components:
The CD-ROM labeled "Wolf Creek Updated Safety Analysis Report" contains the following files:

001_NUSAR.pdf	2.52 MB, publicly available
002_NUSARC01.pdf	661 KB, publicly available
003_NUSARC02.pdf	33.4 MB, publicly available
004_NUSARC02FIGURES.pdf	32.6 MB publicly available
005_NUSARC03.pdf	22.0 MB, publicly available
006_NUSARC04.pdf	3,209 KB, publicly available
007_NUSARC05.pdf	1,572 KB, publicly available
008_NUSARC06.pdf	16.6 MB, publicly available
009_NUSARC07.pdf	1.91 MB, publicly available
010_NUSARC08.pdf	1.02 MB, publicly available
011_NUSARC09.pdf	14.3 MB, publicly available
012_NUSARC10.pdf	875 KB, publicly available
013_NUSARC11.pdf	1.64 MB, publicly available
014_NUSARC12.pdf	909 KB, publicly available
015_NUSARC13.pdf	335 KB, publicly available
016_NUSARC14.pdf	370 KB, publicly available
017_NUSARC15.pdf	9.70 MB, publicly available
018_NUSARC16.pdf	58 KB, publicly available
019_NUSARC17.pdf	92 KB, publicly available
020_NUSARC18.pdf	985 KB, publicly available
021_NUSARNRCQ.pdf	308 KB, publicly available
022_USAR Rev. 21-loep.pdf	446 KB, publicly available

Subject
Enclosed is the CD-ROM submittal of the station-controlled drawings that are considered incorporated by reference into the Wolf Creek Updated Safety Analysis Report (USAR). In accordance with 10 CFR 2.390, this enclosure is considered sensitive unclassified information and therefore warrants withholding.

Contact

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Document Components:
The CD-ROM labeled "Updated Safety Analysis Report Controlled Figure Drawings Only" contains the following files:

001_Chapter 1.pdf	18.6 MB, sensitive unclassified information
002_Chapter 2.pdf	2.33 MB, sensitive unclassified information
003_Chapter 5.pdf	3.07 MB, sensitive unclassified information
004_Chapter 6.pdf	4.14 MB, sensitive unclassified information
005_Chapter 7.pdf	1.68 MB, sensitive unclassified information
006_Chapter 8.pdf	2.01 MB, sensitive unclassified information
007_Chapter 9.pdf	39.4 MB, sensitive unclassified information
008_Chapter 10.pdf	16.6 MB, sensitive unclassified information
009_Chapter 11.pdf	3.12 MB, sensitive unclassified information
010_Chapter 12.pdf	2.66 MB, sensitive unclassified information
011_Chapter 18.pdf	207 KB, sensitive unclassified information
012_Index Removed Figure List.pdf	90.1 KB, sensitive unclassified information

Subject
Enclosed is the CD-ROM submittal of the station Fire Hazards Analysis and Quality Program Manual that are considered incorporated by reference into the Wolf Creek Updated Safety Analysis Report (USAR).

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Document Components:
The CD-ROM labeled "Updated Safety Analysis Report Fire Hazards Analysis and Quality Program Manual" contains the following files:

001_Quality Program Manual.pdf	263 KB, publicly available
002_E-19900.pdf	2.21 MB, publicly available
003_E-19905.pdf	0.98 MB, publicly available
004_E-19910.pdf	1.83 MB, publicly available
005_WIP-E-9910.pdf	57.5 KB, publicly available
006_XX-E-013-Rev 1.pdf	1.35 MB, publicly available
007_XX-E-013-Rev.1-CN001.pdf	167 KB, publicly available
008_XX-E-013-Rev.1-CN002.pdf	192 KB, publicly available
009_XX-E-013-Rev.1-CN003.pdf	155 KB, publicly available
010_XX-E-013-Rev.1-CN005.pdf	120 KB, publicly available
011_XX-E-013-Rev.1-CN006.pdf	214 KB, publicly available
012_XX-E-013-Rev.1-CN007.pdf	174 KB, publicly available
013_XX-E-013-Rev.1-CN008.pdf	135 KB, publicly available
014_XX-E-013-Rev.1-CN009.pdf	107 KB, publicly available
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016_M-663-00017A W03 CS to B1-98.pdf	43.1 MB, publicly available
017_M-663-00017A W03 B1-99 to B2-408.pdf	41.6 MB, publicly available
018_M-663-00017A W03 B2-41 to B6-2.pdf	49.9 MB, publicly available
019_M-663-00017A W03 B6-3 to B8-3.pdf	46.7 MB, publicly available
020_M-663-00017A W03 B8-4 to B8-147.pdf	48.4 MB, publicly available
021_M-663-00017A W03 B8-148 to B13-25.pdf	39.5 MB, publicly available
022-M-663-00017A W03 B13-26 to G1A-63.pdf	46.3 MB, publicly available
023_M-663-00017A W03 G1A-64 to G2B-61.pdf	47.7 MB, publicly available
024_M-663-00017A W03 G2B-62 to G3D-50.pdf	34.0 MB, publicly available
025_M-663-00017A W03 ATT G4 to ATT H.pdf	35.6 MB, publicly available

Enclosure IV to WM 08-0003

TRM Replacement Pages

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.16 RCS Chemistry

TR 3.4.16 RCS chemistry shall be maintained within the limits specified in Table TR 3.4.16-1.

APPLICABILITY: According to Table TR 3.4.16-1.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more chemistry parameters > steady state limit and ≤ transient limit in MODES 1, 2, 3, and 4.	A.1	Restore parameter to within steady state limit.	24 hours
B. Required Action and Completion Time of Condition A not met. OR One or more chemistry parameters > transient limit in MODES 1, 2, 3, and 4.	B.1 AND B.2	Be in MODE 3. Be in MODE 5.	6 hours 36 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. ------------NOTE-------------- Required Action C.2 must be completed whenever Condition C is entered. ----------------------------------	C.1 Initiate action to reduce pressurizer pressure to ≤ 500 psig.	Immediately
	AND	
Steady state limit exceeded for Chloride or Fluoride for > 24 hours while in other than MODES 1, 2, 3, and 4. OR Transient limit exceeded for Chloride or Fluoride while in other than MODES 1, 2, 3, and 4.	C.2 Determine RCS is acceptable for continued operation.	Prior to increasing pressurizer pressure > 500 psig OR Prior to entering MODE 4

TECHNICAL SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
TSR 3.4.16.1	--------------------------NOTE--------------------------------- Not required to be met for Chloride and Fluoride until 72 hours after entry into MODE 6 from a defueled state. ---	
	Verify RCS chemistry parameters within limits provided in Table TR 3.4.16-1.	72 hours

Table TR 3.4.16-1
RCS Chemistry Limits

PARAMETER	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	STEADY STATE LIMIT	TRANSIENT LIMIT
1. Dissolved Oxygen	At all times [(a)]	≤ 0.10 ppm	≤ 1.00 ppm
2. Chloride	At all times[(b)]	≤ 0.15 ppm	≤ 1.50 ppm
3. Fluoride	At all times[(b)]	≤ 0.15 ppm	≤ 1.50 ppm

(a) Not required with T_{avg} less than or equal to 250°F.

(b) Limit not applicable with no fuel in the reactor vessel.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.17 *Structural Integrity*

TR 3.4.17 The structural integrity of ASME Code Class 1, 2, and 3 components shall be maintained.

APPLICABILITY: MODES 1, 2, 3, 4, 5, and 6.

ACTIONS

---NOTE---
Separate Condition entry is allowed for each component.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ------------NOTE-------------- Not applicable to moderate-energy Class 2 or 3 piping. --------------------------------- One or more ASME Code Class 1, 2, or 3 component(s) contain(s) through-wall flaw. <u>OR</u> Structural integrity of one or more ASME Code Class 1, 2, or 3 component(s) not maintained.	A.1 Declare the affected component(s) inoperable. <u>AND</u> A.2 Enter applicable Conditions and Required Actions of Technical Specification or Technical Requirement for the affected component(s).	Immediately Immediately

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. ASME Code Class 2 or 3 piping in moderate-energy fluid system(s) contains through-wall flaw.	B.1	Verify structural integrity is still maintained.	72 hours
	OR		
	B.2	Isolate the affected component(s) from service.	72 hours
C. Required Action and associated Completion Time of Condition B not met. OR Structural integrity of ASME Code Class 2 or 3 piping in moderate-energy fluid system(s) containing through-wall flaw not maintained.	C.1	Declare the affected component(s) inoperable.	Immediately

TECHNICAL SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
TSR 3.4.17.1	Perform inservice inspection of ASME Section XI Code Class 1, 2, and 3 components.	In accordance with Inservice Inspection Program

3.7 PLANT SYSTEMS

3.7.20 Snubbers

TR 3.7.20 All required snubbers shall be OPERABLE, and no system, or portion thereof, has experienced an unexpected, potentially damaging transient.

APPLICABILITY: MODES 1, 2, 3, and 4,
MODES 5 and 6 for snubbers located on systems required OPERABLE.

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each affected system.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ------NOTE------ Required Action A.3 must be completed whenever this Condition is entered. ------------ One or more required snubbers inoperable.	A.1 Declare the system inoperable if it is known or indeterminate that the attached inoperable snubber is required for system OPERABILITY. <u>AND</u> A.2 Refer to Technical Specification LCO 3.0.8. <u>AND</u> A.3 Perform an engineering evaluation per section 5 of Table TR 3.7.20-4 on the attached component.	Immediately Immediately 72 hours

(continued)

ACTION (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. A system, or portion thereof, has experienced an unexpected, potentially damaging snubber transient.	B.1	Perform an inspection of required snubbers affected by transient in accordance with Table TR 3.7.20-1.	Within 6 months following the event
C. Required Action and associated Completion Time not met.	C.1	Declare attached system inoperable.	Immediately

TECHNICAL SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
TSR 3.7.20.1	Perform visual inspections of each required snubber in accordance with Table TR 3.7.20-2.	In accordance with Table TR 3.7.20-3
TSR. 3.7.20.2	----------------------------NOTE-------------------------------- This surveillance shall not be performed in MODES 1 and 2. -- Perform a functional test on a representative sample of each type of snubber in accordance with Table TR 3.7.20-4.	18 months
TSR 3.7.20.3	Verify that the service life of mechanical snubbers is not exceeded.	In accordance with Snubber Service Life Program

PAGE (1)	REVISION NO. (2)	CHANGE DOCUMENT (3)	DATE EFFECTIVE/ IMPLEMENTED (4)
TAB – Title Page TRM			
Title Page			
TAB – Table of Contents			
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ii	25	DRR 05-1996	9/28/05
iii	25	DRR 05-1996	9/28/05
TAB – 1.0 USE AND APPLICATION			
1.0-1	21	DRR 04-1535	12/1/04
TAB – 3.0 APPLICABILTY			
3.0-1	20	DRR 04-1410	10/7/04
3.0-2	20	DRR 04-1410	10/7/04
3.0-3	12	DRR 02-0515	5/2/02
3.0-4	20	DRR 04-1410	10/7/04
TAB – 3.1 REACTIVITY CONTROL SYSTEMS			
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3.1-2	3	DRR 99-1581	12/18/99
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3.1-4	11	DRR 02-0412	4/5/02
3.1-5	11	DRR 02-0412	4/5/02
3.1-6	11	DRR 02-0412	4/5/02
3.1-7	11	DRR 02-0412	4/5/02
3.1-8	13	DRR 02-1063	9/26/02
3.1-9	11	DRR 02-0412	4/5/02
3.1.10	11	DRR 02-0412	4/5/02
TAB – 3.3 INSTRUMENTATION			
3.3-1	23	DRR 05-0708	4/20/05
3.3-2	23	DRR 05-0708	4/20/05
3.3-3	23	DRR 05-0708	4/20/05
3.3-4	23	DRR 05-0708	4/20/05
3.3-5	23	DRR 05-0708	4/20/05
3.3-6	23	DRR 05-0708	4/20/05
3.3-7	23	DRR 05-0708	4/20/05
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3.3-10	23	DRR 05-0708	4/20/05
3.3-11	23	DRR 05-0708	4/20/05
3.3-12	23	DRR 05-0708	4/20/05
3.3-13	23	DRR 05-0708	4/20/05
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3.3-15	23	DRR 05-0708	4/20/05
3.3-16	23	DRR 05-0708	4/20/05
3.3-17	23	DRR 05-0708	4/20/05
3.3-18	23	DRR 05-0708	4/20/05
3.3-19	23	DRR 05-0708	4/20/05
3.3-20	23	DRR 05-0708	4/20/05
3.3-21	23	DRR 05-0708	4/20/05
3.3-22	23	DRR 05-0708	4/20/05

BASES

TECHNICAL SURVEILLANCE REQUIREMENTS (continued)

TSR 3.3.16.2

A CHANNEL OPERATIONAL TEST is to be performed every 31 days on each required channel to ensure the entire channel will perform the intended function. This test verifies the capability of the explosive gas monitoring instrumentation to detect levels of oxygen and hydrogen prior to them reaching flammable levels. The Frequency is based on operating experience.

TSR 3.3.16.3

A CHANNEL CALIBRATION is performed every 92 days. CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to measured parameters with the necessary range and accuracy. The Frequency is based on operating experience.

For the Inlet Hydrogen Monitor, the CHANNEL CALIBRATION shall include the use of standard gas samples containing a nominal 1% by volume hydrogen, at least 10% by volume helium, and balance nitrogen mixture; a nominal 4% by volume hydrogen and balance nitrogen mixture; and a 100% helium gas. For the Outlet Hydrogen Monitor, the CHANNEL CALIBRATION shall include the use of standard gas samples containing a nominal 1% by volume hydrogen, at least 10% by volume helium, and balance nitrogen mixture; a nominal 4% by volume hydrogen and balance nitrogen mixture; and a 100% helium gas.

For the Inlet Oxygen Monitor, the CHANNEL CALIBRATION shall include the use of standard gas samples containing a nominal 3% by volume oxygen, balance nitrogen mixture, and a nominal 5% by volume oxygen, balance nitrogen mixture. For the Outlet Oxygen Monitor, the CHANNEL CALIBRATION shall include the use of standard gas samples containing a nominal 25 ppm by volume oxygen, balance nitrogen mixture, and a nominal 75 ppm by volume oxygen, balance nitrogen mixture.

REFERENCES

1. USAR, Section 11.3.1.

2. WCAP-11618, “MERITS Program-Phase II, Task 5, Criteria Application,” including Addendum 1 dated April, 1989.

3. USAR Change Request 97-204.

4. Technical Specification 5.5.12, “Explosive Gas and Storage Tank Radioactivity Monitoring Program.”

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TAB – 3.3 INSTRUMENTATION (continued)			
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3.3-24	23	DRR 05-0708	4/20/05
3.3-25	23	DRR 05-0708	4/20/05
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3.3-29	26	DRR –6-0050	2/28/06
3.3-30	23	DRR 05-0708	4/20/05
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3.3-32	23	DRR 05-0708	4/20/05
TAB – 3.4 REACTOR COOLANT SYSTEM (RCS)			
3.4-1	3	DRR 99-1581	12/18/99
3.4-2	3	DRR 99-1581	12/18/99
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3.4-10	3	DRR 99-1581	12/18/99
3.4-11	3	DRR 99-1581	12/18/99
TAB – 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)			
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TAB – 3.6 CONTAINMENT SYSTEMS			
3.6-1	17	DRR 04-0452	5/26/04
3.6-2	3	DRR 99-1581	12/18/99
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TAB – 3.7 PLANT SYSTEMS			
3.7-1	3	DRR 99-1581	12/18/99
3.7-2	28	DRR 06-1349	7/24/06
3.7-3	25	DRR 05-1996	9/28/05
3.7-4	25	DRR 05-1996	9/28/05
3.7-5	25	DRR 05-1996	9/28/05
3.7-6	25	DRR 05-1996	9/28/05
3.7-7	26	DRR 06-0050	2/28/06
3.7-8	25	DRR 05-1996	9/28/05
3.7-9	32	DRR 07-1058	07/10/07
3.7-10	32	DRR 07-1058	7/10/07
3.7-11	25	DRR 05-1996	9/28/05
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3.7-18	25	DRR 05-1996	9/28/05
3.7-20	25	DRR 05-1996	9/28/05
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5.0-4	24	DRR 05-1376	6/28/05
5.0-5	17	DRR 04-0452	9/28/05
5.0-6	25	DRR 05-1996	2/16/05
5.0-7	16	DRR 03-1498	11/4/03

Note 1 The page number is listed on the center of the bottom of each page.

Note 2 The revision number is listed in the lower right hand corner of each page. The Revision number will be page specific.

Note 3 The change document will be the document requesting the change. Therefore, the change document should be a DRR number in accordance with AP 26A-002.

Note 4 The date effective or implemented is the date the Technical Requirement pages are to be issued by Document Control.

BASES

ACTIONS

A.1 and A.2 (continued)

In addition to restoring operation to within limits, an engineering evaluation of the structural integrity of the pressurizer is required within 72 hours to determine if operation may continue. This may require event-specific stress analyses or inspections. WCAP-14717 (Ref. 6) provides guidance for engineering evaluation of transients. A favorable evaluation must be completed before continuing operation. The Completion Time of 72 hours is consistent with that allowed in Technical Specification 3.4.3, "RCS Pressure and Temperature Limits."

A Note is provided to clarify that Required Action A.2 must be completed whenever this Condition is entered. The Note emphasizes the need to perform the engineering evaluation of the effects of the excursion outside the allowable limits. Restoration to within limits is insufficient without the evaluation of the structural integrity of the pressure boundary of the pressurizer.

B.1

In the event that the Required Action and associated Completion Time are not met, Required Action B.1 requires initiation of a Performance Improvement Request (PIR) immediately to address why the pressurizer pressure/temperature limit was not restored to OPERABLE status within the Completion Time. As part of the initiation of the PIR, action shall be implemented in a timely manner to place the unit in a safe condition as determined by plant management. The PIR should provide an accurate description of the problem, the Required Action and associated Completion Time not complied with. The intent of this Required Action is to utilize the corrective action program to assure prompt attention and adequate management oversight to minimize the additional time pressurizer temperature is not within limits.

TECHNICAL SURVEILLANCE REQUIREMENTS

TSR 3.4.3.1

This TSR verifies the rate of heatup is within limit. The 30 minute Frequency is considered reasonable based on the instrumentation available in the control room to monitor the status of the RCS. The TSR has been modified by a Note which requires the TSR to be performed only during pressurizer heatup.

BASES

TECHNICAL SURVEILLANCE REQUIREMENTS (continued)

TSR 3.4.3.2

TSR 3.4.3.2. verifies the rate of cooldown is within limits. "Step-wise" cooling must be avoided as discussed in Reference 4. The Frequency of 30 minutes is considered reasonable based on the instrumentation available in the control room to monitor the status of the RCS. The TSR has been modified by a Note, which requires the TSR to be performed only during pressurizer cooldown.

TSR 3.4.3.3

This TSR verifies that the maximum spray water temperature differential does not exceed 583°F. This is to guard against subjecting the spray nozzle to undue thermal stresses. The 12 hour Frequency is considered adequate based upon operating experience. The TSR has been modified by a Note, which requires the TSR to be performed only during auxiliary spray operation.

REFERENCES

1. 10 CFR 50.2, "Definitions."
2. 10 CFR 50.55a, "Codes and Standards."
3. ASME Boiler and Pressure Vessel Code, Section III, Appendix G.
4. Westinghouse letter SAP-90-263, "Reactor Coolant System Accelerated Cooldown," dated November 5, 1990.
5. USAR, Section 5.3.3.1.
6. WCAP-14717, Rev. 1, Supplement 1, "Basis and Application of Pressurizer Heatup and Cooldown Limits," November 1999.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

TR B 3.4.16 RCS Chemistry

BASES

BACKGROUND

The Reactor Coolant System (RCS) water chemistry is selected to minimize corrosion. A periodic analysis of the coolant chemical composition is performed to verify that the reactor coolant quality meets the specification (Ref. 1). This Technical Requirement places limits on the dissolved oxygen, chloride and fluoride content of the RCS to minimize corrosion.

Limiting dissolved oxygen content of the RCS limits the amount of general corrosion and reduces the possibility of stress corrosion. General corrosion is a contributing factor in reactor coolant activity and must be controlled for ALARA (as low as reasonably achievable) considerations as well as structural integrity considerations.

Both chlorides and fluorides have been shown to cause stress corrosion if present in the RCS in sufficiently high concentrations at high pressure and temperature conditions. Stress corrosion can lead to either localized leakage or catastrophic failure of the RCS. The associated effects of exceeding the dissolved oxygen, chloride, and fluoride limits are time and temperature dependent. Corrosion studies show that operation may be continued with contaminant concentration levels in excess of the steady state limits, up to the transient limits, for the specified limited time intervals without having a significant effect on the structural integrity of the RCS.

APPLICABLE SAFETY ANALYSES

The limitations on RCS chemistry ensure that corrosion of the RCS is minimized and reduces the potential for RCS LEAKAGE or failure due to stress corrosion. Maintaining the chemistry within the steady state limits provides adequate corrosion protection to ensure the structural integrity of the RCS over the life of the plant. It is not, however, a consideration in the analyses of Design Basis Accidents.

The associated effects of exceeding the oxygen, chloride, and fluoride limits are time and temperature dependent. Corrosion studies show that operation may be continued with contaminant concentration levels in excess of the steady state limits, up to the transient limits, for the specified limited time intervals without having a significant effect of the structural integrity of the RCS.

BASES

TR

TR 3.4.16 establishes the steady state and transient limits on concentration of dissolved oxygen, chloride and fluoride in the RCS. These limits ensure that dissolved oxygen, chloride and fluoride concentrations are maintained at levels low enough to prevent unacceptable degradation of the RCS pressure boundary.

APPLICABILITY

Concentrations of dissolved oxygen, chloride and fluoride in the RCS must be maintained within limits at all times as specified in Table TR 3.4.16-1. Applicability is modified by a Note on Table TR 3.4.16-1 indicating that dissolved oxygen is not required with $T_{avg} \leq 250°F$ and the chloride and fluoride limits are not applicable when there is no fuel in the reactor vessel.

ACTIONS

A.1

If one or more chemistry parameters are not within steady state limits in MODES 1, 2, 3, or 4, the parameter(s) must be restored to their steady state limit within 24 hours. This allows time to take corrective actions to restore the contaminant concentrations to within the steady state limits.

B.1 and B.2

With one or more chemistry parameters not within transient limits in MODES 1, 2, 3, or 4, or if the Required Action of Condition A is not met within the associated Completion Time, the plant must be placed in a condition where the limit is not applicable or where corrosion rates are reduced. This is accomplished by placing the plant in MODE 3 within 6 hours and MODE 5 within 36 hours. In MODE 5, the dissolved oxygen limit is not applicable and stress corrosion rates are reduced. The 6 hours allotted to reach MODE 3 is a reasonable time, based on operating experience, to shutdown the plant from full power in an orderly manner and without challenging plant systems. The Completion Time of 36 allows for a reasonable time to reach MODE 5 from full power conditions in and orderly manner and without challenging plant systems.

If high chloride or fluoride concentrations are the reason for entering Condition B, and the condition is not corrected before entering MODE 5, Required Actions C.1 and C.2 must be performed.

BASES

ACTIONS (continued)

C.1

If RCS chloride or fluoride concentration are not within steady state limits for more than 24 hours in any condition other than MODES 1, 2, 3, or 4, or if RCS chloride or fluoride concentration are not within transient limits for any amount of time in any condition other than MODES 1, 2, 3, or 4, action must be immediately initiated to reduce pressurizer pressure to ≤ 500 psig unless it is already below 500 psig. The immediate Completion Time is consistent with the required times for actions to be performed without delay and in a controlled manner.

A Note is added to Condition C stating that Required Action C.2 must be completed whenever this Condition is entered.

C.2

In addition to Required Action C.1, an engineering evaluation must be performed to determine the effects of the out-of-limit condition on the structural integrity of the RCS and that the RCS remains acceptable for continued operation. These actions must be taken prior to increasing pressurizer pressure above 500 psig or prior to entry in MODE 4. These evaluations are necessary because of the time/temperature/concentration dependency of the effects of exceeding the limits. Corrosion evaluations for conditions outside the limits are made on a case by case basis.

TECHNICAL SURVEILLANCE REQUIREMENTS

TSR 3.4.16.1

TSR 3.4.16.1 requires through a representative sample that the chemistry parameters are within their limits of Table TR 3.4.16-1. The TSR is modified by a Note stating that the chloride and fluoride limit is not required to be met until 72 hours after entry into MODE 6 from a defueled state. Representative samples for chloride and fluoride concentration can not be obtained during defueled conditions when the RCS and Residual Heat Removal System is not recirculating. Therefore, the Note allows entry into MODE 6 when a representative sample can be obtained. A Frequency of 72 hours provides adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

REFERENCES

1. USAR, Section 5.2.3.2.

B 3.4 REACTOR COOLANT SYSTEM

TR B 3.4.17 Structural Integrity

BASES

BACKGROUND	The quality group classification for each water- and steam- containing pressure component is shown in USAR Table 3.2-1. The components are classified according to their safety significance as dictated by service and functional requirements and by the consequences of their failure. The quality group classifications and code requirements for the quality of plant process systems meet the intent of Regulatory Guides 1.26 and 1.143 (Ref. 1). The design, fabrication, inspection, and testing requirements of each classification provide the required degree of conservatism in assuring component pressure integrity and OPERABILITY (Ref. 1). The Code requirements applicable to each quality group classification are identified in USAR Table 3.2-2. The quality group classifications and the interfaces between classifications in a system having components of different classifications are indicated on the piping and instrumentation diagram or flow diagram of that system (Ref. 1).
APPLICABLE SAFETY ANALYSES	Certain components which are designed and manufactured to the requirements of specific sections of the ASME Boiler and Pressure Vessel Code are part of the primary success path and function to mitigate DBAs and transients. However, the OPERABILITY of these components is addressed in the relevant specifications that cover individual components.
TR	TR 3.4.17 requires the structural integrity of ASME Code Class 1, 2, and 3 components be maintained. Structural integrity is the functional capability (i.e., pressure retaining capability) of ASME Code Class 1, 2, and 3 systems during all design and operational conditions. In those areas where conflict may exist between the Technical Requirements and the ASME Boiler and Pressure Vessel Code, the Technical Requirements Manual takes precedence.
APPLICABILITY	The structural integrity of the ASME Code Class 1, 2, and 3 components is required during MODES 1, 2, 3, 4, 5, and 6. This TR applies whenever Code Class 1, 2, or 3 components are OPERABLE, not just during the performance of inservice inspection examinations.

BASES

ACTIONS

A Note has been added to the ACTIONS to clarify the application of Completion Time rules. The Conditions of this Requirement may be entered independently for each component. The Completion Time(s) for the inoperable component will be tracked separately for each component starting from the time the Condition was entered for that component.

A.1 and A.2

Condition A applies to one or more degraded or non-conforming ASME Code Class 1, 2, or 3 component(s) (excluding moderate-energy Class 2 or 3 piping) containing the specific degradation of a through-wall flaw or degraded or non-conforming ASME Code Class 1, 2, or 3 component(s), not including through-wall flaws, for which structural integrity is not maintained. Condition A is modified by a Note which limits the applicability of the Condition of one or more ASME Code Class 1, 2, or 3 component(s) containing a through-wall flaw by excluding moderate-energy Class 2 or 3 piping. With one or more ASME Code Class 1, 2, or 3 component(s) containing a through-wall flaw (excluding moderate energy Class 2 or 3 piping) or one or more ASME Code Class 1, 2, or 3 components, not including through-wall flaws (no limitation on component applicability) for which structural integrity is not maintained, Required Actions A.1 and A.2 require the affected component(s) to be declared inoperable and the applicable Conditions and Required Actions of the Technical Specifications or Technical Requirement be entered for the affected component. The immediate Completion Time is consistent with the required times for actions to be performed without delay and in a controlled manner.

Moderate-energy fluid systems are systems that during normal plant conditions meet both of the following conditions:

a. maximum operating temperature is 200°F or less, and

b. maximum operating pressure is 275 psig or less.

In accordance with 10 CFR 50.55a(g), structural integrity must be maintained in conformance with ASME Code Section XI for those components that are subject to Code requirements. Depending on the type of degraded or nonconforming condition, structural integrity can be determined by meeting applicable ASME Section III requirements, Section XI acceptance standards, Section XI IWB-3600 analytical

BASES

ACTIONS

A.1 and A.2 (continued)

evaluation procedures, or applicable NRC approved ASME Code Cases. In addition, NRC Inspection Manual, Part 9900, Technical Guidance (Regulatory Issue Summary 2005-020 (Ref. 2), Appendix C, Section C.10 (Piping and Pipe Support Requirements), C.11 (Flaw Evaluation), and C.12 (Operational Leakage From Code Class 1, 2, and 3 Components) contain specific NRC guidance and acceptable evaluation procedures than can be used to determine structural integrity and OPERABILITY for ASME Code Class 1, 2, and 3 components. Modified NRC OPERABILITY guidance was provided in Reference 7.

If a flaw is discovered by any means (including surveillance, maintenance activity, or inservice inspection) in a system subject to Code requirements (whether during normal plant operation, plant transition, or shutdown operation), the flaw must be promptly evaluated to assure structural integrity is maintained. If the flaw results in through-wall leakage of a Class 1, 2, or 3 component (excluding moderate-energy Class 2 or 3 piping), the component containing the flaw is inoperable.

If a flaw evaluation determines that structural integrity is not maintained, the component and associated system containing the flaw are inoperable. The system may be considered OPERABLE with the component isolated as long as the system is capable of performing its specified safety function. If the flaw evaluation determines that structural integrity is maintained, the component and associated system is OPERABLE. However, a determination should be made as to how long the flawed component will remain OPERABLE before the flaw grows to exceed acceptable limits and criteria.

Non Through-Wall Flaws

Use of Generic Letter (GL) 90-05 was included in NRC Inspection Manual, Part 9900, Technical Guidance (Ref. 2) (Appendix C, Section C.11) addressing flaws in Class 3 piping not resulting in through-wall leaks. GL 90-05 was originally issued to address temporary non-Code repairs. It also provided NRC approved guidance in assessing structural integrity in moderate-energy Class 3 piping as part of the licensee relief request when a Code repair was deferred. 10 CFR 50.55a(b)(2)(xiii) (Ref. 3) modified the GL 90-05 guidance in 1999 by approving ASME Section XI Code Case N-513 and Case N-523-1 (Ref. 6). The Federal Register issuing this amendment noted "These Code Cases were developed to address criteria for temporary acceptance of flaws (including through-wall leaking) of moderate energy Class 3 piping where a Section

BASES

ACTIONS

A.1 and A.2 (continued)

XI Code repair may be impractical for a flaw detected during plant operation (i.e., a plant shutdown would be required to perform the Code repair). In the past, licensees had to request NRC staff approval to defer Section XI Code repair for these Class 3 moderate energy (200 deg. F, 275 psig) piping systems.......The use of Code Case N-513, with the limitations, and Code Case N-523-1 will obviate the need for licensees to request approval for deferring repairs; thus saving NRC and licensee resources."

The NRC has approved ASME Code Case N-513-1 in Regulatory Guide 1.147, Rev. 14, with limitations that must be implemented with the Case. Code Case N-513-1 has been adopted for use at WCGS. Code Case N-523-1 has been incorporated into Appendix IX of the ASME Section XI edition and addenda used at WCGS. As a result, Appendix IX is used in lieu of Code Case N-523-1 for resolution of degraded and non-conforming conditions (including through-wall leaks) by installation of a structural mechanical clamp to restore structural integrity.

Therefore, when Code Case N-513-1 is applicable (i.e., the scope of Code Case N-513-1 and the NRC limitations applied to the use of the Case allow the Case to be used), Case N-513-1 should be considered before use of GL 90-05 to temporarily accept degraded and nonconforming conditions in Class 2 and 3 moderate-energy piping. GL 90-05 guidance on temporary non-Code repairs remains applicable. In addition, if Code Case N-513-1 is not applicable to a degraded or nonconforming condition in moderate-energy Class 3 piping, GL 90-05 should be considered recognizing the associated need to request relief from the NRC.

Alternative evaluation procedures and/or acceptance criteria may also be considered for confirming structural integrity of degraded or nonconforming conditions. However, when alternative evaluation procedures and/or acceptance criteria are used as a basis for structural integrity and acceptable continued service, the component and associated system containing the degraded or nonconforming condition is considered inoperable until NRC approval of procedures and criteria is obtained (Ref. 2).

BASES

ACTIONS

A.1 and A.2 (continued)

Through-Wall Flaws

The NRC also provided guidance in NRC Inspection Manual, Part 9900, Technical Guidance, (Regulatory Issue Summary 2005-020 (Ref. 2)) for addressing specific degraded and nonconforming conditions consisting of through-wall pressure boundary leaks. In NRC Inspection Manual, Part 9900, Technical Guidance, Appendix C, Section C.12, the NRC stated that "Upon discovery of leakage from a Class 1, 2, or 3 pressure boundary component (pipe wall, valve body, pump casing, etc.) the licensee must declare the component inoperable." Evidence of leakage from the pressure boundary indicates the presence of a through-wall flaw. The NRC further stated that because the size of the through-wall flaw on the inside surface is unknown, the component is declared inoperable while methods such as ultrasonic examination are performed to characterize the actual geometry of the through-wall flaw. However, Reference 7 provided modified NRC guidance for moderate-energy Class 2 or 3 piping with through-wall leakage.

In Reference 2, Appendix C, Section C.12, the NRC noted that structural mechanical clamps may be used on high-energy Class 2 or 3 piping 2 inches and smaller to restore structural integrity of through-wall leaks. However, as discussed above, this piping must be declared inoperable when the through-wall leakage is discovered and must remain inoperable until the Section XI Appendix IX requirements are met for the installation of the mechanical clamp.

If NRC approved guidance or generically approved alternatives are not available for resolution of the through-wall leaks in ASME Code Class 1, 2, or 3 components (excluding moderate-energy Code Class 2 or 3 piping) the component(s) must have a repair/replacement activity performed in accordance with ASME Section XI, or relief from Code requirements must be requested and approval obtained from the NRC prior to declaring the component(s) OPERABLE.

B.1 and B.2

Condition B applies to ASME Code Class 2 or 3 piping in moderate-energy fluid system(s) containing a through-wall flaw. Moderate-energy fluid systems are systems that during normal plant conditions meet both of the following conditions:

a. maximum operating temperature is 200°F or less, and

b. maximum operating pressure is 275 psig or less.

BASES

ACTIONS

B.1 and B.2 (continued)

The NRC provided guidance in Reference 2 for addressing the specific degraded and non-conforming condition consisting of through-wall pressure boundary leaks. However, Reference 7 provided modified NRC guidance for moderate-energy Class 2 or 3 piping with through-wall leakage. The modified guidance did not require moderate-energy Class 2 or 3 piping with through-wall leakage to be declared inoperable upon discovery of the leakage. The modified guidance stated that through-wall leakage immediate OPERABILITY determination should be based on reasonable expectation of OPERABILITY in accordance with the guidance of Reference 2. It further stated that through-wall leakage prompt OPERABILITY determination should be based on actual NDE results to characterize the flaw(s) dimensions causing the through-wall leakage and engineering analysis methods acceptable to the NRC, i.e., NRC approved Code Cases and GL 90-05.

Required Action B.1 stipulates verifying within 72 hours that structural integrity is still maintained with a through-wall flaw. Structural integrity for through-wall leakage is determined in accordance with applicable ASME Code Cases and NRC Inspection Manual, Part 9900, Technical Guidance (Ref. 2), Appendix C, Section C.11 (Flaw Evaluation), and C.12 (Operational Leakage From Code Class 1, 2, and 3 Components), as modified by Reference 7. These contain specific NRC guidance and acceptable evaluation procedures to be used in determining structural integrity and OPERABILITY for moderate-energy ASME Code Class 2 or 3 piping.

For Class 3 moderate-energy piping, the structural integrity of the piping may be evaluated by fully characterizing the extent of the through-wall flaw using volumetric methods and evaluating the flaw using the criteria of paragraph C.3.a of Enclosure 1 to GL 90-05 (Ref. 4). If the flaw meets the criteria, the piping is still considered OPERABLE but degraded until relief from the applicable Code requirement or requirements is obtained from the NRC. The structural integrity of leaking Class 2 or 3 moderate-energy piping may also be evaluated using criteria of Code Case N-513-1. The NRC has approved Code Case N-513-1 (Ref. 5) in Regulatory Guide 1.147, Rev. 14, with limitations that must be implemented along with Code Case N-513-1. If the piping meets the criteria of ASME Code Case N-513-1, with NRC limitations, the piping may be deemed OPERABLE and continued temporary service of the degraded piping is permitted.

BASES

ACTIONS

B.1 and B.2 (continued)

Additional Bases description for use of GL 90-05 and Code Case N-513-1 is included in Required Actions A.1 and A.2 above.

In Reference 2, Appendix C, Section C.12, the NRC noted that structural mechanical clamps may be used on 6 inch and smaller moderate-energy Class 2 or 3 piping to restore structural integrity of through-wall leaks. However, if the structural integrity has not been verified by use of GL 90-05 or Code Case N-513-1, the piping containing the through-wall leak is inoperable until the ASME Section XI, Appendix IX requirements are met for the installation of the mechanical clamp.

If NRC approved guidance or generically approved alternatives are not available for resolution of through-wall leaks in moderate-energy Class 2 or 3 piping, the piping containing the leak must have a repair/replacement activity performed in accordance with ASME Section XI, or relief from Code requirements must be requested and approval obtained from the NRC prior to declaring the piping OPERABLE.

The Completion Time of Required Action B.1 is expected to be a maximum time limit. The guidance of Reference 2 and Reference 7 would expect an earlier Completion Time if possible, consistent with the safety significance of the component(s) containing the flaw. With this clarification, the 72 hours is reasonable and consistent with the guidance in References 2 and 7.

Alternatively, Required Action B.2 allows the affected component(s) to be isolated within 72 hours.

C.1

If the Required Actions and associated Completion Times of Condition B are not met or if structural integrity of the Class 2 or 3 piping (in moderate-energy fluid system(s)) containing a through-wall flaw is not maintained, the affected component(s) should be immediately declared inoperable.

BASES

TECHNICAL SURVEILLANCE REQUIREMENTS

TSR 3.4.17.1 requires performing inservice inspections of ASME Section XI Code Class 1, 2, and 3 components in accordance with the Inservice Inspection Program described in TR 5.5.6, "Inservice Inspection Program."

Inservice inspection of ASME Code Class 1, 2, and 3 components are performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code (Ref. 1) and applicable Addenda, as required by 10 CFR 50.55a(g), to ensure that the structural integrity of these components will be maintained at an acceptable level throughout the life of the plant. Exception to these requirements apply where relief has been granted by the Commission pursuant to 10 CFR 50.55a(a)(3) and (g)(6)(i). The surveillance intervals specified in Section XI of the ASME Code apply.

During design and construction, components of the Reactor Coolant System were designed to provide access to permit inservice inspection in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, 1974 Edition and Addenda through Summer 1975.

REFERENCES

1. USAR, Section 3.2.2

2. Nuclear Regulatory Issue Summary 2005-020 (including NRC Inspection Manual, Part 9900, Technical Guidance, "Operability Determinations & Functionality Assessments for Resolution of Degraded or Nonconforming Conditions Adverse to Quality or Safety").

3. 10 CFR 50.55a(b)(2)(xiii), "Flaws in Class 3 Piping," Federal Register (Vol. 64, No. 183), September 22, 1999.

4. Generic Letter 90-05, "Guidance for Performing Temporary Non-Code Repair of ASME Code Class 1, 2, and 3 Piping," June 15, 1990; and August 16, 1990, NRR letter by J. E. Richardson, "Follow-up on Generic Letter 90-05 Regarding Guidance for Performing Temporary Non-Code Repair of ASME Code Class 1, 2, and 3 Piping," (letter 90-02380).

5. Code Case N-513-1, "Evaluation Criteria for Temporary Acceptance of Flaws in Class 2 and 3 Piping."

6. ASME Section XI, Appendix IX, "Mechanical Clamping Devices for Class 2 and 3 Piping Pressure Boundary" (previously contained in Code Case N-523-1).

BASES

REFERENCES (continued)	7.	NRC Memorandum, "Inspection Manual Part 9900 Operability Guidance Involving Structural Integrity of ASME Code Class 2 and 3 Piping," June 22, 2007.

BASES

ACTIONS

A Note has been added to the ACTIONS to clarify the application of Completion Time rules. The Conditions of this Technical Requirement may be entered independently for each affected system. The Completion Time(s) of the inoperable snubber will be tracked separately for each affected system starting from the time the Condition was entered for that system as a result of discovery of an inoperable snubber.

A.1, A.2, and A.3

If one or more required snubbers have been declared inoperable, declare the system inoperable if it is known or indeterminate that the attached inoperable (nonfunctional) snubber is required for system OPERABILITY.

The system may be declared OPERABLE while utilizing Technical Specification (TS) LCO 3.0.8. Technical Specification LCO 3.0.8 provides an allowance under which systems are not considered to be inoperable solely due to one or more snubbers not capable of performing their associated support function(s). The following are applicable for utilizing LCO 3.0.8:

a. Determine whether a TS system is rendered inoperable by a inoperable snubber

   (1) If it is determined that the supported TS system(s) do not require the snubber(s) to be OPERABLE in order to support OPERABILITY of the system(s) LCO 3.0.8 is not needed.

   (2) If the LCO(s) associated with any supported TS system(s) are not currently applicable (i.e., the plant is not in a MODE or other specified condition in the Applicability of the LCO), LCO 3.0.8 is not needed.

   (3) If the support TS system(s) are inoperable for reasons other than snubbers, LCO 3.0.8 cannot be used.

b. Determine the design basis of the inoperable snubber. When a snubber is to be rendered inoperable for testing or maintenance or is discovered to be inoperable, the design function of the snubber must be determined in order to determine if LCO 3.0.8 may be used.

BASES

ACTIONS

A.1, A.2, and A.3 (continued)

(1) If the design function of the snubber is to react to only seismic loads, LCO 3.0.8 may be applied.

(2) If the design function of the snubber includes both seismic loads and non-seismic loads, any TS systems supported by the inoperable snubber must be able to remain OPERABLE if subjected to the non-seismic loads with the snubber removed. If the supported TS system will remain OPERABLE when subjected to non-seismic loads, LCO 3.0.8 may be applied.

(3) If the design function of the snubber includes only non-seismic loads, LCO 3.0.8 cannot be used.

c. When LCO 3.0.8.a is used, at least one train of the Auxiliary Feedwater (AFW) System not associated with the inoperable snubber must be OPERABLE.

d. When LCO 3.0.8.b is used, at least one train of the AFW System not associated with the inoperable snubber must be OPERABLE. If the inoperable snubber(s) supports all trains of the AFW System or if the AFW System becomes inoperable due to an emergent condition, some alternative means of core cooling must be available.

e. The LCO 3.0.8 requirement to assess and manage risk is met by programs to comply with the requirements of paragraph (a)(4) of the Maintenance Rule, 10 CFR 50.65, to assess and manage risk resulting from maintenance activities. LCO 3.0.8 should be considered with respect to other plant maintenance activities, and integrated into the existing Maintenance Rule process to the extent possible so that maintenance on any unaffected train or subsystem is properly controlled, and emergent issue properly addressed. The risk assessment need not be quantified, but may be a qualitative awareness of the vulnerability of systems and components when one or more snubbers are not able to perform their associated support function.

f. A record of the implementation of any applicable Tier 2 restrictions, and the associated plant configuration, shall be available on a recoverable basis for NRC inspection.

BASES

ACTIONS

A.1, A.2, and A.3 (continued)

Condition A is modified by a Note that requires that Required Action A.3 be completed whenever Condition A is entered. Thus, if the snubber is restored to OPERABLE status, the Condition will require the completion of an engineering evaluation per section 5 of Table TR 3.7.20-4.

The engineering evaluation is performed to:

a. Determine the cause of the failure

As a result of this evaluation, the need for testing other snubbers will be considered. The results from the testing will be used to consider expanded functional testing and cause examination with consideration of manufacturing and design deficiency.

b. Determine the impact on the supported component

This evaluation shall determine if the inoperable snubber has adversely affected the attached component.

The 72 hours is based on engineering experiences and is reasonable, considering the time it will take to identify the problem and take the proper corrective actions.

B.1

If the plant has experienced an unexpected, potentially damaging snubber transient, an inspection per Table TR 3.7.20-1 is performed on all snubbers attached to sections of systems that have experienced the transient. The potential impact of the transient is assessed by reviewing operating data and by visually inspecting the associated system. In addition to the visual inspection, the freedom-of-motion of the mechanical snubber(s) is verified per Table TR 3.7.20-1.

BASES

ACTIONS

B.1 (continued)

The Completion Time of 6 months has been assigned based upon industry practice.

C.1

If Required Actions and associated Completion Times of Condition A or B are not met, the supported system or component is immediately declared inoperable.

TECHNICAL SURVEILLANCE REQUIREMENTS

Surveillance Testing is performed in accordance with the applicable requirements of ASME Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components" (Ref. 1).

NRC letter dated June 2, 2006 (Ref. 10), approved the proposed alternative to use TRM, Section 3.7.20, for snubber visual inspection and functional testing in lieu of the applicable ASME Code requirements specified in Section XI, Article IWF-5000 for the third 10-year inservice inspection interval. The NRC safety evaluation specifies that changes to the TRM snubber visual inspection and functional testing requirements shall be submitted to the NRC for authorization pursuant to 10 CFR 50.55a(a)(3) or as an exemption pursuant to 10 CFR 50.12.

Permanent or other exemptions from the surveillance program for individual snubbers may be granted by the Commission if a justifiable basis for exemption is presented and, if applicable, snubber life destructive testing was performed to qualify the snubber for the applicable design conditions at either the completion of their fabrication or at a subsequent date. Snubbers so exempted shall be listed in the list of individual snubbers indicating the extent of the exemptions.

In order to establish the inspection frequency for each type of snubber on a safety related system, it was assumed that the frequency of snubber failures and initiating events is constant with time and that the failure of any snubber could cause the system to be unprotected and to result in failure during an assumed initiating event. Inspections performed before that interval has elapsed may be used as a new reference point to determine the next inspection. However, the results of such early inspections performed before the original required time interval has elapsed (nominal time less 25%) may not be used to lengthen the required inspection interval. Any inspection whose results require a shorter inspection interval will override the previous schedule.

BASES

TECHNICAL SURVEILLANCE REQUIREMENTS

TSR 3.7.20.1

TSR 3.7.20.1 comprises a visual inspection of the snubbers. A pre-fuel load visual inspection and functional test has been performed on each snubber using the acceptance criteria listed in Table TR 3.7.20-2. The baseline takes into account that the snubbers have experienced thermal cycling and normal operating service as a result of previous hot functional testing. The initial inservice inspection has been performed on the snubbers prior to completion of the first refueling outage. The frequency of subsequent surveillances depends on the number of snubbers found inoperable from each previous inspection as provided in Table TR 3.7.20-3 and the Inservice Inspection Program as described in TR 5.5.6. The acceptance criteria and corrective actions are listed in Table TR 3.7.20-2.

The visual inspections are designed to detect obvious indications of inoperability of the snubbers. Removal of insulation or direct contact with the snubbers is not required initially. However, suspected causes of inoperability are to be investigated and all snubbers of the same type and all snubbers subjected to the same failure mode are to be inspected more frequently.

The visual inspection frequency is based upon maintaining a constant level of snubber protection during an earthquake or severe transient and the number of unacceptable snubbers found during the previous inspection. As a result, the required inspection intervals vary inversely with the number of inoperable snubbers found during an inspection. If a snubber fails the visual acceptance criteria, the snubber is declared unacceptable and cannot be declared OPERABLE via functional testing. However, if the cause of rejection is understood and remedied for that type of snubber and for any other type of snubbers that may be generically susceptible and OPERABILITY verified by testing, that snubber may be reclassified acceptable for the purpose of establishing the next surveillance interval.

Snubbers may be categorized according to accessibility as noted in the Notes to Table TR 3.7.20-3. The accessibility of each snubber is determined based on radiation level as well as other factors such as temperature, atmosphere, location, etc. The recommendations of Regulatory Guide 8.8, "Information Relevant to Maintaining Occupational Radiation Exposure as Low a Practicable," (Ref. 7) and Regulatory Guide 8.10, "Operation Philosophy for Maintaining Occupational Radiation Exposure as Low as Practicable," (Ref. 8) are considered in planning and implementing the visual inspection program.

TECHNICAL SURVEILLANCE REQUIREMENTS

TSR 3.7.20.1 (continued)

Since the visual inspections are augmented by a functional testing program, the visual inspection need not be a hands on inspection, but shall require visual scrutiny sufficient to assure that fasteners or mountings for connecting the snubbers to supports or foundations have no visible bolts, pins or fasteners missing, or other visible signs of physical damage such as cracking or loosening.

TSR 3.7.20.2

This TSR is modified by a Note which restricts the performance of thisTSR to during periods of plant shutdown.

TSR 3.7.20.2 comprises the functional testing of snubbers. The testing for these snubbers have been separated into two sample plans as described in Table TR 3.7.20-4. Sample Plan 1.a (10%) is typically used for the snubbers with small population. Sample plan 1.b. (Figure TR 3.7.20-1) is typically used for snubbers with large population. Figure TR 3.7.20-1 was developed using "Wald's Sequential Probability Ratio Plan" as described in "Quality Control and Industrial Statistics" by Acheson J. Duncan.

The sample plan shall be selected prior to the test period and cannot be changed during the test period.

Snubber functional testing is performed to the requirements of Table TR 3.7.20-4 and performed prior to the completion of each refueling outage. The 18 month Frequency, in conjunction with the Note, is based on the need to perform this surveillance under the conditions that apply during a unit outage.

TSR 3.7.20.3

This TSR addresses the monitoring of the service life of the snubbers in accordance with the Snubber Service Life Program described in TR 5.5.5. The requirement to monitor the snubber service life is included to ensure that the snubbers periodically undergo a performance evaluation in view of their age and operating conditions.

REFERENCES

1. ASME Boiler and Pressure Vessel Code, Section III and XI.

2. Regulatory Guide 1.124, "Design Limits and Loading Combinations for Class 1 Linear-Type Component Supports," Revision 1, January 1978.

3. Regulatory Guide 1.130, "Design Limits and Loading Combinations for Class 1 Plate-and Shell-Type Component Supports," Revision 1, October 1978.

4. "Zion Probabilistic Safety Study", Commonwealth Edison Company, September 1981.

5. "Millstone Unit 3 Probabilistic Safety Study," North-East Utilities Company, August 1983.

6. NRC Staff Review of Nuclear Steam Supply System Vendor Owners Groups' Application of the Commission's Interim Policy Statement Criteria to Standard Technical Specifications. Attachment to letter dated May 1988 from T. E. Murley, NRC to W. S. Wilgus, Chairman the B&W Owners Group.

7. Regulatory Guide 8.8, "Information Relevant to Maintaining Occupational Radiation Exposure as Low as Practicable."

8. Regulatory Guide 8.10, "Operating Philosophy for Maintaining Occupational Radiation Exposure as Low as Practicable."

9. WCAP-11618, "MERITS Program-Phase II, Task 5, Criteria Application," including Addendum 1 dated April, 1989, Section 3.7.9.

10. NRC letter (D. Terao to R. Muench) dated June 2, 2006, "Wolf Creek Generating Station – Relief Request I3R-03 for the Third 10-Year Interval Inservice Inspection and Examination of Snubbers (TAC NO. MC8571)."

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iii	21	DRR 04-1535	12/1/04
TAB – B 3.0 APPLICABILTY			
B 3.0-1	3	DRR 99-1581	12/18/99
B 3.0-2	3	DRR 99-1581	12/18/99
B 3.0-3	3	DRR 99-1581	12/18/99
B 3.0-4	20	DRR 04-1410	10/7/04
B 3.0-5	20	DRR 04-1410	10/7/04
B 3.0-6	20	DRR 04-1410	10/7/04
B 3.0-7	20	DRR 04-1410	10/7/04
B 3.0-8	20	DRR 04-1410	10/7/04
B 3.0-9	20	DRR 04-1410	10/7/04
B 3.0-10	20	DRR 04-1410	10/7/04
B 3.0-11	20	DRR 04-1410	10/7/04
B 3.0-12	20	DRR 04-1410	10/7/04
B 3.0-13	20	DRR 04-1410	10/7/04
B 3.0-14	20	DRR 04-1410	10/7/04
TAB – B 3.1 REACTIVITY CONTROL SYSTEMS			
B 3.1.7-1	3	DRR 99-1581	12/18/99
B 3.1.7-2	3	DRR 99-1581	12/18/99
B 3.1.7-3	3	DRR 99-1581	12/18/99
B 3.1.8-1	11	DRR 02-0412	4/5/02
B 3.1.8-2	11	DRR 02-0412	4/5/02
B 3.1.9-1	15	DRR 03-0861	7/10/03
B 3.1.9-2	8	DRR 01-0475	5/1/01
B 3.1.9-3	8	DRR 01-0475	5/1/01
B 3.1.9-4	8	DRR 01-0475	5/1/01
B 3.1.9-5	8	DRR 01-0475	5/1/01
B 3.1.9-6	28	DRR 06-1349	7/24/06
B 3.1.9-7	28	DRR 06-1349	7/24/06
B 3.1.9-8	3	DRR 99-1581	12/18/99
B 3.1.10-1	10	DRR 02-0122	2/28/02
B 3.1.10-2	10	DRR 02-0122	2/28/02
B 3.1.10-3	13	DRR 02-1063	9/26/02
B 3.1.10-4	17	DRR 04-0452	5/26/04
TAB – B 3.3 INSTRUMENTATION			
B 3.3.3-1	23	DRR 05-0708	4/20/05
B 3.3.3-2	25	DRR 05-1996	9/28/05
B 3.3.3-3	23	DRR 05-0708	4/20/05
B 3.3.3-4	28	DRR 06-1349	7/24/06
B 3.3.3-5	28	DRR 06-1349	7/24/06
B 3.3.3-6	28	DRR 06-1349	7/24/06
B 3.3.9-1	8	DRR 01-0475	5/1/01
B 3.3.9-2	3	DRR 99-1581	12/18/99
B 3.3.9-3	8	DRR 01-0475	5/1/01
B 3.3.9-4	8	DRR 01-0475	5/1/01
B 3.3.10-1	3	DRR 99-1581	12/18/99

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TAB – B 3.3 INSTRUMENTATION (continued)			
B 3.3.10-2	3	DRR 99-1581	12/18/99
B 3.3.10-3	3	DRR 99-1581	12/18/99
B 3.3.11-1	3	DRR 99-1581	12/18/99
B 3.3.11-2	3	DRR 99-1581	12/18/99
B 3.3.11-3	3	DRR 99-1581	12/18/99
B 3.3.11-4	3	DRR 99-1581	12/18/99
B 3.3.11-5	3	DRR 99-1581	12/18/99
B 3.3.12-1	3	DRR 99-1581	12/18/99
B 3.3.12-2	19	DRR 04-1019	9/1/04
B 3.3.12-3	19	DRR 04-1019	9/1/04
B 3.3.13-1	23	DRR 05-0708	4/20/05
B 3.3.13-2	3	DRR 99-1581	12/18/99
B 3.3.13-3	3	DRR 99-1581	12/18/99
B 3.3.13-4	3	DRR 99-1581	12/18/99
B 3.3.14-1	3	DRR 99-1581	12/18/99
B 3.3.14-2	3	DRR 99-1581	12/18/99
B 3.3.14-3	17	DRR 04-0452	5/26/04
B 3.3.14-4	3	DRR 99-1581	12/18/99
B 3.3.15-1	3	DRR 99-1581	12/18/99
B 3.3.15-2	3	DRR 99-1581	12/18/99
B 3.3.15-3	3	DRR 99-1581	12/18/99
B 3.3.15-4	3	DRR 99-1581	12/18/99
B 3.3.16-1	3	DRR 99-1581	12/18/99
B 3.3.16-2	3	DRR 99-1581	12/18/99
B 3.3.16-3	3	DRR 99-1581	12/18/99
B 3.3.16-4	8	DRR 01-0475	5/1/01
B 3.3.16-5	31	DRR 07-0657	5/1/07
B 3.3.17-1	19	DRR 04-1019	9/1/04
B 3.3.17-2	3	DRR 99-1581	12/18/99
B 3.3.17-3	11	DRR 02-0412	4/5/02
B 3.3.17-4	11	DRR 02-0412	4/5/02
B 3.3.17-5	3	DRR 99-1581	12/18/99
B 3.3.18-1	14	DRR 02-1459	2/12/03
B 3.3.18-2	14	DRR 02-1459	2/12/03
B 3.3.18-3	26	DRR 06-0050	2/28/06
B 3.3.18-4	26	DRR 06-0050	2/28/06
B 3.3.18-5	26	DRR 06-0050	2/28/06
B 3.3.18-6	26	DRR 06-0050	2/28/06
B 3.3.18-7	26	DRR 06-0050	2/28/06
TAB – B 3.4 REACTOR COOLANT SYSTEM (RCS)			
B 3.4.3-1	10	DRR 02-0122	2/28/02
B 3.4.3-2	10	DRR 02-0122	2/28/02
B 3.4.3-3	10	DRR 02-0122	2/28/02
B 3.4.3-4	31	DRR 007-0657	5/1/07
B 3.4.10-1	4	DRR 00-0148	4/24/00
B 3.4.10-2	13	DRR 02-1063	9/26/02
B 3.4.16-1	3	DRR 99-1581	12/18/99
B 3.4.16-2	33	DRR 07-1554	9/28/07
B 3.4.16-3	33	DRR 07-1554	9/28/07
B 3.4.17-1	3	DRR 99-1581	12/18/99

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TAB – B 3.4 REACTOR COOLANT SYSTEM (RCS) (continued)			
B 3.4.17-2	33	DRR 07-1554	9/28/07
B 3.4.17-3	33	DRR 07-1554	9/28/07
B 3.4.17-4	33	DRR 07-1554	9/28/07
B 3.4.17-5	33	DRR 07-1554	9/28/07
B 3.4.17-6	33	DRR 07-1554	9/28/07
B 3.4.17-7	33	DRR 07-1554	9/28/07
B 3.4.17-8	33	DRR 07-1554	9/28/07
B 3.4.17-9	33	DRR 07-1554	9/28/07
B 3.4.18-1	3	DRR 99-1581	12/18/99
B 3.4.18-2	3	DRR 99-1581	12/18/99
B 3.4.18-3	3	DRR 99-1581	12/18/99
TAB – B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)			
B 3.5.1-1	3	DRR 99-1581	12/18/99
B 3.5.1-2	3	DRR 99-1581	12/18/99
B 3.5.1-3	3	DRR 99-1581	12/18/99
TAB – B 3.6 CONTAINMENT SYSTEMS			
B 3.6.1-1	26	DRR 06-0050	2/28/06
B 3.6.1-2	17	DRR 04-0452	5/26/04
B 3.6.1-3	17	DRR 04-0452	5/26/04
B 3.6.1-4	17	DRR 04-0452	5/26/04
B 3.6.1-5	17	DRR 04-0452	5/26/04
B 3.6.1-6	17	DRR 04-0452	5/26/04
B 3.6.1-7	26	DRR 06-0050	2/28/06
TAB – B 3.7 PLANT SYSTEMS			
B 3.7.7-1	8	DRR 01-0475	5/1/01
B 3.7.7-2	8	DRR 01-0475	5/1/01
B 3.7.7-3	28	DRR 06-1349	7/24/06
B 3.7.7-4	8	DRR 01-0475	5/1/01
B 3.7.8-1	25	DRR 05-1996	9/28/05
B 3.7.8-2	25	DRR 05-1996	9/28/05
B 3.7.8-3	25	DRR 05-1996	9/28/05
B 3.7.13-1	15	DRR 03-0861	7/10/03
B 3.7.13-2	5	DRR 00-0958	8/17/00
B 3.7.13-3	26	DRR 06-0050	2/28/06
B 3.7.17-1	26	DRR 06-0050	2/28/06
B 3.7.17-2	26	DRR 06-0050	2/28/06
B 3.7.17-3	26	DRR 06-0050	2/28/06
B 3.7.19-1	5	DRR 00-0958	8/17/00
B 3.7.19-2	5	DRR 00-0958	8/17/00
B 3.7.19-3	5	DRR 00-0958	12/18/99
B 3.7.20-1	3	DRR 99-1581	12/18/99
B 3.7.20-2	3	DRR 99-1581	12/18/99
B 3.7.20-3	32	DRR 07-1058	7/10/07
B 3.7.20-4	32	DRR 07-1058	7/10/07
B 3.7.20-5	32	DRR 07-1058	7/10/07
B 3.7.20-6	32	DRR 07-1058	7/10/07
B 3.7.20-7	32	DRR 07-1058	7/10/07

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TAB – B 3.7 PLANT SYSTEMS (continued)			
B 3.7.20-8	32	DRR 07-1058	7/10/07
B 3.7.20-9	32	DRR 07-1058	7/10/07
B 3.7.21-1	3	DRR 99-1581	12/18/99
B 3.7.21-2	3	DRR 99-1581	12/18/99
B 3.7.21-3	3	DRR 99-1581	12/18/99
B 3.7.21-4	3	DRR 99-1581	12/18/99
B 3.7.22-1	3	DRR 99-1581	12/18/99
B 3.7.22-2	9	DRR 01-1236	9/19/01
B 3.7.22-3	3	DRR 99-1581	12/18/99
B 3.7.22-4	3	DRR 99-1581	12/18/99
B 3.7.23-1	21	DRR 04-1535	12/1/04
B 3.7.23-2	21	DRR 04-1535	12/1/04
B 3.7.23-3	21	DRR 04-1535	12/1/04
B 3.7.23-4	21	DRR 04-1535	12/1/04
B 3.7.23-5	21	DRR 04-1535	12/1/04
B 3.7.24-1	21	DRR 04-1535	12/1/04
B 3.7.24-2	21	DRR 04-1535	12/1/04
B 3.7.24-3	21	DRR 04-1535	12/1/04
B 3.7.24-4	21	DRR 04-1535	12/1/04
B 3.7.24.5	21	DRR 04-1535	12/1/04
B 3.7.24-6	21	DRR 04-1535	12/1/04
TAB – B 3.8 ELECTRICAL POWER SYSTEMS			
B 3.8.11-1	3	DRR 99-1581	12/18/99
B 3.8.11-2	27	DRR 06-0799	5/18/06
B 3.8.11-3	27	DRR 06-0799	5/18/06
B 3.8.11-4	27	DRR 06-0799	5/18/06
B 3.8.11-5	27	DRR 06-0799	5/18/06
B 3.8.11-6	27	DRR 06-0799	5/18/06
TAB – B 3.9 REFUELING OPERATIONS			
B 3.9.7-1	3	DRR 99-1581	12/18/99
B 3.9.7-2	3	DRR 99-1581	12/18/99
B 3.9.9-1	3	DRR 99-1581	12/18/99
B 3.9.9-2	3	DRR 99-1581	12/18/99
B 3.9.9-3	17	DRR 04-0452	5/26/04
TAB – B 3.10 EXPLOSIVE GAS AND STORAGE TANK RADIOACTIVITY MONITORING			
B 3.10.1-1	3	DRR 99-1581	12/18/99
B 3.10.1-2	3	DRR 99-1581	12/18/99
B 3.10.1-3	3	DRR 99-1581	12/18/99
B 3.10.2-1	3	DRR 99-1581	12/18/99
B 3.10.2-2	3	DRR 99-1581	12/18/99
B 3.10.2-3	3	DRR 99-1581	12/18/99
B 3.10.3-1	3	DRR 99-1581	12/18/99
B 3.10.3-2	3	DRR 99-1581	12/18/99
B 3.10.3-3	3	DRR 99-1581	12/18/99
B 3.10.3-4	3	DRR 99-1581	12/18/99

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Note 1 The page number is listed on the center of the bottom of each page.

Note 2 The revision number is listed in the lower right hand corner of each page. The Revision number will be page specific.

Note 3 The change document will be the document requesting the change. Therefore, the change document should be a DRR number in accordance with AP 26A-002.

Note 4 The date effective or implemented is the date the Technical Requirement Bases pages are issued by Document Control.