UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
611 RYAN PLAZA DRIVE, SUITE 400
ARLINGTON, TEXAS 76011-8064

JAN 21 1996

EA 96-470

Neil S. Carns, President and
Chief Executive Officer
Wolf Creek Nuclear Operating Corporation
P.O. Box 411
Burlington, Kansas 66839

SUBJECT: PREDECISIONAL ENFORCEMENT CONFERENCE (NRC INSPECTION REPORT 50-482/96-21 AND NOTICE OF VIOLATION)

Dear Mr. Carns:

This refers to the predecisional enforcement conference conducted in the Region IV office on January 16, 1997. This conference related to the discussion of apparent violations identified in NRC Inspection Report 50-482/96-21 and was held at the request of Region IV.

The licensee presented a summary of the causes for the apparent violations and their corrective actions. With respect to the first apparent violation with four examples, the licensee admitted that four violations had occurred, but disagreed that there was a programmatic problem. In addition, the licensee stated that the first example was a 10 CFR 50.71(e) violation, and not a 10 CFR 50.59 violation. For the second apparent violation, the licensee admitted that the violation had occurred; however, the licensee did not believe that the violation was safety significant. The licensee indicated that the violation had the same root cause as apparent violation 3 and should, therefore, be included in the third violation. For the third apparent violation with nine examples, the licensee agreed that five of the violations had occurred, but disagreed with four of the cited examples. Although the licensee provided some information to support this view, they were not prepared to provide specific information to support the basis for this view. Nevertheless, the licensee agreed that the examples indicated a programmatic breakdown in the implementation of their corrective action program. During the meeting, the licensee agreed to supply the NRC with additional information identified in Enclosure 1.

The additional licensee information, the attendance list, the licensee's presentation, the meeting agenda, the apparent violations, the inspection report, and a copy of the safety evaluation for Plant Modification Request PMR 00903, which was provided during the conference, are enclosed to this summary.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this summary and its enclosures will be placed in the NRC Public Document Room.

Should you have any questions concerning this matter, we will be pleased to discuss them with you.

Sincerely,

Thomas P. Gwynn, Director
Division of Reactor Safety

Docket No.: 50-482
License No.: NPF-42

Enclosures:
1. Additional Licensee Information
2. Attendance List
3. Licensee Presentation
4. Meeting Agenda, Apparent Violations, Inspection Report
5. Licensee's Plant Modification Request PMR 00903

cc w/ Enclosures 1 & 2:
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bcc to DMB (IE01)

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01/ /97		01/ /97		01/ /97		01/ /97		01/21/97	

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ENCLOSURE 1

ADDITIONAL INFORMATION THE LICENSEE AGREED TO SUPPLY DURING THE ENFORCEMENT CONFERENCE

This information consisted of the following:

- A list of errors that the licensee found in the inspection report.
- Information on the applicability of the safety evaluation for Plant Modification Request PMR 00903 to the essential service water strainers differential pressure.
- The number of hours that the two centrifugal charging pumps were operable and the licensee's procedures that put one of the pumps in the pull to lock position from the second apparent violation.
- Clarifying information on examples three, four, six, seven and nine of the third apparent violation.

ENCLOSURE 2

LIST OF PERSONNEL ATTENDING EA 96-470 ENFORCEMENT CONFERENCE, JANUARY 16, 1997

PREDECISIONAL ENFORCEMENT CONFERENCE ATTENDANCE

LICENSEE/FACILITY	Wolf Creek Nuclear Operating Corporation
DATE/TIME	January 16, 1997, 1 p.m.
CONFERENCE LOCATION	Region IV, Arlington, Texas
EA NUMBER	EA 96-470

NRC REPRESENTATIVES

NAME (PLEASE PRINT)	ORGANIZATION	TITLE
Michael Vasquez	NRC RIV	Enforcement Specialist
Pat Gwynn	NRC RIV	Director, DRS
KEN BROCKMAN	NRC RGN IV	DEPUTY DIRECTOR, DIV. Rx Safety
W.H. Bateman	NRC/NRR	Project Director
Chris A. VanDenburgh	NRC/R IV	BRANCH CHIEF - ENGINEERING
Jeff F. Tedrow	NRC/SRI	Senior Resident Inspector CNS
J. Frederick Ringwald	NRC RIV	Senior Resident Inspector WC
James Stone	NRC/NRR	Senior Project Manager
PAULA GOLDBERG	NRC/RIV	Reactor Inspector
SCOTT FREEMAN	NRC/RIV	Reactor Engineer
DONALD B. ALLEN	NRC/RIV	Reactor Engineer
DAVID N. GRAVES	NRC RIV	Senior Project Engineer
W.D. Johnson	NRC RIV	Chief, Project Branch B

PREDECISIONAL ENFORCEMENT CONFERENCE ATTENDANCE

LICENSEE/FACILITY	Wolf Creek Nuclear Operating Corporation
DATE/TIME	January 16, 1997, 1 p.m.
CONFERENCE LOCATION	Region IV, Arlington, Texas
EA NUMBER	EA 96-470

LICENSEE REPRESENTATIVES

NAME (PLEASE PRINT)	ORGANIZATION	TITLE
Chris Younie	WCNOC	Mgr. Operations
Todd M. Anselmi	WCNOC	ISI Engineer
Ron Sims	WCNOC	Mgr System Engineering
Britt McKinney	WCNOC	Plant Mgr
Kevin Davison	WCNOC	Supervisor Operations Support
Richard Flannigan	WCNOC	MgR. Nuclear Engg + Licensing
Patricia Loftus	WCNOC	Mgr, Industry & Regulatory Affairs
Rick Muench	WCNOC	V.P. Engineering
Clay Warren	WCNOC	COO
Terry M. Damashek	WCNOC	Supervisor Licensing

ENCLOSURE 3

COPY OF SLIDES PRESENTED BY WOLF CREEK NUCLEAR OPERATING CORPORATION DURING PREDECISIONAL ENFORCEMENT CONFERENCE EA 96-470, JANUARY 16, 1997

Wolf Creek Nuclear Operating Corporation

Predecisional Enforcement Conference

January 16, 1997

Agenda

» Opening Comments: Clay Warren, Chief Operating Officer

» Engineering Issues: Rick Muench, Vice President Engineering

- Proposed Violation of 50.59:
  - ESW backwash setpoints
  - Frequency of RCP Flywheel Inspections
  - ESW underground piping test requirements
  - Frequency of turbine valve testing

» Operations Issues: Chris Younie, Manager Operations, and
Kevin Davison, Supervisor Operations Support

- Proposed violation of Technical Specifications due to two CCPs being Operable
- Proposed violation for inadequate corrective action from Performance Improvement Request 93-0131 regarding weaknesses found in the Technical Specification Clarification program

» Closing Statements: Britt McKinney, Plant Manager

Proposed 50.59 Violation

- WCNOC agrees that four Level IV violations occurred
- WCNOC's position is that aggregation is not consistent with NUREG 1600 due to:
  - » Unique circumstances
  - » Non-programmatic root causes
  - » No safety significance
  - » Incidence rate is relatively low

Proposed 50.59 Violation

- Three of the four examples occurred prior to implementation of the revised USQD procedure issued in 1995 and related training which occurred during 1996

- Incidence Rate
  - » 1996 E&TS results:
    - – 23 USQDs reviewed with one discrepancy in the level of documentation; however the conclusion was accurate

Proposed 50.59 Violation

- Incidence Rate
  - 29 regulatory screenings reviewed with three examples of inaccurate conclusions; one example being from 1985; one impacting 50.71(e) not 50.59
  - » Auxiliary Feedwater System Functional Assessment results:
    - 237 design packages reviewed with nine discrepancies identified - no cases where the conclusions were incorrect
- No 50.59 programmatic issue

ESW Backwash Setpoints

- WCNOC agrees that a change was made to the facility that should have been incorporated into the USAR
- USQD performed for the modification
- This is a 50.71(e) USAR update issue not 50.59
- Root Cause
  - » Personnel error in completing the Licensing Screening Form

ESW Backwash Setpoints

- Safety Significance:
  - » No safety significance: As the vendor originally recommended, strainer backwash is initiated at a pressure drop of 2 psid greater than the clean pressure drop
  - » Actual backwash pressure is significantly higher (144 psid) than the 21 psid assumed by the vendor

ESW Backwash Setpoints

- Corrective Actions
  - » USAR change request initiated
  - » 53 change packages from the same time period were reviewed with no additional USAR update errors found
  - » Confidence that the ESW SFA would have discovered this error
- Related Corrective Actions
  - » Regulatory procedures and training program changes since 1985
  - » Compliance culture training in 1997

RCP Flywheel Inspections

- WCNOC made an exception to Regulatory Guide 1.14 regarding the frequency of RCP Flywheel Inspections without prior NRC approval
- Root Cause:
  - » Personnel error: Inappropriate application of regulatory guidance

RCP Flywheel Inspections

- Safety Significance
  - » No significance: Three year UT of the RCP flywheel bore and keyway was completed satisfactorily
- Corrective Actions:
  - » Operability evaluation performed
  - » License amendment requested
  - » USAR change initiated
  - » Compliance culture training in 1997

Turbine Valve Testing

- Documentation in a USQD was incomplete to support the conclusion
- Root Cause
  - » Personnel error

Turbine Valve Testing

- Safety Significance
  - » A change in the frequency of the testing does not increase the probability of an accident
  - » Failure of the turbine does not effect safety related equipment
  - » Operating experience
- Corrective Actions
  - » USQD revised to include additional information
  - » Work product evaluations

ESW Underground Pipe Testing

- The Essential Service Water (ESW) System is a redundant system as stated in the USAR
- Each train of the ESW System is a non-redundant system
- There is no regulation, regulatory guidance, code guidance or Wolf Creek license basis that requires that these two definitions be consistent

ESW Underground Pipe Testing

- The ESW System underground piping test is consistent with the ASME Code
- The change in the ESW underground piping test does not require NRC prior approval
- In Part 9900 of the Inspection Manual, NRC reserves the right to disagree with the application of the ASME Code

ESW Underground Pipe Testing

- The NRC has verbally indicated to Wolf Creek that they do not agree with this use of the Code
- Recommend that the NRC document their position to the industry
- Wolf Creek performed an operability evaluation:
  - » ESW pump test - system pressurized to normal operating pressure

ESW Underground Pipe Testing

- » No measurable changes in system parameters which would indicate the presence of measurable leakage
- » Overall structural integrity has been evaluated to not be a problem
- » Leakage exceeding acceptance criteria would be evident

- Action
  - » Wolf Creek will revise the test procedure for Refuel IX

Summary

Issue	Date	Circumstances	Adequate Regulatory Screening	Adequate USQD Performed	USAR Updated	Relief Request/License Amendment Requested
RCP Flywheel	2/95	Misapplication of NRC Guidance	No	Yes	Yes	No
ESW Underground Piping	11/95	ASME Code Definition	No	N/A	N/A	No
ESW Backwash Sepoints	1985	Personnel Error	No	Yes	No	N/A
Turbine Throttle Valve Testing Frequency	6/96	Incomplete Documentation	Yes	No	Yes	N/A

New USQD Training/Regulatory Screening Procedure: September 1995

USQD/Regulatory Screening Training: December 1995 - December 1996

Work Product Evaluations Start: July 1996

Summary

- WCNOC agrees there are three examples of 50.59 errors and one example of a 50.71(e) error; however WCNOC disagrees that they indicate a programmatic breakdown of the 50.59 process
- WCNOC considers aggregation not consistent with the NUREG 1600, and that each of the examples is a reasonable Level IV violation

Operation's Concerns

Chris Younie and Kevin Davison
WCNOC Operations

Technical Specification Violation

- It is Wolf Creek's position that TSC 009-85 is an additional example of ineffective corrective action, not a separate violation. This is based on:
  - » All TSC problems stem from a single root cause and
  - » There are no special circumstances that separate TSC 009-85 from the TSC issues reported

Technical Specification Violation

- Technical Specification Clarification 009-85 allowed operation of both CCPs in Mode 5 which when performed in 1985 and 1994 conflicted with Technical Specifications (TS 3.5.3/3.5.4)
  - » Duration of both CCPs operable is limited to 8 minutes

Technical Specification Violation

- Safety Significance
  - » There is no safety significance to performing this evolution
  - » A single PORV has sufficient capacity to relieve the mass addition of 2 CCPs without exceeding 10 CFR 50 Appendix “G” limits

Technical Specification Violation

- The Incident Investigation Team determined the root cause was WCNOC's organizational culture was misaligned with the regulatory environment. This misalignment was evidenced in the following areas:

Technical Specification Violation

» Technical Specification Application

– Wolf Creek's "mind set" was to assess plant conditions and utilize operational knowledge in the application of Technical Specifications

» Misapplication of the TSC Process

– This misapplication resulted in instances where the clarification resulted in a change to Technical Specifications without prior regulatory approval.

Technical Specification Violation

» Standards

– This "mind set" also influenced the standards applied to TSC review, approval and internal assessment of the health of the TSC process

Technical Specification Violation

- This apparent violation should be combined with apparent violation 9621-06 as an example of inadequate corrective action since:
  - » There are no special circumstances that separate this occurrence from the other TSCs reported
  - » The root cause and corrective actions apply to all TSC concerns

Inadequate Corrective Actions

- WCNOC identified weaknesses in 1993 with specific TSCs and with the TSC process. The corrective actions did not prevent recurrence or identify all existing problems
- Wolf Creek agrees that TSC 009-85 and the other inappropriate TSCs reported constitutes inadequate corrective action

Inadequate Corrective Actions

- Safety Significance
  » TSC 009-85 allowed two CCPs to be operable in modes 4, 5, and 6
    - No safety significance since a single PORV has sufficient capacity to relieve the mass addition of two CCPs without exceeding Appendix "G" limits

  » TSC 010-85 allowed for daily containment inspections vice per entry inspections
    - No safety significance since Generic Letter 93-05 (Line Item TS Improvements) allows for daily inspections. A License Amendment Request has been generated

Inadequate Corrective Actions

- Safety Significance (continued)
  - » TSC 033-85 allowed containment penetration vent and drain valves to be opened without considering that evolution to be a breach of containment integrity provided dedicated operators were stationed to close the valves
    - Low safety significance since dedicated operators were assigned. No release resulted from LLRT activities
    - Administrative requirements existed requiring valves to be closed if direct communication with the Control Room was lost

Inadequate Corrective Actions

- Safety Significance (continued)
  - » TSC 004-86 allowed ECCS Accumulators to be considered operable based on contained volume and pressure vice absence of alarms as previously required
    - – Low safety significance since Technical Specification required water volume and pressure was maintained. Amendment 103 issued 11/22/96 allowing this condition

Inadequate Corrective Actions

- Safety Significance (continued)
  - » TSC 016-86 allowed hydrostatic testing between first off and second off boundary valves with required temperature control
    - – Low safety significance since hydrostatic pressure for these test is less than cold hydro pressure. The piping in these areas is not subject to embrittlement

Inadequate Corrective Actions

- Safety Significance (continued)
  - » TSC 005-94 EDG allowed Hot Restart testing to be separated from the 24 run if a pre-warming diesel run was conducted
    - – Low safety significance since previous surveillance requirement allowed credit for the 24 diesel run if the Hot Restart test was not successful. Amendment 101 separated Hot Restart testing from the 24 hour run

Inadequate Corrective Actions

- Root Cause
  - » The root cause for inadequate corrective action is a misalignment between the organizational culture and the regulatory environment
  - » A mind set existed which used operational knowledge in the application of Technical Specifications, and in some cases compromised literal compliance

TSC Corrective Actions

- Immediate Corrective Actions:
  - » WCNOC performed an extensive evaluation of the existing TSCs and the TSC procedure
    - – WCNOC identified five additional instances where Technical Specifications were violated. LERs were submitted for these instances
      - Three TSC were reviewed and found to <u>not</u> violate Technical Specification requirements and did not constitute a change to the existing specifications (TSC 026-85 - QPTR; TSC 001-94 - Source Range; TSC 002-96 - Source Range Power Supplies)

TSC Corrective Actions

- Immediate Corrective Actions (continued):
  - This review identified ten TSCs which were no longer needed and one TSC which was overly conservative. These TSCs have been deleted (Total of 17 deleted)
  - Three TSC were identified as needing revision
    - » Chief Operating Officer issued letter to all personnel detailing expectations for compliance with requirements

TSC Corrective Actions

- Immediate Corrective Actions (continued):
  - » An Incident Investigation Team was chartered to determined the root cause and appropriate corrective actions
- Additional Corrective Actions:
  - » Improved Standard Technical Specification program underway
  - » Safety System Functional Assessments, previously committed to, will confirm that these corrective actions are appropriate

TSC Corrective Actions

- Additional Corrective Actions (continued):
  - » All site personnel will meet with Chief Operating Officer to reinforce expectations for compliance with requirements
  - » Cultural Survey will determine content of future compliance training

Summary

- Wolf Creek's position is that apparent violations 9621-05 and 06 should be combined into a single violation of inadequate corrective action
- Wolf Creek acknowledges weaknesses in our Corrective Action Program
  » Corrective Actions for this weakness were discussed with NRC staff on December 6, 1996

Summary

- Corrective Actions taken were prompt and comprehensive
- Low safety significance
- Recent examples of site-wide identification of literal compliance issues demonstrates acceptance of implemented Corrective Actions.

Summary Statements

Britt McKinney

WCNOC Plant Manager

Summary Statements

- WCNOC Culture:
  - » Management fosters a favorable environment to raise issues
  - » Literal Compliance:
    - – October 1996: Chief Operating Officer and Plant Manager meet with PSRC
    - – November 1996: Chief Operating Officer notification (letter) to Station employees

Summary Statements

- WCNOC understands the problem
- Initiatives to improve organizational performance:
  - » NSRC and PSRC membership strengthened:
    - – New Chair on each committee
    - – Use of industry experience
    - – Line organization ownership of issues
    - – Quarterly review of selected plant safety topics
  - » System Self Assessments on key systems:
    - – Design basis review
    - – Technical Specification review
  - » USA conducting SA/QV (1/6/97 to 1/17/97)

Summary Statements

» Corrective Action Program changes:
  - Electronic initiation allows ease of tracking and trending all PIRs (10-14-96)
  - PIR coordinators assigned to each group (in place)
  - Corrective Action Review Board (in place)
  - Resource loading of PIRs (1st qtr '97)
  - FPI employee culture survey (1st qtr '97)

» FPI Root Cause and Causal Factor training for PIR coordinators and line managers (1st qtr '97)

» MARC training for supervisors/managers (1st qtr '97)

Summary Statements

- Mitigating factors:
  - » Low Safety Significance on the individual items discussed in the violations
  - » Equipment remained operable and could perform its safety function
  - » No evidence of significant scope and content breakdown/inadequacy in the USAR/Technical Specifications
  - » Control of the Licensing Basis is occurring
  - » No modifications required to restore Design Basis
  - » All occurred before listed initiatives began

ENCLOSURE 4

NRC MEETING AGENDA, APPARENT VIOLATIONS, AND INSPECTION REPORT SUPPLIED DURING PREDECISIONAL ENFORCEMENT CONFERENCE EA 96-470, JANUARY 16, 1997

PREDECISIONAL ENFORCEMENT CONFERENCE AGENDA

CONFERENCE WITH WOLF CREEK OPERATING CORPORATION

JANUARY 16, 1997

NRC REGION IV, ARLINGTON, TEXAS

1. INTRODUCTIONS/OPENING REMARKS - T.P. GWYNN, DIRECTOR DIVISION OF REACTOR SAFETY
2. ENFORCEMENT PROCESS - M. VASQUEZ, ENFORCEMENT SPECIALIST
3. APPARENT VIOLATIONS & REGULATORY CONCERNS - K.E. BROCKMAN, DEPUTY DIRECTOR DIVISION OF REACTOR SAFETY
4. LICENSEE PRESENTATION
5. BREAK (10-MINUTE NRC CAUCUS IF NECESSARY)
6. RESUMPTION OF CONFERENCE
7. CLOSING REMARKS - C. WARREN, WCNOC
8. CLOSING REMARKS - T.P. GWYNN, DIRECTOR DIVISION OF REACTOR SAFETY

APPARENT VIOLATIONS*

PREDECISIONAL ENFORCEMENT CONFERENCE

WOLF CREEK GENERATING STATION

JANUARY 16, 1997

**NOTE: THE APPARENT VIOLATIONS DISCUSSED AT THIS PREDECISIONAL ENFORCEMENT CONFERENCE ARE SUBJECT TO FURTHER REVIEW AND MAY BE REVISED PRIOR TO ANY RESULTING ENFORCEMENT ACTION.*

APPARENT VIOLATION

FIRST APPARENT VIOLATION

1. 10 CFR 50.59 (a)(1) allows the holder of a license to make changes to the facility and procedures as described in the final safety analysis report without prior Commission approval unless the proposed change involves a change in the Technical Specifications or an unreviewed safety question. 10 CFR 50.59(b)(1) requires that the licensee shall maintain records of changes to the facility and that these records include a written safety evaluation which provides the basis for the determination that the change does not involve an unreviewed safety question. 10 CFR 50.59 (c) requires licensees to submit an application for an amendment for changes which involve a change to the Technical Specifications.

   Contrary to the above,

   1. On March 13, 1984, the licensee issued Set Point Change Request EF-84-01, which changed the operation of the essential service water self cleaning strainer as described in Table 9.2-5 of the Updated Safety Analysis Report, without a determination that the change did not involve an unreviewed safety question. Specifically, Table 9.2-5 specified that the set point for initiation of the backwash was 3.0 psi whereas the set point change allowed a new setpoint of approximately 5.0 psi. This resulted in operation of the system contrary to the Updated Safety Analysis Report description through October 25, 1996.

   2. On January 11, 1995, the licensee issued Updated Safety Analysis Report Change Request 95-003 which revised Chapters 3A and 5.4.1 of the Updated Safety Analysis Report to include an exemption to Regulatory Guide 1.14, "Reactor Coolant Pump Flywheel Integrity," commitments for scheduled surface and ultrasonic examinations of reactor coolant pump flywheels. The licensee failed to properly determine that this change involved a change to the Technical Specifications. Specifically the Regulatory Guide schedule, as specified by reference in Technical Specification 4.4.10, which was superseded by Technical Specification 6.8.5.b on October 2, 1995, required the flywheel surface and ultrasonic examination at approximately 10-year intervals. This schedule was changed to 12 years without prior NRC approval. The change resulted in a failure to meet the requirements of Technical Specifications 4.4.10 and 6.8.5.b for the 10-year inspection of the "D" Reactor Coolant Pump Flywheel.

   3. On December 13, 1995, the licensee's screening for revisions to Procedures STS PE-049C, "A Train Underground Essential Service Water System Piping Flow Test," and STS PE-049D, "B Train Underground Essential Service Water System Piping Flow Test," failed to indicate that Chapter 9.2 of the Updated Safety Analysis Report was affected by the change. The procedure changes

reclassified the systems as non-redundant whereas the Updated Safety Analysis Report, provided a description of the essential service water system as redundant. As a result, the licensee failed to either submit a request for an alternative to the inservice inspection requirements or to process a change to Chapter 9.2 of the Updated Safety Analysis Report and determine whether the change involved an unreviewed safety question.

4. On March 26, 1996, the licensee performed a 10 CFR 50.59 unreviewed safety question determination regarding changing the main turbine overspeed protection test frequency from every 7 days to every 92 days, without providing supporting documentation to conclude that an unreviewed safety question was not involved. The unreviewed safety question determination did not address the licensee's experience with the testing of these valves and did not contain any information as to the acceptability, by the turbine vendor, of the decreased surveillance frequency of the turbine valves.

APPARENT VIOLATION

SECOND APPARENT VIOLATION

2. Technical Specification 3.5.4 requires one centrifugal charging pump be inoperable when in cold shutdown (Mode 5) with the reactor vessel head on.

Contrary to the above, on October 24, 1994, March 22, 1996, and March 26, 1996, the licensee maintained two centrifugal charging pumps operable while the plant was in cold shutdown with the reactor vessel head on. These actions were performed as allowed by plant procedures which were revised in accordance with licensee interpretations of Technical Specification requirements.

THIS APPARENT VIOLATION IS SUBJECT TO FURTHER REVIEW AND MAY BE REVISED

APPARENT VIOLATION

THIRD APPARENT VIOLATION

3. Criterion XVI of Appendix B to 10 CFR Part 50 requires, in part, that measures be established to assure that conditions adverse to quality are promptly identified and corrected.

Contrary to the above, on March 31, 1994, the licensee's corrective actions in response to Quality Assurance Audit K381 findings regarding the use of technical specification interpretations which could potentially conflict with Technical specification requirements or operability determinations, were not adequate to identify potential conflicts between the interpretations and the Technical Specifications. Specifically, the licensee's screenings of Technical Specification Clarifications listed below, which were performed to resolve the concerns of the Quality Assurance audit findings and which involved changes to the Technical Specifications, failed to properly determine that changes to the Technical Specifications were involved. As a result, prior Commission approval to change the Technical Specifications was not obtained prior to implementation which resulted in non-compliances with Technical Specification requirements.

1. Technical Specification Clarification 009-85 allowed two centrifugal charging pumps to be available while in cold shutdown. This clarification involved a change to Technical Specification 3.5.4 which specified only one centrifugal charging pump be operable in cold shutdown. This change was implemented without prior Commission approval. Utilization of this clarification resulted in non-compliance with the Technical Specifications on October 24, 1994; March 22, 1996; and March 26, 1996.

2. Technical Specification Clarification 010-85 allowed daily containment closeout inspections following multiple containment entries in one day. This clarification involved a change to Technical Specifications 3.5.3 and 4.5.2 which specify a containment visual inspection for loose debris be performed following each containment entry.

3. Technical Specification Clarification 026-85 allowed increasing power while the Quadrant Power Tilt Ratio exceeded the prescribed limit of 1.02. This clarification involved a change to Technical Specification 3.2.4.a.4 which prohibited increasing power with the Quadrant Power Tilt Ratio greater than 1.02.

4. Technical Specification Clarification 033-85 allowed containment penetrations be considered operable if dedicated operators were assigned to close inoperable containment isolation valves. This clarification involved a change to Technical Specification 3.6.1.1 which specified that all containment penetrations be operable by automatic isolation valves.

THIS APPARENT VIOLATION IS SUBJECT TO FURTHER REVIEW AND MAY BE REVISED

APPARENT VIOLATION

5. Technical Specification Clarification 004-86 allowed cold leg accumulators be considered operable upon receipt of level and pressure alarms if accumulator level and pressure was within prescribed limits. This clarification involved a change to Technical Specification Surveillance Requirements 4.5.1 and 4.0.3 which required the accumulators be considered inoperable upon receipt of alarms. Utilization of this clarification resulted in non-compliance with the Technical Specifications on September 25, 1996.

6. Technical Specification Clarification 001-94 allows the reactor coolant system to be cooled down, an activity which involves a positive reactivity change, with one source range channel of nuclear instrumentation inoperable. This clarification involved a change to Technical Specification 3.3.1 which specified that with one source range channel inoperable, all operations involving positive reactivity changes be suspended.

7. Technical Specification Clarification 004-94 deleted emergency diesel generator testing of the redundant diesel if the inoperable diesel was rendered inoperable by a support system failure. This clarification involved a change to Technical Specification 3.8.1.1 which specified that the redundant emergency diesel generator be tested within 24 hours if one emergency diesel generator was inoperable for any reason except for preplanned preventative maintenance, testing, or maintenance to correct a deficiency which, if left uncorrected, would not affect the operability of the diesel generator.

8. Technical Specification Clarification 005-94 allowed hot restart testing of an emergency diesel generator be performed any time before or after the 24 hour load test as long as the hot restart test was performed within 5 minutes of a 2 hour diesel run. This clarification involved a change to Technical Specification 4.8.1.1.2.g.7 which specified that a hot restart test be performed within 5 minutes following the 24 hour test. Utilization of this clarification resulted in non-compliance with the Technical Spec fications on October 15, 1994; October 17, 1994; March 23, 1996; and March 26, 1996.

9. Technical Specification Clarification 002-96 allows one of the two required source range neutron flux monitors to be considered operable when in the refueling condition when powered from a non-safety related power supply. This clarification involved a change to Technical Specification 3.9.2 which specifies that two source range neutron flux monitors to be operable and powered by its normal safety related power supply when in the refueling condition.

THIS APPARENT VIOLATION IS SUBJECT TO FURTHER REV[illegible] AND MAY BE REVISED

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
611 RYAN PLAZA DRIVE, SUITE 400
ARLINGTON, TEXAS 76011-8064

December 31, 1996

EA 96-470

Neil S. Carns, President and
Chief Executive Officer
Wolf Creek Nuclear Operating Corporation
P.O. Box 411
Burlington, Kansas 66839

SUBJECT: NRC INSPECTION REPORT 50-482/96-21 AND NOTICE OF VIOLATION

Dear Mr. Carns:

An NRC inspection was conducted October 7-11 and 21-25, 1996, at your Wolf Creek Generating Station reactor facility. The enclosed report presents the scope and results of that inspection. The overall conclusions of the inspection were discussed with Mr. O. Maynard and others of your staff on October 25, 1996. An exit meeting was held with your staff on November 8, 1996. In addition, the overall results of this inspection were discussed with Mr. Terry Damashek, on December 31, 1996.

The inspection team found numerous problems in your implementation of the 10 CFR 50.59 review process, which resulted in the use of incorrect Technical Specification clarifications, the failure to perform required inservice inspection and testing, and operating the facility differently than described in the Updated Safety Analysis Report. Design basis notebooks were found to be uncontrolled and out-of-date, which hindered your staff's ability to access design basis information. As a result, your staff had difficulty retrieving and communicating design information and using this information to support subsequent engineering calculations, modifications, adequate surveillance testing, and operability determinations.

Although system engineer knowledge was excellent, it appeared to be the result of the personal initiative taken by system engineers and their immediate supervisors, and not the result of any specific management guidance or administrative requirement. Training guidance was found to be very general and did not provide a minimum standard for system engineer training or knowledge. Communication of management expectations for system engineering had improved; however, the previous NRC engineering inspection performed in May 1995, found similar weaknesses in the management and supervisory oversight of the system engineering program, indicating ineffective corrective action.

The inspection identified several violations of NRC requirements involving: (1) failure to maintain design control, in that, the containment air cooler heat removal calculations assumed incorrect essential service water flow rates; (2) the failure to follow administrative

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procedures for performing operability determinations; and (3) the failure to implement Technical Specification surveillance requirements regarding verification of the correct position of mechanical position stops. The violations are cited in the enclosed Notice of Violation (Notice) and the circumstances surrounding the violations are described in detail in the enclosed report. Please note that you are required to respond to this letter and should follow the instructions specified in the enclosed Notice when preparing your response. The NRC will use your response, in part, to determine whether further enforcement action is necessary to ensure compliance with regulatory requirements.

The inspection also identified three apparent violations that are being considered for escalated enforcement action in accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions" (Enforcement Policy), NUREG-1600. Specifically, the first apparent violation involved four examples where your 10 CFR 50.59 safety review process failed to properly determine that changes to your facility, as described in the Updated Safety Analysis Report, and changes to the Technical Specifications were involved. As a result, a determination that an unreviewed safety question did not exist or prior NRC approval was not obtained before the changes were implemented. The second apparent violation involved plant operation in the cold shutdown condition for an extended period with two centrifugal charging pumps operable contrary to Technical Specification requirements. The third apparent violation involved inadequate corrective action for a quality assurance finding regarding the use of Technical Specification interpretations, which failed to identify and correct conflicting positions between the interpretations and the Technical Specifications. These examples indicate a potential programmatic breakdown of the design control process, which also involved a failure of the Plant Safety Review Committee to identify the problem. The examples are discussed in detail in Sections E2.2, E2.3, and E2.7 of the enclosed inspection report. In addition, please be advised that the number and characterization of apparent violations described in the enclosed inspection report may change as a result of further NRC review.

A predecisional enforcement conference to discuss these apparent violations has been scheduled for January 16, 1997. This conference will be open for public observation in accordance with a recent change to the enforcement policy (61FR65088). The decision to hold a predecisional enforcement conference does not mean that the NRC has determined that a violation has occurred or that enforcement action will be taken. This conference is being held to obtain information to enable the NRC to make an enforcement decision, such as a common understanding of the facts, root causes, missed opportunities to identify the apparent violation sooner, corrective actions, significance of the issues and the need for lasting and effective corrective action. In addition, this is an opportunity for you to point out any errors in our inspection report and for you to provide any information concerning your perspectives on: 1) the severity of the violations, 2) the application of the factors that the NRC considers when it determines the amount of a civil penalty that may be assessed in accordance with Section VI.B.2 of the Enforcement Policy, and 3) any other application of the Enforcement Policy to this case, including the exercise of discretion in accordance with Section VII.

You will be advised by separate correspondence of the results of our deliberations on this matter. No response regarding these apparent violations are required at this time.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter, its enclosure and your response will be placed in the NRC Public Document Room (PDR). To the extent possible, your response should not include any personal privacy, proprietary, or safeguards information so that it can be placed in the PDR without redaction.

Should you have any questions concerning this inspection, we will be pleased to discuss them with you.

Sincerely,

/s/

Thomas P. Gwynn, Director
Division of Reactor Safety

Docket No.: 50-482
License No.: NPF-42

Enclosures:
Notice of Violation
NRC Inspection Report
50-482/96-21

cc w/enclosures:
Vice President Plant Operations
Wolf Creek Nuclear Operating Corp.
P.O. Box 411
Burlington, Kansas 66839

Jay Silberg, Esq.
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Office of the Governor
State of Kansas
Topeka, Kansas 66612

Attorney General
Judicial Center
301 S.W. 10th
2nd Floor
Topeka, Kansas 66612-1597

County Clerk
Coffey County Courthouse
Burlington, Kansas 66839-1798

Public Health Physicist
Division of Environment
Kansas Department of Health and Environment
Bureau of Air & Radiation
Forbes Field Building 283
Topeka, Kansas 66620

Mr. Frank Moussa
Division of Emergency Preparedness
2800 SW Topeka Blvd
Topeka, Kansas 66611-1287

ENCLOSURE 1

NOTICE OF VIOLATION

Wolf Creek Nuclear Operating Corporation — Docket No.: 50-482

Wolf Creek Generating Station — License No.: NPF-42

During an NRC inspection conducted on October 7-11 and 21-25, 1996, three violations of NRC requirements were identified. In accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions," NUREG-1600, the violations are listed below:

A. 10 CFR 50, Appendix B, Criterion III, requires, in part, that measures be established to assure that regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions. These measures shall include provisions to assure that appropriate quality standards are specified and included in design documents.

Contrary to the above, on October 18, 1996, the design basis was not correctly translated into specifications for Configuration Change Package 07111, Revision 1, which was approved with an incorrect assumed essential service water flow rate. Specifically, the basis for the suitability of the containment air coolers with reduced heat removal capacity used calculations with an assumed essential service water flow rate of 4000 gpm rather than the actual flow rate of 2000 gpm available to the coolers.

This is a Severity Level IV violation (Supplement I) (50-482/96021-01).

B. Criterion V of Appendix B to 10 CFR Part 50 requires, in part, that activities affecting quality shall be prescribed by documented instructions, procedures, and drawings appropriate to the circumstances, and shall be accomplished in accordance with these instructions, procedures, or drawings.

Procedure ADM 02-024, "Technical Specification Operability," requires operability determinations to include a determination of the requirement or commitment established for the equipment.

Contrary to the above, on October 22, 1996, at 2:10 pm, the shift supervisor reviewed a statement that listed conflicting Updated Safety Analysis Report, Technical Specification and Calculation GN-MW-005 information, which pertained to containment air cooler essential service water flow rates, and performed an operability determination without including the requirement established for the equipment. Specifically, the shift supervisor relied on an out-of-date Calculation GN-MW-005, which assumed a cooler group (i.e., two coolers) flow rate of 4000 gpm, instead of determining the actual requirement for containment air cooler group essential service water flow rate of 2000 gpm.

This is a Severity Level IV violation (Supplement I)(50-482/96021-05).

C. Technical Specification 6.8.1.a states, in part, that written procedures shall be established, implemented and maintained, covering the applicable procedures recommended in Appendix A of Regulatory Guide 1.33, Revision 2.

Regulatory Guide 1.33, Appendix A, Section 3.n, requires procedures for startup, operation, and shutdown of the chemical and volume control system.

Procedure STS BG-004, "CVCS Seal Injection and Return Flow Balance," Revision 4, provides procedural guidance for setting the positions of seal injection throttle valves BGV-198, BGV-199, BGV-200, and BGV-201, and performing Technical Specification Surveillance Requirement 4.5.2.g (verifying the correct position of mechanical position stops) for these valves.

Contrary to the above, on October 23, 1996, Procedure STS BG-004 did not specifically require operators to tighten or verify the mechanical position stops for valves BGV-198, BGV-199, BGV-200, and BGV-201.

This is a Severity Level IV violation (Supplement I) (50-482/96021-06).

Pursuant to the provisions of 10 CFR 2.201, Wolf Creek Nuclear Operating Corporation is hereby required to submit a written statement or explanation to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555 with a copy to the Regional Administrator, Region IV, 611 Ryan Plaza Drive, Suite 400, Arlington, Texas 76011, and a copy to the NRC Resident Inspector at the facility that is the subject of this Notice, within 30 days of the date of the letter transmitting this Notice of Violation (Notice). This reply should be clearly marked as a "Reply to a Notice of Violation" and should include for each violation: (1) the reason for the violation, or, if contested, the basis for disputing the violation, (2) the corrective steps that have been taken and the results achieved, (3) the corrective steps that will be taken to avoid further violations, and (4) the date when full compliance will be achieved. Your response may reference or include previous docketed correspondence, if the correspondence adequately addresses the required response. If an adequate reply is not received within the time specified in this Notice, an order or a Demand for Information may be issued as to why the license should not be modified, suspended, or revoked, or why such other action as may be proper should not be taken. Where good cause is shown, consideration will be given to extending the response time.

Because the response will be placed in the NRC Public Document Room, to the extent possible, it should not include any personal privacy, proprietary, or safeguards information so that it can be placed in the Public Document Room without redaction. However, if it is necessary to include such information, it should clearly indicate the specific information that should not be placed in the Public Document Room, and provide the legal basis to support the request for withholding the information from the public.

Dated at Arlington, Texas
this 31st day of December, 1996

ENCLOSURE 2

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket No.:	50-482
License No.:	NPF-42
Report No.:	50-482/96-21
Licensee:	Wolf Creek Nuclear Operating Corporation
Facility:	Wolf Creek Generating Station
Location:	1550 Oxen Lane, NE Burlington, Kansas
Dates:	October 7-11 and 21-25, 1996
Team Leader:	J. Tedrow, Senior Resident Inspector
Inspectors:	R. Azua, Project Engineer P. Campbell, Mechanical Engineer M. Fallin, Consultant, Scientech, Inc. P. Goldberg, Reactor Inspector F. Ringwald, Senior Resident Inspector J. Stone, Project Manager
Approved By:	C. VanDenburgh, Chief, Engineering Branch Division of Reactor Safety

Attachment: Supplemental Information

TABLE OF CONTENTS

EXECUTIVE SUMMARY

Wolf Creek Generating Station
NRC Inspection Report 50-482/96-21

This team inspection evaluated the effectiveness of the Wolf Creek system and design engineering organizations to respond to routine and reactive site activities which included the identification and resolution of technical problems. The performance of safety and operability evaluations, and self-assessment activities were also included in this inspection.

Engineering

- The inspection team found that modification packages included appropriate safety evaluations, and appropriately specified post-modification testing. In addition, associated drawings and procedures were generally updated as required, and the engineering calculations were satisfactory. However, the inspection identified a design control violation regarding the use of outdated calculations for capping containment air cooler tubes. In addition, the team considered the licensee's control of the design basis information to support the safety function of the emergency core cooling system to properly operate following a postulated internal missile generation and impact to be poor (Sections E1.2 and E1.4).

- The inspection team determined that the administrative procedures that the licensee had developed for the review and evaluation of changes in accordance with 10 CFR 50.59 were appropriate. However, the team found numerous discrepancies between the Updated Safety Analysis Report and the actual plant conditions and identified problems in the licensee's implementation of the 10 CFR 50.59 review process. The team identified one apparent violation involving four examples, which were indicative of a programmatic breakdown in the control of this activity. These examples involved: (1) the operation of the essential service water self-cleaning strainer backwash setpoint differently than described in the Updated Safety Analysis Report, (2) the performance of inservice inspection and testing of the reactor coolant pump flywheel examination differently than described in the Technical Specifications, (3) the performance of underground pressure testing of essential service water piping differently than described by the Updated Safety Analysis Report, and (4) the performance of a safety evaluation regarding changing the main turbine overspeed protection test frequency without performing sufficient evaluation to conclude that an unreviewed safety question was not involved (Sections E2.2, E.2.3, and E2.7).

- Although the licensee's corrective action for a 1993 quality assurance audit required the performance of a 10 CFR 50.59 screening of Technical Specification clarifications, the screening did not identify potential conflicts between the Technical Specifications and the clarifications. Specifically, the licensee screenings of nine Technical Specification clarifications, which were performed to resolve the concerns of the quality assurance audit, failed to determine that these clarifications involved unauthorized changes to the Technical Specification requirements. In addition, a followup quality assurance audit failed to recognize that the conditions

found during the original audit were not corrected. This failure was identified as an apparent violation involving inadequate corrective action. The inspectors also noted that the screenings of the Technical Specification clarifications were subsequently reviewed by the Plant Safety Review Committee, and they also failed to identify the issues involving Technical Specification compliance (Section E2.3).

- Based on the number of findings in the 10 CFR 50.59 area and the recent indications of improper screenings for Updated Safety Analysis Report change requests, the team concluded that training did not appear to have been effective in avoiding continuing deficiencies (Section E2.3).

- The team identified that a shift supervisor violated the licensee's administrative procedures regarding operability determinations when he relied, in part, on an out-of-date calculation. Previous examples identified by NRC inspectors indicated a declining trend in the performance of on-shift operability determinations (Section E2.4).

- The team found that housekeeping was generally very good and noted that the material condition of system components had little evidence of boric acid leakage and few deficiencies. A very good threshold for deficiency identification had been established. However, the inspection team identified that system walkdowns by the safety injection system engineers did not include all plant areas where system components were located (Section E2.5).

- The team considered temporary shielding controls to be weak because they did not require an engineering review of erected temporary shielding and periodic inspections of installed temporary shielding. In addition, the residual heat removal system engineer was not knowledgeable of the condition of temporary shielding, even though it had been installed for several years (Section E2.5).

- The licensee managed the engineering open item workload appropriately, but the licensee did not have a formal program to control the backlog. The inspectors were concerned that the program had a high threshold for backlog criteria, and failed to trend the impact on engineering personnel workload (Section E2.6).

- In general, the inspection team found that surveillance tests for the systems selected had been accomplished in accordance with Technical Specification requirements and were performed at the correct periodicity. However, the team identified one violation associated with an inadequate procedure to verify emergency core cooling throttle valve mechanical position stops (Section E2.7).

- Uncontrolled and out-of-date design basis notebooks hindered the licensee's control of design basis information. The licensee's control of design basis information was found to be weak, in that, it did not provide a central location for the design basis information. In general, licensee personnel had difficulty retrieving some design basis information (Section E3.1).

- Although system engineering knowledge was excellent, it appeared to be the result of the personal initiative taken by system engineers and their immediate supervisors, and not due to any specific management guidance or administrative requirement. Training guidance was found to be very general and did not provide a minimum standard for system engineer training or knowledge. Overall, licensee management communication of system engineering expectations has improved; however, the weaknesses identified in the previous NRC engineering inspection in May 1995, had not been corrected (Sections E5.1 and E6.1).

Report Details

III. Engineering

E1 Conduct of Engineering

E1.1 General Comments (37550)

Using Inspection Procedure 37550, the team reviewed three safety-related systems to verify the licensee's ability to maintain these systems in an operable status. The three systems reviewed were: (1) essential service water, (2) residual decay heat removal, and (3) safety injection. The team reviewed the adequacy of the licensee's plant modification processes (permanent and temporary), engineering calculations, performance improvement requests, and documentation of work performed on system components.

E1.2 Permanent Plant Modification Review

a. Inspection Scope

The team reviewed several safety-related plant modification records listed in the attachment to verify conformance with applicable installation and testing requirements as prescribed by procedures. Specific attributes reviewed and/or verified by the team included: (1) 10 CFR 50.59 safety evaluations, (2) post-modification testing requirements, (3) safety-related drawing updates, (4) Updated Safety Analysis Report updates, (5) training requirements, and (6) field installation.

b. Observations and Findings

In general, the team found the modification packages reviewed included appropriate safety evaluations. The specified post-modification testing in the modification packages was appropriate and associated drawings and procedures were generally updated as required.

Outdated Calculations Used for Capping Containment Air Cooler Tubes

The essential service water system supplies the containment air coolers under accident conditions. The system contains four coolers total, with two coolers for each of two safety-related trains of essential service water. Each cooler has 12 coils with 32 circuits of 6 multiple passes, totaling 2304 tubes per cooler.

The team reviewed Configuration Change Package CCP-07111, Revision 0, which was initiated on October 17, 1996, to address a leaking tube which had developed in one of the 12 cooler coils in Containment Air Cooler SGN-01C, one of the two in the A train of essential service water. The package was issued to assess the effect of plugging (or capping) the tube and continuing to use the cooler. A 7-day action statement was entered on October 17, 1996, and an engineering review was initiated. The assessment for this change concluded that up to 64 tubes could be

plugged based on Calculations SA-90-030, CWR-02424-90, and GN-MW-005. The team noted that these calculations used a flow rate of 2000 gpm through each cooler instead of more recent calculations which were based on a flow rate of 1000 gpm through each cooler.

Change Package CCP-07111, Revision 1, was issued, and approved by the Plant Safety Review Committee, on October 18, 1996, because cooler SGN-01C continued to have leakage problems. Plans were to install a blind flange on the supply header flange and on the return header flange to the leaking coil. The one affected coil was to be abandoned in place until it could be replaced. The change package stated that the removal of one coil bundle, 32 circuits, would reduce total flow through the containment cooler pair by a maximum of 2 percent and referenced Calculation GN-MW-005, Revision 0. The change package also stated that the removal of one coil bundle would reduce the heat transfer capacity Containment Coolers SGN01A and SGN01C, by approximately 1/24, which was previously analyzed under Calculation SA-90-030. Calculations SA-90-030, dated April 23, 1990, and GN-MW-005, dated April 25, 1990, used a flow rate of 2000 gpm per cooler (4000 gpm per pair of coolers).

Change Package CCP-07111, Revision 2, was issued, and approved by the Plant Safety Review Committee, on October 20, 1996, when a second coil on cooler SGN-01C developed a leak. The package stated that one objective was to allow up to three cooling coils to be blanked off if needed. The package stated that the removal of one coil bundle, 32 circuits, will reduce total flow through the containment cooler pair by a maximum of 2 percent for a total of 6 percent with three coils removed and again referenced Calculation GN-MW-005, Revision 0. The change package also stated that a sensitivity study was performed to determine the effect of degraded performance of containment coolers on the containment pressure and temperature response following a postulated main steam line break accident. The change package referenced Calculation SA-90-025, dated April 9, 1990, which also used 2000 gpm flow through each cooler, for this sensitivity study.

Subsequent to these calculations, the licensee had identified that the essential service water system total flow had degraded due to erosion and corrosion in the system and was concerned that the analyzed flow rate to the containment air coolers, along with other cooling loads, may not be assured. Calculation SA-90-057, dated November 1990, determined the containment peak temperature and pressure that would result if the capacity of the containment air coolers were assumed to be only 45 percent of the original capacity due to a reduction in the flow rate through each cooler from 2000 to 1000 gpm, or 4000 gpm per train to 2000 gpm per train. The calculation supported Technical Specification Amendment 50, issued November 4, 1991, which changed the required minimum flow rate specified in Technical Specification 4.6.2.3.b from 4000 gpm per cooler group to 2000 gpm per cooler group. Calculation SA-90-057 concluded that sufficient heat removal capability existed with the lower flow rate.

The licensee's most recent flow balancing of the essential service water system was conducted in the 1994 refueling outage and set the measured flows, by throttling valves to the desired position, as follows:

Train A:	Cooler SGN01A	1022 gpm
	Cooler SGN01C	1034 gpm
Train B:	Cooler SGN01B	1150 gpm
	Cooler SGN01D	1440 gpm

The team determined that Calculations GN-MW-005, SA-90-025, and SA-90-30 did not reflect the current operation of the coolers (i.e., 1000 gpm current flow versus 2000 gpm flow) and predated the calculation for 1000 gpm and the subsequent Technical Specification change. Both Revisions 1 and 2 of Change Package CCP-07111 included an unreviewed safety question determination concluding that the removal of three coils from service did not constitute an unreviewed safety question. The conclusion was based on the outdated calculations discussed above. None of the referenced calculations based on a 2000 gpm flow rate for each cooler were denoted as either out-of-date or as not reflecting the current configuration of the equipment. However, the essential service water system engineer, who coordinated the efforts, was aware that the flow rate had been reduced to approximately 1000 gpm per cooler subsequent to the Technical Specification amendment.

Performance Improvement Request PIR-962669 was initiated on October 20, 1996, based on questions from the Plant Safety Review Committee on the 10 CFR 50.59 safety determination associated with Change Package CCP-07111, Revision 2. In this improvement request, the difference in the margins between the capacity of the coolers with 1000 gpm versus 2000 gpm was explained, and the impact of blocking three coils was addressed. The improvement request concluded that the containment peak pressure would not be exceeded based on Calculation SA-90-057 results. As of October 25, 1996, Change Package CCP-07111, Revision 2, had not been revised to reference the design information that reflected current operation of the coolers with a flow rate of 1000 gpm each (or 2000 gpm flow rate per a group of two coolers). However, the team considered the information provided in the improvement requests addressed the current operability conclusion of the coolers with the blocked coils (2 of 12 in the C cooler).

10 CFR 50, Appendix B, Criterion III, requires, in part, that measures be established to assure that regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions. These measures shall include provisions to assure that appropriate quality standards are specified and included in design documents. The suitability of continued use of Containment Air Cooler SGN-01C with 2 of 12 coils blocked from essential service water flow,

assessed in Change Package CCP-07111, was determined based on calculations that did not reflect the current operating configuration of the equipment (i.e., the reduction in flow requirements from 4000 gpm per cooler group to 2000 gpm per cooler group), which is considered to be a violation of 10 CFR 50, Appendix B, Criterion III (50-482/96021-01).

Licensee management stated that they considered references to outdated calculations and information to be acceptable as long as current data was utilized in present calculations. The team recognized that the licensee could have used the calculations based on 2000 gpm flow per cooler as a comparison analysis for 1000 gpm flow per cooler if the engineering analysis had stated such.

c. Conclusions

In general, the team found the modification packages reviewed included appropriate safety evaluations. The specified post-modification testing in the modification packages was appropriate and associated drawings and procedures were generally updated as required. The team identified one violation regarding the use of outdated calculations for capping containment air cooler tubes.

E1.3 Temporary Plant Modification Review

a. Inspection Scope

The team reviewed a number of the licensee's active safety-related temporary modifications listed in the Attachment. This effort was performed to verify that these modifications were in conformance with plant procedures. In addition, nonsafety-related temporary modifications were also reviewed to determine if they were appropriately categorized, and if 10 CFR 50.59 evaluations were appropriately performed.

b. Observations and Findings

The team identified that the licensee had only 14 temporary modifications installed in the plant. Of these modifications, five were identified as safety related. The team reviewed these temporary modifications against the requirements of Administrative Procedure AP 21I-001, "Temporary Modifications," Revision 1, and did not note any discrepancies. Affected procedures and drawings were also reviewed to determine if appropriate changes were annotated. No problems were noted.

The licensee had assigned an engineering supervisor to monitor temporary modifications in the plant. The licensee maintained a computerized log of these modifications, with assigned durations. The team interviewed the engineering

supervisor and found him to be cognizant of the temporary modifications installed in the plant. The team noted that this effort was designed to identify those temporary plant modifications that could be easily removed or corrected, and to make sure that long term corrective actions were applied to the remaining temporary modifications in a reasonable time.

c. Conclusions

The licensee efforts in reducing the number of temporary modifications in the plant have been very successful.

E1.4 Review of Engineering Calculations

a. Inspection Scope

The team reviewed the adequacy of several design engineering calculations listed in the Attachment associated with the three subject systems to determine whether the calculation assumptions were technically reasonable and properly supported.

b. Observations and Findings

The team found that the licensee's calculations were satisfactory. The calculations reviewed provided sufficient information and assumptions to reach the conclusion stated. The team found some minor mistakes in the calculations regarding the correct atmospheric pressure for the elevation of the plant, and conversion of pump horsepower to heat transferred to the coolant system, which did not adversely affect the calculation's conclusion. Licensee personnel were informed of these mistakes for correction.

Inadequate Support of Design Basis

The team reviewed Calculation IMS-01, "Missiles," Revision 0, to verify a statement in the Updated Safety Analysis Report, Section 6.3.1.1, regarding the design bases for the emergency core cooling system. The Updated Safety Analysis Report contained general information that stated the system was designed to withstand the effect of generated missiles. The calculation also contained an unlisted attachment which listed the summary of rotating equipment in safety-related areas, by room number. This attachment utilized Resolutions (1) and (2) which stated that room coolers and pumps were not considered to be credible missile sources based on "The Internal Missile Hazards Analysis Program Overview," Items B.4.C and B.4.A. The team requested these documents for review, but the licensee was unable to locate or retrieve them during the inspection. No other documentation was available to justify these assumptions. The team was, therefore, unable to determine if the design of the system was adequate to support the system's safety function under postulated generated missiles.

On November 8, 1996, the licensee obtained the missing information from the architect-engineer. These documents were provided to the team on November 12,

1996. The documents were hand-written and contained justification for omitting the pumps as credible missile sources due to the thickness of the pump casings. The licensee stated that they disagreed with the inspection team's finding, in that, the missing information was not part of the design bases of the plant and, therefore, need not be readily available. The team noted that the missing information was an element of the licensing basis for the emergency core cooling system as described in the Updated Safety Analysis Report, Section 6.3.1.1, in Safety Design Basis Two. Since the design basis includes information identifying the specific safety functions of the system and supporting analysis for reference bounds for the system design, the team considered the plant design basis to be inadequately supported without this documentation. Due to the difficulty the licensee experienced with retrieving this information, the team considered the licensee's control of this design information to be poor.

c. Conclusions

In general, the calculations were found to be satisfactory. The control of the design basis information to support the safety function of the emergency core cooling system to properly operate following a postulated internal missile generation and impact was considered to be poor.

E1.5 Review of Performance Improvement Requests

a. Inspection Scope

The licensee issued performance improvement requests as a means to identify problems with components and systems and to place these problems in their corrective action system for resolution. The team reviewed performance improvement requests listed in the Attachment associated with the three subject systems to determine the adequacy of the resolution, whether the systems' operability was properly determined, and that the proposed corrective actions were adequate to preclude recurrence.

b. Observations and Findings

The team found that the performance improvement requests had resolutions with proper engineering justification and that the proposed corrective actions were adequate to preclude recurrence.

E1.6 Work Package Review

a. Inspection Scope

The team reviewed work packages listed in the Attachment associated with the three subject systems, and work history printouts, to determine if repetitive problems existed and to determine the present material condition of the system. This information was compared with the results of the system walkdowns.

b. Observations and Findings

The team found that the work packages were performed in accordance with their instructions and the engineering staff was knowledgeable of the work performed. No recurrent problems were noted. The team's walkdown results indicated that the licensee was maintaining the systems in good condition and a very low threshold for deficiency identification had been established.

E2 Engineering Support of Facilities and Equipment

E2.1 General Comments (37550)

To ascertain engineering support of plant activities, the team walked down the selected systems with the system engineer, reviewed the system description provided in the Updated Safety Analysis Report, compared the Updated Safety Analysis Report description with design basis information, evaluated the engineering work backlog, compared surveillance testing records and test procedures with design basis information and Technical Specifications, and reviewed the engineering disposition of selected industry events for lessons learned.

E2.2 Review of Facility and Equipment Conformance to the Final Safety Analysis Report Description

a. Inspection Scope

A recent discovery of a licensee operating its facility in a manner contrary to the Safety Analysis Report description highlighted the need for a special focused review that compares plant practices, procedures and/or parameters to the Safety Analysis Report description. While performing the inspections discussed in this inspection report, the inspectors reviewed the applicable sections of the Final Safety Analysis Report that related to the selected inspection areas.

b. Observations and Findings

The team found that the Final Safety Analysis Report was generally consistent with the actual plant configuration. The team noted several discrepancies in the descriptions as noted below:

Improper Change to Essential Service Water Self-Cleaning Strainer Backwash Setpoint

The team reviewed Section 9.2.1, "Station Service Water System," and Table 9.2-5, "Essential Service Water System Component Data," of the Wolf Creek Updated Safety Analysis Report. The team noted that Table 9.2-5 for the essential service water system self-cleaning strainers listed a strainer capacity of 15,000 gpm with a maximum differential pressure of 3.0 psi. The team asked the licensee to verify the capacity at this differential pressure. The licensee stated that the signal to start the self-cleaning strainers was 5.0 ± 0.5 psi not the 3.0 psi stated in

Table 9.2-5. During the first week of the inspection, the licensee was not able to determine the reason for the difference in the maximum strainer differential pressure.

During the inspection, the licensee contacted the strainer vendor to determine if a maximum strainer differential pressure of 5.5 psi was acceptable. The licensee stated that setting the maximum differential pressure at 6.0 psi would not cause any physical damage to the strainer. However, it might detract from the strainers ability to self clean upon initiation of the backwash cycle. The licensee stated that the vendor indicated that a pressure drop of 1.0 psi, clean, across the strainer was based on laboratory tests and did not account for the pressure drop across the inlet and outlet connections, or specific piping connections. In addition, the vendor recommended a strainer backwash initiation at a pressure drop 2.0 psi greater than the clean pressure drop.

The team reviewed vendor data on the strainers. One chart plotted pressure loss versus flow. The team noted that for a clean strainer there was a pressure drop of 1.0 psi at a flow of 15,000 gpm. The team reviewed another plot of pressure loss versus percent of strainer clogged. The team noted that, with a differential pressure of 5.0 psi, the plot indicated that the strainer surface was 95 percent clogged. In addition, the team reviewed the licensee's data on strainer differential pressure and system flow. The team found that, since 1994, the normal differential pressure across the strainers has been approximately 3.0 to 3.5 psi and the system flow was approximately 15,000 gpm. In addition, the team reviewed startup test data from 1984 which listed a strainer differential pressure less than 1 psi at a flow over 15000 psi. The licensee could not explain what caused the pressure to increase from less than 1.0 psi in 1984 to more than 3.0 psi in 1994.

The team considered the Updated Safety Analysis Report setpoint discrepancy to be important since a change in strainer differential pressure could directly affect system flow rates. Based on reviewing the licensee's recent test data, which showed system flow greater than the design flow rate of 15,000 gpm, the team concluded that there were no operability concerns on account of the discrepancies.

10 CFR 50.59(a)(1) allows the holder of a license to make changes to the facility and procedures as described in the final safety analysis report without prior Commission approval unless the proposed change involves a change in the Technical Specifications or an unreviewed safety question.

The team reviewed setpoint Change Request EF-84-01, dated March 13, 1984. This document requested a setpoint change for the self cleaning strainer pressure instruments to change the setpoint to 5.5 psid. The cover sheet was annotated with an "N/A" following questions concerning if any Updated Safety Analysis Report section or limit was affected by the change. The modification had a 10 CFR 50.59 screening, but no safety analysis. The team found that the screening stated that the change described in the primary document did not involve a change

to the Updated Safety Analysis Report. However, the strainer table was a part of the Updated Safety Analysis Report and included the 3.0 psi maximum differential pressure for a dirty strainer. The team considered the licensee's failure to perform a safety evaluation to be the first example of an apparent violation of 10 CFR 50.59 (50-482/96021-02).

Emergency Core Cooling System Water Hammer

The team noted that Updated Safety Analysis Report, Section 6.3.2.2, stated that all emergency core cooling system discharge piping is water solid during plant operation and, therefore, water hammer in the injection line is precluded. The team questioned this statement since solid pipe operation alone will not always preclude waterhammer events depending upon the piping configuration and flow characteristics. The licensee responded by acknowledging that this statement was not appropriate. The licensee initiated Plant Improvement Request 96-2675 and stated that the Updated Safety Analysis Report would be revised to clarify the water hammer statement. The licensee provided applicable sections of the safety evaluation report which discussed how the residual heat removal system design features and proper venting and filling procedures prevented water hammer. The team concluded that no operability concern existed.

Containment Pressure Used in Pump Net Positive Suction Head Calculations

In Section 6.3.2.2 of the Updated Safety Analysis Report discussion about net-positive suction head, the statement is made that the calculation of available net-positive suction head in the recirculation mode assumes that the vapor pressure of the liquid in the sump is equal to the containment ambient pressure. This is the case only when containment ambient pressure is atmospheric in accordance with Regulatory Guide 1.1, "Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps." The actual net-positive suction head calculations use atmospheric ambient conditions. The team considered the Updated Safety Analysis Report statement to be misleading. Licensee personnel acknowledged the inspector's comment and initiated an Updated Safety Analysis Report change to clarify the wording.

Incorrect Capacity of Essential Service Water Pump Prelube Storage Tank

The team reviewed Section 9.2.1.2.2.2 of the Updated Safety Analysis Report, which stated that the essential service water prelube storage tank size was based on supplying a minimum of 6 gpm water for 5 minutes to the essential service water pump bearings without any makeup from the essential service water line. The team asked the licensee how they verify this statement.

The licensee verified that the tank would hold enough water to supply 30 gallons of water without any makeup. However, the licensee determined that the maximum flow to the bearings would only be 1.0 to 1.5 gpm due to the size of the piping from the prelube tank to the pump bearings. The licensee stated that there was no operability concern since the pump vendor had installed bronze bearings in the

pump because of the possibility for pump start without prelubrication. Therefore, the tank was not needed for pump operability requirements. In addition, the team determined that the licensee did not know the necessary flow rate of water to properly lubricate the bearings as recommended by the pump vendor to reduce wear. The team noted that Table 9.2-5 listed the capacity of the prelube tank to be 43 gpm. The team determined that 43 gallons was the volume of the tank with a usable volume of 35 gallons. The licensee prepared Plant Improvement Request 96-2617, dated October 16, 1996, to resolve these discrepancies and correct the Updated Safety Analysis Report.

c. Conclusions

Although there were numerous discrepancies between the Updated Safety Analysis Report and the actual plant conditions, the inspection team determined that the discrepancies did not present an operability concern. The inspection team identified one apparent violation regarding operation of the essential service water self-cleaning strainer backwash setpoint differently than described in the Updated Safety Analysis Report. In addition, the team noted that the licensee had difficulty in retrieving design information.

E2.3 10 CFR 50.59 Implementation (37001)

a. Inspection Scope

The team reviewed the licensee's program guidance, training program information, a sample of 50.59 screenings and associated unreviewed safety question determinations, a sample of 50.59 screenings that did not require an unreviewed safety question determination, and interviewed a number of individuals who perform 50.59 screenings and prepare unreviewed safety question determinations. In addition, a sample of Updated Safety Analysis Report changes were reviewed.

b. Observations and Findings

The licensee's safety evaluation process for changes to the facility is controlled by Procedure AP 26A-003, "Screening and Evaluating Changes, Tests, and Experiments," Revision 1. This procedure was recently revised in February 1996. The procedure delineated the licensee's methods, training requirements, and responsibilities to determine and document whether facility changes can be made without prior NRC approval. The process used to determine if an unreviewed safety question exists is a two step process.

- The first step was a screening process that made a determination as to whether or not the proposed change was a change to the facility as described in the Technical Specifications, Updated Safety Analysis Report, nonradioactive liquid or gaseous discharges, nonradiological solid waste, thermal discharges, security plan, safeguards contingency plan, security guard training plan, radiological emergency plan, an NRC or INPO commitment, and physical changes within the site boundaries. If the answer

to all questions was negative, then a change to the Updated Safety Analysis Report was deemed not to exist and the change could proceed without an unreviewed safety question determination prepared. An affirmative answer to any of the questions required further evaluation. Only if the screening determined that it was a change to the Updated Safety Analysis Report, was an unreviewed safety question determination required.

- The second step involved documentation of an unreviewed safety question determination on Form APF 26A-003-03, "10 CFR 50.59 Unreviewed Safety Question Determination," by answering a series of questions and recording the basis for each answer. If the answer to all questions was "no," then an unreviewed safety question did not exist and the change could be implemented without prior approval of the NRC. If the answer to any question was "yes," then NRC approval was required prior to implementing the proposed change. Procedure AP 26B-003, "Revisions to the Updated Safety Analysis Report," provided instructions for issuing changes to the Updated Safety Analysis Report.

The team determined that these procedures provided appropriate guidance for the development and approval of reviews and approvals under 10 CFR 50.59.

The licensee developed a training program for personnel that performed 50.59 screenings and prepared unreviewed safety question determinations. The team's review of the training program determined that the program covered all the essential aspects of the 50.59 screenings and unreviewed safety question determinations. In addition, there was a requirement that by the end of calendar year 1996, personnel performing 50.59 screenings and preparing unreviewed safety question determinations must have taken the training. The need for requalification training will be determined by significant changes to Procedure AP 26A-003, an increasing trend in the number of Plant Improvement Requests indicating deficiencies in completed screenings or unreviewed safety question determinations, self-assessment results and quality assurance audit results.

The team evaluated the implementation of the 50.59 program by reviewing a sample of completed 50.59 screenings and determinations as contained in the Wolf Creek Generating Station Annual Safety Evaluation Report for 1995, a listing of the changes approved since January 1, 1996, and interviewing a number of personnel involved in the preparation of 50.59 screenings and determinations. Several deficiencies were identified as delineated below:

Inadequate Justification of Change to Turbine Overspeed Protection

Unreviewed Safety Question Determination 59 96-0067 and associated Updated Safety Analysis Report Change Request 96-044 evaluated and changed the surveillance frequency for the four high pressure turbine stop valve, six low pressure turbine reheat stop valves and six low pressure reheat intercept valves from once per seven days to once per 92 days. This change was based on NRC's Generic Letter 93-05, "Line-Item Technical Specifications Improvements to Reduce

Surveillance Requirements for Testing During Power Operations." The guidance provided in the generic letter for changing the turbine valve surveillance frequency, requested that licensees include a statement in their amendment request that the proposed change is compatible with plant operating experience and a statement that the turbine manufacturer concurred with the proposed change. However, the inspection team noted that the unreviewed safety question determination did not address the licensee's experience with the testing of these valves and did not contain any information as to the acceptability, by the turbine vendor, of the decreased surveillance frequency of the turbine valves. Based upon interviews with licensee personnel, the team determined that the licensee had not fully considered these factors and that the turbine vendor had not been contacted.

10 CFR 50.59 (b)(1) requires that records of changes include a written safety evaluation which provides the bases for the determination that the change, test, or experiment does not involve an unreviewed safety question. Even though the turbine test frequency change did not involve a license amendment, the licensee should have been aware of the specific information the NRC deemed appropriate to include in this unreviewed safety question determination based on the generic letter. Therefore, the team determined that the basis included with this change did not provide adequate information to come to the conclusion that an unreviewed safety question did not exist. The team considered the failure to fully evaluate that the change did not involve an unreviewed safety question to be the second example of an apparent violation of 10 CFR 50.59 (50-482/96021-02).

The licensee subsequently informed the team that the information needed to justify the change did not involve an unreviewed safety question was available and the determination would be revised to include it.

<u>Inadequate Screenings of Technical Specification Clarifications</u>

The team reviewed several proposed Updated Safety Analysis Report changes, including three that would have incorporated Technical Specification clarifications into the Updated Safety Analysis Report. These clarifications had been screened and determined to neither change the Updated Safety Analysis Report nor the Technical Specifications and had been issued for review and approval.

Change Request 96-094 was written to add existing Technical Specification Clarification 009-85 for a Technical Specification that had been relocated to Chapter 16 of the Updated Safety Analysis Report. The clarification allowed closing the breaker and operation of a second centrifugal charging pump while swapping pumps when in operating Modes 4, 5, or 6. The team reviewed current Technical Specifications 3/4.5.3 (applicable in Mode 4) and 3/4.5.4 (applicable in Modes 5 and 6) and determined that both allowed only one charging pump to be operable. On October 2, 1995, a change to Technical Specification 3/4.5.4 (Amendment 89) was made that added a 4-hour action period to disable one pump.

The team determined that the licensee had changed Operating Procedure SYS BG-201, "Shifting Charging Pumps," in 1985 to incorporate the Technical

Specification clarification. The clarification received a further screening in March 1994 as a result of a quality assurance finding. The team was informed that the operating procedure had been previously used, and the 4-hour action period exceeded, on March 22 and 26, 1996. In addition, during two occasions on October 24, 1994, while the plant was in Mode 5, both charging pumps were operable. The team considered the initial screening done for the operating procedure and the subsequent screening done for this clarification in 1994 to be inadequate as they changed a Technical Specification requirement and resulted in operation of a second charging pump while in Mode 5, contrary to Technical Specification 3.5.4. Failure to perform the required actions of Technical Specification 3.5.4 is considered to be an apparent violation of the Technical Specification (50-482/96021-03).

The licensee subsequently voided this proposed change request and the Technical Specification clarification. A revision to the operating procedure was also initiated to prohibit this action.

Following the identification of the team's concerns about Technical Specification clarifications, the licensee formed an internal investigation team to review and determine the adequacy of all 45 active clarifications and whether or not compliance with Technical Specification requirements was being achieved. As a result of that continuing review, the licensee identified two additional clarifications which were improperly screened and that resulted in Technical Specification non-compliance as follows:

- Technical Specification Clarification 004-86 allowed cold-leg accumulators to be considered operable upon receipt of level and pressure alarms if accumulator level and pressure was within prescribed limits. This clarification involved a change to Technical Specification Surveillance Requirements 4.5.1 and 4.0.3, which required the accumulators be considered inoperable upon receipt of alarms.

  The licensee determined that from September 25 to October 2, 1996, the associated level alarm was energized and the Technical Specification action statement was not met because of the failure of one level indication channel on Cold Leg Accumulator B. The Technical Specification action statement required restoration within 1 hour or hot shutdown within the next 6 hours followed by reactor coolant system depressurization below 1000 psig within the next 6 hours. The team noted, however, that the alarm function did not affect the ability of the system to perform its safety function.

- Technical Specification Clarification 005-94 allowed hot restart testing of an emergency diesel generator to be performed any time before or after the 24-hour load test, as long as the hot restart test was performed within 5 minutes of a 2-hour diesel run. This clarification involved a change to Technical Specification 4.8.1.1.2.g.7, which specified that a hot restart test be performed within 5 minutes following the 24-hour test. There was a footnote to the Technical Specification that allowed the hot restart test to be

done following a warmup run if it failed the hot restart test following the load test. This clarification allowed the complete decoupling (i.e., allowing the hot restart test to be performed anytime after engine warmup and not requiring a failure of the hot restart test following the load test) of the load test and the hot restart test. This Technical Specification was changed by the NRC with Amendment 101, issued on August 8, 1996, and allows the decoupling of these two requirements. This amendment was implemented by the licensee on November 7, 1996.

The licensee determined that prior to issuance of this amendment, hot restart testing of the diesels was not performed in accordance with the Technical Specifications. Specifically, during Refueling Outage 7, Emergency Diesel Generator A was load tested on September 17, 1994, and the hot restart test was not performed until October 15, 1994. Emergency Diesel Generator B was load tested on September 16, 1994, and the hot restart test was not performed until October 17, 1994.

The licensee also determined that during Refueling Outage 8, Emergency Diesel Generator A was load tested on February 6, 1996, and the hot restart test was not performed until March 26, 1996. Emergency Diesel Generator B was load tested on March 16, 1996, and the hot restart test was not performed until March 23, 1996. Again, since the licensee's hot restart test method was allowed by the Technical Specifications under certain conditions, the team considered the consequences of these violations to be minor.

In addition, the team evaluated the licensee's review of all the clarifications and identified the following clarifications that provided guidance contrary to Technical Specification requirements and could have resulted in non-compliance due to inadequate screenings:

- Technical Specification Clarification 010-85 allowed daily containment closeout inspections following multiple containment entries in one day. This clarification involved a change to Technical Specifications 3.5.3 and 4.5.2 which specify a containment visual inspection for loose debris be performed following each containment entry.

- Technical Specification Clarification 026-85 allowed increasing power while the quadrant power tilt ratio exceeded a prescribed limit. This clarification involved a change to Technical Specification 3.2.4.a.4 which prohibited increasing power with the quadrant power tilt ratio greater than the prescribed limit.

- Technical Specification Clarification 033-85 allowed containment penetrations to be considered operable if dedicated operators were assigned to close inoperable containment isolation valves. This clarification involved a change to Technical Specification 3.6.1.1 which specified that all containment penetrations be operable by automatic isolation valves.

- Technical Specification Clarification 001-94 allows the reactor coolant system to be cooled down, an activity which involves a positive reactivity change, with one source range channel of nuclear instrumentation inoperable. This clarification involved a change to Technical Specification 3.3.1, Table 3.3-1, Functional Unit 6.b, "Source Range Shutdown," Action 5, which specified that with one source range channel inoperable, all operations involving positive reactivity changes be suspended.

- Technical Specification Clarification 004-94 deleted emergency diesel generator testing of the redundant diesel if the inoperable diesel was rendered inoperable by a support system failure. This clarification involved a change to Technical Specification 3.8.1.1 which specified that the redundant emergency diesel generator be tested within 24 hours if one emergency diesel generator was inoperable for any reason except for preplanned preventive maintenance, testing, or maintenance to correct a deficiency which, if left uncorrected, would not affect the operability of the diesel generator. This clarification extended this footnote to include inoperable support systems on one diesel as a condition that would not require a start test of the other diesel. This Technical Specification was changed by the NRC with Amendment 101, issued on August 8, 1996, and was implemented by the licensee on November 7, 1996.

- Technical Specification Clarification 002-96 allows one of the two required source range neutron flux monitors to be considered operable when in the refueling condition when powered from a nonsafety-related power supply. This clarification involved a change to Technical Specification 3.9.2, which specifies that two source range neutron flux monitors to be OPERABLE in the refueling condition (Mode 6). Although Technical Specification 3.9.2 does not specify the power source requirement, the definition of OPERABILITY does include a requirement for electric power, which refers to the normal safety-related power supply.

The licensee provided the result of an audit done of the existing clarifications by their quality assurance group in February 1993. This audit identified the following potential consequences that could result in the use of Technical Specification clarifications:

- Failure to comply with Regulatory, Technical Specification, or other applicable requirements;

- Poor performance ratings, concerns, or more severe actions from the NRC for a potentially inadequate or incorrect Technical Specification clarification program;

- Inappropriate actions being taken by operators;

- Potentially non-conservative actions which could require NRC approval prior to implementation; and/or

- Overly conservative actions for plant shutdown without consideration of other risks involved.

As a result of that audit, the licensee reviewed membership on the Technical Specification clarification committee for appropriateness; reviewed guidance for preparation of clarifications; and performed a 10 CFR 50.59 review (screenings) of all current clarifications. In addition, the screenings of the clarifications were reviewed and approved by the Plant Safety Review Committee. These activities resulted in voiding eleven clarifications, revision of six clarifications, and one clarification was considered for a Technical Specification amendment. The remaining clarifications were deemed by the licensee to meet requirements. This action was completed in March 1994. The quality assurance group performed a follow up audit to evaluate the effectiveness of the corrective actions which concluded that the corrective actions were adequate to resolve the concern. This audit and review of the completed corrective actions failed to identify additional potential conflicts between the clarifications and Technical Specifications.

10 CFR 50, Appendix B, Criterion XVI, requires in part, that measures be established to assure that conditions adverse to quality are promptly identified and corrected. The team determined that the licensee's corrective actions, done following the quality assurance finding, were inadequate and failed to identify the conflicting statements in the clarifications with the Technical Specifications. Based upon the numerous deficiencies in this area, the team concluded that a programmatic breakdown in the licensee's 10 CFR 50.59 screening program had occurred. This breakdown included the licensee's quality assurance group which initially identified potential concerns with the clarifications, but did not properly assess the adequacy of the licensee's corrective action, and the Plant Nuclear Safety Review Committee which reviewed the clarification screenings and also failed to note that changes to the Technical Specifications were involved. The failure to perform adequate corrective action for the identified clarification deficiencies is contrary to the requirements of 10 CFR 50, Appendix B, Criterion XVI, and is considered to be an apparent violation (50-482/96021-04).

At the time of the exit meeting on November 8, 1996, the licensee had reviewed the clarifications and determined that occasions had occurred in which the Technical Specifications were violated and planned to submit five licensee event reports on these items.

Improper Change to Reactor Coolant Pump Flywheel Inspection Frequency

The team reviewed Updated Safety Analysis Report Change Request 95-003, "Screening for Licensing Basis Changes," approved January 11, 1995, regarding a change in the examination schedule for the reactor coolant pump flywheels. Specifically, the description of the proposed change stated that Regulatory Guide 1.14, Revision 1, required a 10-year reactor coolant pump motor flywheel

examination coinciding with the inservice inspection program interval. This change clarified the intended examination schedule by revising Chapters 3A and 5.4.1 of the Updated Safety Analysis Report to include an exception to the Regulatory Guide examination schedule. The examination schedule was changed to 12 years to accommodate the "D" reactor coolant pump flywheel which had not been inspected per the previously established schedule. The response to Screening Question 2 on whether the change results in a revision to the Operating License, including the Technical Specifications, was marked "No."

Technical Specification 4.4.10, which was applicable January 9, 1995, stated that each reactor coolant pump flywheel shall be inspected in accordance with the recommendations of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, August 1975. This Technical Specification was subsequently superseded by Technical Specification 6.8.5.b in License Amendment 89, issued October 2, 1995, which contained the same statement. Regulatory Guide 1.14, "Reactor Coolant Pump Flywheel Integrity," Revision 1, 1975, Paragraph C.4.b.(2) states that a surface examination of all exposed surfaces and complete ultrasonic volumetric examination of the flywheel be performed at approximately 10-year intervals, during the plant shutdown coinciding with the inservice inspection schedule as required by Section XI of the ASME Code.

The interval for inservice inspection is based on 120 months pursuant to 10 CFR 50.55a(g)(4), with the initial interval beginning on the date of commercial operation. Commercial operation for the Wolf Creek plant commenced September 3, 1985. Provisions in Paragraph IWA-2400(c) allowed that each inspection interval may be decreased or extended by as much as 1 year. The provisions of Paragraph C.4.b of Regulatory Guide 1.14 specified that the surface and ultrasonic examination of the flywheel be performed ". . . at approximately 10-year intervals." Therefore, using the code provisions for the inservice inspection interval, the surface examination of all of the reactor coolant pump flywheels should have been completed by September 3, 1996. The licensee confirmed on October 25, 1996, that the surface and ultrasonic examination of the "D" reactor coolant pump flywheel has not yet been performed and is currently scheduled for the Fall 1997 refueling outage during reactor coolant pump maintenance.

Section 50.59, "Changes, Tests, and Experiments," allows licensees to make changes to licensed facilities or to perform tests and experiments at licensed facilities when these changes, tests, and experiments (1) do not change the parameters specified in the facility operating license, including Technical Specifications, or (2) present an unreviewed safety question. If the changes, tests, or experiments change the operating license, including Technical Specifications, or present an unreviewed safety question, NRC approval is required prior to implementing the change or performing the tests or experiments. By reference in the Technical Specifications, any exceptions to the reactor coolant pump motor flywheel inspection program delineated in paragraph C.4.b of Regulatory Guide 1.14, must be approved by the NRC.

examination coinciding with the inservice inspection program interval. This change clarified the intended examination schedule by revising Chapters 3A and 5.4.1 of the Updated Safety Analysis Report to include an exception to the Regulatory Guide examination schedule. The examination schedule was changed to 12 years to accommodate the "D" reactor coolant pump flywheel which had not been inspected per the previously established schedule. The response to Screening Question 2 on whether the change results in a revision to the Operating License, including the Technical Specifications, was marked "No."

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Section 50.59, "Changes, Tests, and Experiments," allows licensees to make changes to licensed facilities or to perform tests and experiments at licensed facilities when these changes, tests, and experiments (1) do not change the parameters specified in the facility operating license, including Technical Specifications, or (2) present an unreviewed safety question. If the changes, tests, or experiments change the operating license, including Technical Specifications, or present an unreviewed safety question, NRC approval is required prior to implementing the change or performing the tests or experiments. By reference in the Technical Specifications, any exceptions to the reactor coolant pump motor flywheel inspection program delineated in paragraph C.4.b of Regulatory Guide 1.14, must be approved by the NRC.

The team considered a change to the examination schedule would result in a change to the Technical Specifications by reference in paragraph C.4.b of Regulatory Guide 1.14. Therefore, the proposed change to the examination would require NRC approval prior to implementing the change. Failing to properly perform the screening for the proposed change to the surface examination schedule for the reactor coolant pump flywheels to identify a change to the Technical Specification is contrary to 10 CFR 50.59 and is considered to the third example of the apparent violation discussed in Section E2.2 of this report (50-482/96021-02).

After being informed of this discrepancy, the licensee performed an operability determination for the "D" reactor coolant pump which concluded that the pump was capable of performing its safety related design function. This determination was based upon satisfactory examination results of the flywheel keyways and bore which were last performed during Refueling Outage 7. In addition, nuclear industry experience has indicated that a decrease in inspection requirements is appropriate in some cases. Based upon this information and consultation with the Office of Nuclear Reactor Regulation, the team concluded that continued operation of the pump until the examination could be performed was not a safety concern.

c. Conclusions

Numerous problems were identified with the licensee's implementation of the 50.59 review process, which were indicative of a programmatic breakdown. Further evidence of a continuing breakdown in the review process was evident by the existence of changes made since 1994 in which the licensee did not recognize changes to the Technical Specifications (reactor coolant pump flywheel issue) or other NRC approved programs (essential service water system buried pipe testing discussed in Section E2.7).

The team determined that the program procedures the licensee has developed for the review and evaluation of changes in accordance with 10 CFR 50.59 were appropriate. Based on the number of findings in the 50.59 area, and the recent indications of improper screenings for Updated Safety Analysis Report change requests, the team concluded that training did not appear to have been effective in avoiding continuing deficiencies.

The licensee's corrective action for a quality assurance audit, initiated in 1993, identified potential problems with the use of Technical Specification clarifications, did not identify potential conflicts between the Technical Specifications and the clarifications. The followup audit by quality assurance failed to recognize that the conditions found during the original audit finding were not corrected. This was considered to be an apparent violation involving inadequate corrective action. In addition, the review of the clarifications by the Plant Safety Review Committee, and their failure to identify continuing issues involving Technical Specification compliance, calls into question the performance of that group.

E2.4 Unsupported Operability Determination

a. Inspection Scope (37550)

The team reviewed one operability determination made during the inspection by a shift supervisor associated with team observations.

b. Observations and Findings

On October 22, 1996, the team noted that the shift supervisor reviewed an informal listing of inspection issues raised by the team. Item 133 noted that several different documents, Technical Specification requirements, Updated Safety Analysis Report sections, and a calculation identified conflicting essential service water flows through the containment air coolers. Item 133 also identified two questions regarding the correct number for essential service water flow through the containment air coolers and, the correct number for heat removal rate of a single containment air cooler. The shift supervisor reviewed this listing, then logged the following entry into the Shift Supervisor Log: "1410 Reviewed Items 130-134 on Engineering and Technical Services NRC Inspection list - No operability/reportability issues noted."

The team asked the shift supervisor what the basis was for the log entry identifying no operability issues for Item 133. The shift supervisor stated his basis was Calculation GN-MW-005, Revision 2, which used 4000 gpm flowrate per cooler group, and that the assumption had been made that, "...the engineers knew what they were doing." The team noted that the flow information used by this calculation had been superseded, and that the present containment cooler flow was 2000 gpm flowrate per cooler group. The team questioned the engineer regarding how the list had been presented to the shift supervisor. The engineer stated that the list had been handed to the shift supervisor, and that there had been no substantive discussion regarding Item 133.

Administrative Procedure ADM 02-024, "Technical Specification Operability," Revision 3, step 5.3.2, required the shift supervisor to perform a number of actions associated with the operability determination to ensure sufficient scope of review. This step required the shift supervisor to determine the requirement or commitment established for the equipment, and why the requirement or commitment may not be met. In cases where the operability determination was not straightforward, Procedure ADM 02-024 also required the shift supervisor to use the information available to make the determination, and start the actions stated in Procedure AP 28-001, "Evaluation of Nonconforming Conditions of Installed Plant Equipment," Revision 4, to obtain sufficient information to completely answer all questions.

The team determined that the operability evaluation performed by the shift supervisor failed to include all the required actions, in that, the shift supervisor did not properly identify the minimum acceptable flow rate for the containment air cooler given the conflicting statements of containment air cooler flow in the

Updated Safety Analysis Report and other documents, and compare the actual cooler flow with the minimum flow requirement as stated in Technical Specification 4.6.2.3.b. This is a violation of 10 CFR 50, Appendix B, Criterion V (50-482/96021-05).

The inspection team noted that NRC Inspection Reports 50-482/96-012, 50-482/96-11, and 50-482/96-09, had previously identified several examples where the NRC had identified poorly supported operability determinations. The team determined that while the previous examples of poorly supported operability evaluations were not identified as violations of requirements, they indicated a declining trend in performance. The violation identified in this paragraph was determined to be more significant than the previous examples, in that, the shift supervisor stated that the operability determination was, at least in part, based on an out-dated calculation and an unsupported reliance on engineering.

c. Conclusions

The team concluded that the shift supervisor violated 10 CFR 50, Appendix B, Criterion V, when an operability determination failed to comply with the licensee's procedure on operability determinations, and relied, at least, in part, on an out-dated calculation. Previous examples identified by NRC inspectors indicated a declining trend in the performance of operability determinations on shift.

E2.5 System Walkdowns (37550)

a. Inspection Scope

The team performed a walkdown of the three subject systems and other selected plant areas to determine the overall material condition of equipment and maintenance of housekeeping. In addition, the team walked down several portions of the spent fuel pool cooling system, component cooling water system, and instrument air system.

b. Observations and Findings

The team found the housekeeping was generally very good. The team noted that the system engineers and design engineers were both knowledgeable of their systems. The engineers demonstrated their knowledge during the walkdown by explaining component deficiencies in detail and relating to the team specific operational problems with system operation. The material condition of system components was noted to be very good with little evidence of boric acid leakage and few deficiencies noted during the walkdown. The team noted that several minor system leaks had been previously identified by licensee personnel which indicated that a very good threshold for deficiency identification had been established.

The team reviewed the system engineers' notebooks for the three systems selected. The team noted that these notebooks were maintained in a well organized manner, and the separate sections were tabbed for easy reference. The safety injection system engineer kept the trend data and system walkdown sheets current, and had a sufficient breadth of material to support the stated description of system engineer responsibilities.

The team asked the safety injection system engineer what the maintenance rule performance goals and actual system performance was for the safety injection system. Both the present and former system engineers knew that the safety injection system performance was exceeding the goal by a wide margin. However, neither engineer could readily identify the actual system performance statistics without speaking with the maintenance rule coordinator. While the team did not view this as a significant weakness, it did indicate that in this case the system engineers did not have ready access to current maintenance rule performance statistics for their system.

Safety Injection System Engineer System Walkdown

The team noted that the safety injection system engineer had been assigned to this system 8 weeks prior to the inspection. During this period, the system engineer had conducted only one joint walkdown with the previous system engineer. The system engineer conducted system walkdowns approximately weekly, but management only required these walkdowns biweekly. The system engineer's supervisor had participated in one of these walkdowns.

During the walkdown with the team, the system engineer did not tour the 1988 foot elevation of the auxiliary building and was, therefore, unaware of a flange leak on the suction line between the refueling water storage tank and the common suction header supplying the eight emergency core cooling pumps. When asked by the team, the prior system engineer stated that walkdowns had included portions of the 1988 elevation of the auxiliary building, but had never included the high radiation area encompassing the pipe chase area. The system engineer indicated that these walkdowns took from 1.5 to 2.5 hours each, but that during some weeks the system engineer would take credit for system engineer presence in the field supporting maintenance as the system walkdown for the week. With the exception of the 1988 elevation of the auxiliary building, the system engineer's walkdown was adequate.

The team discussed with licensee management their expectations for system engineering walkdowns. Management stated that they expected the system engineers to perform walkdowns in all areas containing system components, although less frequently for high radiation areas due to exposure concerns.

Residual Heat Removal Temporary Shielding

The team noted that temporary shielding had been erected on the hot-leg suction piping for both trains of residual heat removal cooling and asked about this situation and potential impact on system operability. The system engineer stated that this shielding was installed in 1991 per a temporary shielding request. The team reviewed the shielding request and scaffolding permits which controlled the erection of scaffolding used to support the shielding off of the system piping. The team noted that the scaffolding permits did not address potential static loads which might be applied if the plastic straps which held the shielding to the scaffolding should fail. Licensee personnel acknowledged this deficiency in the scaffolding evaluation and inspected the erected scaffolding and temporary shielding. Licensee personnel found that portions of the shielding were not secured by tie wraps as specified in the evaluation and decided to remove the scaffolding pending completion of a new evaluation.

The licensee completed a subsequent evaluation which determined that the secured and unsecured shielding would not have adversely affected safety related piping underneath the scaffolding. The team determined that the erected scaffolding and shielding had not been reviewed by engineering personnel and the system engineer was not knowledgeable of the condition of this temporary shielding even though it had been installed for several years. The team considered the temporary shielding controls to be weak for not requiring an engineering review of erected temporary shielding and periodic inspections of installed temporary shielding. The licensee subsequently revised Procedure AP 25A-700, "Use of Temporary Lead Shielding," to require periodic inspections, verify shielding installation conformed with the engineering disposition, and evaluation of the need for permanent shielding if temporary shielding is installed for 6 months.

c. Conclusions

The team found the housekeeping was generally very good. The team noted that, in general, system engineers and design engineers were very knowledgeable of their system. The material condition of system components was noted to be very good with little evidence of boric acid leakage and few deficiencies. A very good threshold for deficiency identification had been established. System walkdowns by the safety injection system engineers did not include all plant areas were system components were located.

The team considered temporary shielding controls to be weak for not requiring an engineering review of erected temporary shielding and periodic inspections of installed temporary shielding. The residual heat removal system engineer was not knowledgeable of the condition of temporary shielding even though it had been installed for several years.

E2.6 Engineering Work Backlog (37550)

a. Inspection Scope

The team discussed the status of the engineering backlog with the Assistant to the Vice President of Engineering. The discussions included actions taken by the engineering organization to reduce the backlog.

b. Observations and Findings

The licensee's engineering backlog program was managed by the Assistant to the Vice President of Engineering. The team interviewed the program manager and found him to be knowledgeable of his responsibilities, but noted that no one had been assigned backup responsibilities for this effort. This observation was compounded by the fact that this program was not procedural[illegible], and that the data was manually collected and tracked. Therefore, the team considered the program to be very susceptible to personnel changes in the organization. In addition, it was noted that the open item information collected had not been trended to determine the overall impact the open items had on the engineering department workload.

The licensee's engineering backlog listed only 65 open items. The team found this number to be artificially low because the licensee's threshold for backlog items was high (i.e., several categories listed backlog criteria as high as 1 to 3 years old). The licensee explained that when the program was initially started in 1992, the backlog criteria was set high intentionally so as to identify those items which were the oldest, while keeping the number of backlog items at manageable levels (i.e., with these backlog criteria, the licensee engineering backlog, at the time, was greater than 700 open items). However, the team noted that by 1994 the licensee had significantly reduced their engineering backlog, but had failed to adjust the backlog criteria. The failure by the licensee to reduce the threshold of the backlog criteria was considered a weakness.

To better understand the work load on engineering personnel, the team questioned the number of open items presently assigned to the department. At the time of this inspection, there were approximately 1508 t[illegible] open items. To determine the impact of the open items and to assess the safety significance of items still open, the team reviewed a number of the open items listed (plant improvement requests, corrective work requests, licensee event reports, etc.). The team determined that the open items had been appropriately categorized and given the appropriate prioritization for correction and closeout.

A number of closed items were also reviewed. Licensee actions in closing these items were considered to be appropriate.

Finally, the team interviewed members of the engineering staff with regard to work backlog. Open items were tracked by engineering supervisors at the group level. Engineers appropriately scheduled and worked on open items according to their prioritization and procedural requirements.

The licensee indicated that according to their records, the overall number of open items that are tracked has been generally on the decline. However, performance improvement requests were the only open item group that had showed a steady increase. The licensee attributed this to a lower threshold for issuance of these reports and a heightened awareness by plant personnel due to increased training in this area.

c. Conclusions

The licensee managed the engineering open item workload appropriately, but the licensee backlog program was found to be behind the industry standard due to the lack of a formalized program, high threshold for backlog criteria, and the failure to trend the impact of the backlog on engineering personnel workload.

E2.7 Surveillance Testing

a. Inspection Scope

The inspector reviewed Technical Specification surveillance requirements for the three systems selected and the most recently completed surveillance tests for each of these surveillance requirements.

b. Observations and Findings

The surveillance for the systems selected accomplished the Technical Specification surveillance requirements and were performed at the correct periodicity. Exceptions are noted below:

Improper Verification of Emergency Core Cooling System Throttle Valve Mechanical Stop Position

Technical Specification Surveillance Requirement 4.5.2.g required the licensee to verify the correct position of each mechanical position stop for the listed emergency core cooling system valves every 18 months. This verification ensures that sufficient cooling flow is available for post-accident conditions. The licensee accomplished this surveillance requirement by performing Procedures STS EM-001, "Emergency Core Cooling System Throttle Valve Verification," Revision 11, and STS BG-004, "Chemical and Volume Control System Seal Injection and Return Flow Balance," Revision 4. These procedures required workers to measure the valve stem height for the valves specified in the Technical Specification.

The team asked how the surveillance procedures verified the position of the mechanical position stops. The 12 EM (Safety Injection) system valves listed in Technical Specification 4.5.2.g, and Valve BGV-202, did not have mechanical position stops, but were locked in place using a locked chain as specified in Procedure AP 21G-001, "Control of Locked Component Status," Revision 7. Seal injection valves BGV-198, BGV-199, BGV-200, and BGV-201 had valve stem locknuts to secure the valve in position, but they were not required to be tightened or verified during performance of the surveillance test. In addition, the team noted that the procedure contained a drawing of the valve which did not indicate the presence of a locking nut.

The team considered the surveillance procedure to be deficient for not including the specific design attributes of the mechanical stops and specific action necessary to verify the correct position of the stops. In response to this concern, the licensee checked the locknuts, and found them tight. The team interviewed two non-licensed operators who had recently performed this surveillance procedure, and found that the operators could not recall whether they tightened the locknuts during this surveillance, or not. The system engineer also interviewed another non-licensed operator who had recently performed this surveillance and also found that the operator could not recall tightening the locknuts. The failure of Procedure STS BG-004 to require the test performer to tighten the locknuts for these valves is a violation of Technical Specification 6.8.1.a (50-482/96021-06).

Improper Essential Service Water Underground Piping Pressure Test

The team reviewed Performance Improvement Request 95-2326, which was initiated on September 20, 1995, to request a change in the test method for essential service water system underground piping pressure tests. The description of the problem stated that past performances of Test Procedure STS PE-049C, "Essential Service Water System Underground Piping Leakage Test," Revision 1, had proven to be very cumbersome and manpower intensive. This test was written to satisfy the requirements of ASME Section XI as implemented by the licensee's inservice inspection program for this Code Class 3 system. The test method being used included the installation of blank flanges, isolating the system, and determination of the rate of pressure loss. Because this portion of pipe is buried underground, the initiator requested that the optional testing requirements in Article WA-5244 of the ASME Code be considered for alternative testing of buried components. Article IWA-5244 contains three options that are based on system redundancy and piping isolation abilities:

(a) In non-redundant systems where the buried components are isolable by means of valves, the visual examination VT-2 shall consist of a leakage test that determines the rate of pressure loss. Alternatively, the test may determine the change in flow between the ends of the buried components. The acceptable rate of pressure loss or flow shall be established by the owner.

(b) In redundant systems where the buried components are nonisolable, the visual examination VT-2 shall consist of a test that determines the change in flow between the ends of buried components. In cases where an annulus surrounds the buried components, the areas at each end of the buried components shall be visually examined for evidence of leakage in lieu of a flow test.

(c) In non-redundant systems where the buried components are nonisolable, such as return lines to the heat sink, the visual examination VT-2 shall consist only of a verification that the flow during operation is non impaired.

In the evaluation for this request, the engineer concluded that each of the two trains of essential service water could be considered a non-redundant system. This interpretation determined that each train provided cooling water only to the loads associated with that train (i.e., Train A of essential service water supplies cooling water to Train A heat loads, and Train B of essential service water supplies cooling water to Train B heat loads, with no other cooling water supply to the separate trains). This interpretation was not based on an ASME Code definition or an official ASME Interpretation.

As a result of the evaluation, the engineer further concluded that paragraph (c) of Article IWA-5244, could be applied to the buried portions of the essential service water system. This conclusion resulted in revisions to Test Procedure STS PE-049C, "A Train Underground Essential Service Water System Piping Flow Test," and development of new Test Procedure STS PE-049D, "B Train B Underground Essential Service Water System Piping Flow Test," which eliminated the previous method of performing the visual examination VT-2 (i.e., determination of the rate of pressure loss) and implemented visual examination VT-2 that consisted only of a verification that the flow during operation is not impaired.

Section 9.2.1.2, "Essential Service Water System," of the Updated Safety Analysis Report states that the essential service water system consists of two redundant cooling water trains. The team considered the licensee's interpretation of system non-redundancy to contradict this statement in the licensing basis.

The 10 CFR 50.59 screening for the test procedure change indicated that Chapter 9.2 of the Updated Safety Analysis Report was reviewed. The screening did not discuss the discrepancy regarding redundant versus nonredundant definitions for the essential service water system trains. The licensee did not submit a request for NRC review and approval of the alternative test method. Neither did the licensee revise Chapter 9.2 of the Updated Safety Analysis Report to indicate that the essential service water system trains could be considered nonredundant systems. Therefore, the team considered that the screening for the proposed change to the underground piping test procedures was deficient for not

identifying that a change to the Updated Safety Analysis Report or inservice inspection program (Technical Specification 4.0.5) was involved. This deficiency is contrary to the requirements of 10 CFR 50.59 and is considered to be the fourth example of the apparent violation (50-482/96026-02).

The revised Procedure STS PE-049C was used for the system pressure test performed for the third 40-month period in the first 120-month interval. The test was completed on January 17, 1996. Likewise, Procedure STS PE-049D was performed during January 1996. Performance of the revised tests resulted in the failure to comply with the requirements of Section XI of the ASME Code for buried piping in redundant systems and non-compliance with Technical Specification 4.0.5.

During the exit meeting, the licensee disagreed with the team's conclusion that this matter was a violation. The licensee stated that since neither the AMSE Code nor the Technical Specifications defined the term "redunuant"; therefore, it was appropriate for them to do so. The licensee's inservice inspection engineer had attended industry working group committee meetings, which discussed pressure testing and the definition of redundant and non-redundant systems. The licensee referred to the 1995 Addenda to the 1995 Edition of Section XI of the ASME Code, Article IWA-5244, which had been changed to differentiate test methods based only on whether the piping is isolable or non-isolable, and removed references to redundant or nonredundant. The inservice inspection engineer utilized this knowledge when interpreting these requirements for underground piping pressure testing. In addition, the onsite Authorized Nuclear Inservice Inspector had reviewed the change to the test procedure and had no comment. However, the inspection team noted that the NRC has not yet endorsed the 1995 Addenda and the Authorized Nuclear Inservice Inspector has no responsibility under 10 CFR 50.59.

c. Conclusions

In general, the team found that the surveillance for the systems selected accomplished the Technical Specification surveillance requirements and were performed at the correct periodicity. However, the team identified one violation associated with an inadequate procedure to verify emergency core cooling throttle valve mechanical position stops, and an example of an apparent violation regarding pressure testing of essential service water system underground piping.

E2.8 Industry Event Assessment and Lessons Learned

a. Inspection Scope

The team reviewed two industry events to determine the licensee's action to prevent similar problems. Industry documented failures of 4.16 kV General Electric Magne-Blast circuit breakers to properly close, and of improper refurbishment of 4.16 kV breakers by overhaul vendors, were selected for review due to generic applicability to the plant.

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b. Observations and Findings

The team found that the licensee had received reports of these events and had taken corrective actions to prevent occurrence of these problems at Wolf Creek. Preventive maintenance procedures and procurement documentation had been reviewed by licensee personnel and appropriate revisions made to identify and correct similar problems.

E3 Engineering Procedures and Documentation

E3.1 Review of Design Basis Documents

a. Inspection Scope

The team reviewed the design basis documents for the essential service water system, the residual heat removal system, and the safety injection system to verify the validity of the design basis and determine the ease of retrieving the information.

b. Observations and Findings

The team reviewed the design basis notebook for the essential service water system and determined that the notebook had beer approved in May 1993 and had not been updated since then. The team noted a statement in the notebook that when the notebook was to be used for design input, the user should take into account the changes issued after the approval date of the notebook. At the time of the notebook approval, the notebook had been a controlled document.

The team reviewed Interoffice Correspondence ED 96-0047, dated September 17, 1996, concerning design basis notebooks. The letter stated that due to downsizing of engineering and the need to reorganize the work effort, design engineering had identified that the notebooks were an opportunity to reduce the demand on engineering services. Some of the licensee's actions were to keep the notebook for information only and not maintain it as a controlled document. In addition, the licensee decided that the system description documents would be used to keep design basis information in the future. The licensee further stated that the extent of information added to the system description would vary depending on the judgement of the responsible engineer. The design engineering manager stated that there was no need for the notebooks since all of the engineers were very experienced and knew where to find the design basis information.

Since the design basis notebooks provided information to support the design and licensing basis and provided the location of other design bases documents, the team considered that uncontrolled and out-dated notebooks hindered the control of design basis information. This conclusion was supported by the fact that no other controlled document provided this information. The team also noted during the inspection, there were times when the licensee had difficulty retrieving design basis information. The team considered the licensee's control of design basis information to be weak for not providing a central location for the design basis information.

c. Conclusions

Uncontrolled and out-of-date design basis notebooks hindered the control of design basis information. The licensee's control of design basis information was found to be weak, in that, it did not provide a central location for the design basis information. Licensee personnel had difficulty retrieving some design basis information.

E5 Engineering Staff Training and Qualification

E5.1 System Engineering Staff Training and Qualification (37550)

a. Inspection Scope

A review was performed of the system engineering training program. The team reviewed Administrative Procedures AP-23-006, "System Engineering Program, " Revision 3, and AP 30F-001, "Engineering Support Personnel Training and Qualification Program," Revision 2. The team discussed the training requirements with a number of system engineers, and members of their direct management, during individual interviews. In addition, the team reviewed the training records for all of the system engineers.

b. Observations and Findings

The team found the guidance for system engineering training and management expectations provided in the licensee's administrative procedures to be general in nature. Training requirements for engineers newly assigned to the system engineering department, were developed by the engineer's immediate supervisor, and were found to consist of "Qualifying Activities," which included "Evaluation of Nonconforming Conditions of Installed Plant Equipment," (i.e. operability determinations) "Engineering Calculations," "Unreviewed Safety Question Determination," etc. Specific training on assigned systems was not required and was left to each engineer's discretion to take system-specific courses that periodically were offered for operations personnel. With regard to those situations in which system engineers were assigned to a specific system, but were later given responsibility for another system, the team noted that little guidance on training was available other than for "Qualifying Activities." Finally, none of the procedures were found to specify a time period for completion of training requirements nor were there any minimum criteria for system engineer acceptance. In response to this concern, the system engineering management issued a performance improvement request.

In spite of the overall general guidance, the team found that the system engineer's knowledge of each of their assigned systems was excellent. This was due, in part, to a significant number of engineers having been involved in operator systems training prior to entering the system engineering program. In addition, the system engineers and their immediate supervisors displayed excellent initiative to improve their knowledge.

For example, the system engineers interviewed were knowledgeable of industry problems and maintained periodic contact with other utilities and equipment vendors. The system engineers also periodically walked down their systems in accordance with a system walkdown schedule that had been reviewed and approved by their immediate supervisors. The system engineering supervisors encouraged their personnel to attend technical presentations, classes, and meetings held by vendors or other utilities. One specific example of the initiative taken by the system engineering supervision involved the reactor coolant system engineer, who had been recently assigned to take responsibility for this system. His supervisor arranged a visit to the Callaway plant, which had an identical reactor coolant system and was in an outage. This afforded the system engineer an opportunity to walk down the reactor coolant system and become familiar with his system which he might not have been able to do at Wolf Creek until their next assigned refueling outage.

Finally, almost all system engineers were found to have completed the appropriate "Qualified Activities" training as indicated by their departments training records. Those cases where engineers had not completed their assigned training was due specifically to the fact that they had recently been assigned to their present position.

c. Conclusions

System engineering knowledge was found to be excellent and was based on the initiative taken by system engineers and their immediate supervisors, and not by any specific guidance provided in administrative procedures available. Training guidance was found to be too general. Specifically, it did not provide a minimum standard for system engineer training or knowledge.

E6 Engineering Organization and Administration

E6.1 System Engineering (37550)

a. Inspection Scope

The team interviewed the system engineering manager, three group supervisors, and seven system engineers. The team focussed on licensee management expectations of the system engineers and the system engineering program. This included the method in which these expectations were communicated to the system engineers, the mechanics of how plant problems were identified and corrected, and the adequacy of communication between the system engineering department and other plant organizations such as operations and maintenance. Additionally, the system engineers were questioned on technical information and outstanding deficiencies for their assigned systems, including actions they were taking to resolve those deficiencies.

b. Observations and Findings

The licensee management expectations of the system engineers and the system engineering program were delineated in licensee Administrative Procedure AP 23-006, "System Engineering Program," Revision 3, and Administrative Instruction AI 23-002, "System Engineering Plant Walkdowns," Revision 0. Licensee management also communicated their expectations verbally either directly or through the group supervisors.

As stated previously in this report (Section E5.1), the team found that the guidance provided in the administrative procedures and instructions were general in nature. More specific guidance was verbally provided to the system engineers, at the group level, by their appropriate supervisors.

The system engineers stated that although engineering management expectations were general in nature, they believed that the guidance being provided presently was an improvement over the lack of guidance that existed in 1995. This improvement was in part the result of Self Assessment Reports SEL 95-039, "System Engineering," dated January 19, 1996, and SEL 96-025, "System Engineering Self Assessment Effectiveness Follow-up," dated September 16, 1996. The system engineers indicated that with a clearer definition of their job scope, they have a better understanding as to what they are required to do and which type of activities they can defer to another organization. The team found that system engineers understood their management's expectation in which they would be the "experts" of their assigned systems and take "ownership" of their assigned responsibilities.

In accordance with the procedural guidance, system engineers also had developed primary trending parameters, and walkdown guidelines for their assigned systems, which were reviewed and approved by their group supervisors. However, the team noted that the consistency of how these two aspects of the system engineers workload were being performed was not closely monitored by engineering management. In addition, the system engineers used system notebooks in an inconsistent manner. Nonetheless, the system engineers knowledge of their individual systems was excellent. Operations and maintenance planning personnel considered the system engineers as the "experts" of their assigned systems and as the focal point for any questions on these systems. Operations personnel indicated that they had confidence in system engineering personnel to provide them the appropriate information to make operability determinations.

System engineers displayed "ownership" of their system by following maintenance activities being performed on their assigned systems. Plus, system engineers periodically reviewed corrective work requests to identify if any applied to their assigned system. As mentioned in Section E2.6, system engineers demonstrated this ownership during the system walkdowns with team members.

The team noted that the system engineering program did not specify the need for backup system engineers for the safety-related equipment. The licensee had an unofficial system engineer backup program, but it did not have any basic training criteria or knowledge expectations. In addition, some of the system engineers were unaware that they had been assigned as backup system engineers, and others were not aware that any backup system engineers had been assigned to their system. Finally, other plant personnel were unaware as to whom were the backup system engineers, and what systems they were responsible for. This is considered to be a weakness in the system engineering program, and behind industry standards.

c. Conclusions

Overall, the system engineers were found to be knowledgeable of management expectations and their responsibilities. Licensee management communication of system engineering expectations has improved. The lack of assigned backup system engineers was considered a program weakness.

E6.2 Design Engineering (37550)

a. Inspection Scope

The team conducted interviews with personnel from the maintenance planning, and operations departments to evaluate the extent and effectiveness of design engineering communications. The team also reviewed a number of change packages and performance improvement requests that required engineering involvement, in an effort to determine how technical issues were resolved.

b. Observations and Findings

The team identified that cooperation and communication among the design engineering department and operations, and maintenance planning departments were good. Engineers indicated that management encouraged identification of plant problems. This has contributed to the increase in the number of performance improvement requests.

The team found that the performance improvement requests and change packages reviewed had technical resolutions with proper engineering justifications and that the proposed corrective actions were adequate.

The team noted that engineers were appropriately utilizing available design basis documents to determine if a proposed change was within the original design basis. All personnel interviewed were aware that the design basis notebooks were not controlled documents, and they only used them as reference documents.

c. Conclusions

The team concluded that the licensee was effectively implementing their program to respond to requests for engineering resolution of plant problems.

E7 Quality Assurance in Engineering Activities

a. Inspection Scope (37550)

The team reviewed four recent quality assurance self assessment reports related to engineering activities. Self Assessment Report SEL 96-033, "Licensee Event Report Program," dated October 2, 1996, SEL 96-025, "System Engineering Self Assessment Effectiveness Follow-Up," dated September 9, 1996, and SEL 95-056, "Auxiliary Feedwater System," dated January 9, 1996, were reviewed to evaluate the effectiveness of the licensee's controls in identification and resolution of plant problems. Although not complete, the inspection team reviewed the assessment plan and preliminary findings for an auxiliary feedwater functional assessment.

b. Observations and Findings

The team found that the self assessments were broad in scope and provided meaningful findings and recommendations for potential program enhancements. As an example, the auxiliary feedwater system self assessment resulted in a number of improvement recommendations. These recommendations encompassed more than enhancements to system performance and reliability but system engineering program enhancements also. One such improvement recommendation included placing the site wide trending program in a centralized location (e.g., trending data is located in several groups and information exchanged is not formalized). Other recommendations included a review of spare parts availability. Although improvements since the previous self assessment (SEL 95-039) had occurred, the system engineering self assessment identified weaknesses in management and supervisory oversight of the system engineers. The self assessments resulted in the issuance of a number of performance improvement requests to address the weaknesses identified.

The team found that the auxiliary feedwater system functional assessment plan included similar items that the team was reviewing. In addition, some of the initial findings from this self assessment effort were similar to those identified in this report.

c. Conclusions

The team concluded that the licensee's self-assessment reports were effective.

E8 Miscellaneous Engineering Issues

E8.1 (Closed) Inspection Followup Item 50-482/9504-03: Use of gear operator stop nut for actuator braking.

The licensee contacted the valve operator manufacturer who reviewed the licensee's procedures for setting the stop nuts and limit switch settings and concurred with the licensee's actions. The load applied to the stop nuts was within rated design load.

E8.2 (Closed) Licensee Event Report 50-482/96001: Loss of circulating water due to icing on traveling screens.

This event was discussed in NRC Inspection Report 50-482/96-03 and was the subject of a violation as listed in NRC letter EA96-124, dated February 29, 1996, item 06014. No new issues were revealed by the licensee event report and followup on the licensee's corrective actions will be performed during the review of the violation.

E8.3 (Closed) Licensee Event Report 50-482/96002: Loss of essential service water train due to icing on trash racks.

This event was discussed in NRC Inspection Report 50-482/96-03 and was the subject of two violations as listed in NRC letter EA96-124, dated February 29, 1996, items 02013 and 04013. No new issues were revealed by the licensee event report and followup on the licensee's corrective actions will be performed during the review of the violations.

V. Management Meetings

X1 Exit Meeting Summary

The team presented the inspection results to members of licensee management at the conclusion of the inspection on October 25, 1996. An exit meeting was held via teleconference on November 8, 1996. The licensee acknowledged the findings presented. The overall scope and results of the inspection were discussed with Mr. Terry Damashek, on December 31, 1996.

The licensee did not identify that any propriety information was reviewed by the team.

ATTACHMENT

SUPPLEMENTAL INFORMATION

PARTIAL LIST OF PERSONS CONTACTED

Licensee

G. Boyer, Director, Site Support
T. Damashek, Supervisor, Regulatory Compliance
R. Flannigan, Manager, Nuclear Engineering
T. Garrett, Manager, Design Engineering
B. Grieves, Supervisor, Systems Engineering
T. Hood, Supervisor, Design Engineering
N. Hoodley, Manager, Support Engineering
R. Hubbard, Superintendent, Operations
O. Maynard, Chief Administrative Officer
B. McKinney, Plant Manager
T. Morrill, Manager, Regulatory Services
R. Muench, Vice President Engineering
G. Neises, Supervisor, Reactor Engineering
D. Neufeld, Acting Manager, Integrated Planning and Scheduling
W. Norton, Manager, Performance Improvement and Assessment
K. Scherrch, Supervisor, Systems Engineering
R. Sims, Manager, Systems Engineering
J. Stamm, Supervisor, Safety Analysis
C. Warren, Chief Operating Officer
C. Younie, Manager, Operations

NRC

S. Freeman, Resident Inspector

INSPECTION PROCEDURES USED

IP 37550	Engineering
IP 37001	10 CFR 50.59 Safety Evaluation Program
IP 92903	Followup - Engineering

ITEMS OPENED AND CLOSED

Opened

50-482/96021-01	VIO	Inadequate Control of Design Bases (Section E1.2)
50-482/96021-02	APV	Four Examples of the Failure to Properly Perform Safety Evaluations (Sections E2.2, E2.3, E2.3, and E2.7)
50-482/96021-03	APV	Failure to disable centrifugal charging pump while in cold shutdown (Section E2.3)
50-482/96021-04	APV	Inadequate Corrective Action for Screening Technical Specification Clarifications (Section E2.3)
50-482/96021-05	VIO	Unsupported Operability Determination for Containment Cooler Flow (Section E2.4)
50-482/96021-06	VIO	Inadequate Procedure for Verification of Emergency Core Cooling Throttle Valves Mechanical Position Stops (Section E2.7)

Closed

50-482/95004-03	IFI	Use of Gear Operator Stop Nut for Actuator Braking (Section E8.2)
50-482/96001	LER	Loss of Circulating Water due to Ice (Section E8.3)
50-482/96002	LER	Loss of Essential Service Water train due to Ice (Section E8.4)

LIST OF DOCUMENTS REVIEWED

Unreviewed Safety Question Determinations

Number	Title
59 93-0211	Main Steam Isolation Actuator Upgrade Modification, Revision 0
59 94-0174	Deletion of Reporting Requirements from Updated Safety Analysis Report for Seismic Monitors, Revision 0
59 95-0003	Reactor Coolant Pump Flywheel Inspection Clarification, Revision 0
59 95-0016	Spent Fuel Pool Surveillance Level Indicator, Revision 0
59 95-0034	Fire Area Combustible Load Evaluation, Revision 0

59 95-0046	Optional Opening Between Room 1203 and Room 1204, Revision 0
59 95-0057	Minimum Acceptance Criteria for Centrifugal Charging Pump B, Revision 0
59 95-0061	Transient Cable Separation Criteria, Revision 0
59 95-0063	Biennial Relevancy Procedure Review Requirements, Revision 0
59 95-0109	Auxiliary Feedwater Pump Turbine Exhaust Line Upgrade, Revision 0
59 95-0129	Emergency Diesel Generator Design Explanation, Revision 0
59 95-0150	Auxiliary Feedwater Flowrate Revision, Revision 0
59 95-0151	Emergency Core Cooling System Flowrate Revision, Revision 0
59 95-0156	Boron Injection Tank Recirculation Pump Removal and Removal of Thermal Relief Valve, Revision 0
59 96-0032	Operation with Polypropylene Filter Membrane Material in Spent Fuel Pool, Revision 0
59 96-0034	Delete Reporting Requirements for Meteorological Tower Instrumentation, Revision 0
59 96-0038	Use of Safety Injection Pump for Boration in Mode 6, Revision 0
59 96-0086	Downgrade of Reactor Coolant Pump #1 Seal Leak Off Pressure Indicator, Revision 0
59 96-0109	Highpressure Feedwater Heater Bypass Test, Revision 0
59 96-0115	Delete Program Descriptions from Updated Safety Analysis Report, Revision 0
59 96-0143	Revise Updated Safety Analysis Report to Reflect use of Auxiliary Feedwater in Residual Heat Removal Process, Revision 0
59 96-0148	Revise Scaffolding Procedure, Revision 0
59 96-0155	Clarification of Regulatory Guide 1.144, Revision 0

Updated Safety Analysis Report Change Requests Associated With Technical Specification Amendments

Number	Title
Amendment 89	Updated Safety Analysis Report Change Request 95-137, dated 12/1/95, Borated Water Sources
Amendment 91	Updated Safety Analysis Report Change Request 95-138, dated 12/1/95, Refueling Water Storage Tank Boron Concentration
Amendment 93	Updated Safety Analysis Report Change Request 96-004, dated 1/11/96, Relocate Time Response Tables to Updated Safety Analysis Report
Amendment 94	Updated Safety Analysis Report Change Request 96-104, dated 9/17/96, Operation of Emergency Fuel Oil Transfer System

Updated Safety Analysis Report Change Requests

Number	Title
87-022	Corrections to Typographical Errors in Chapter 6, dated 7/15/87
96-031	Surveillance Frequencies for Main Dam, Saddle Dams, and Baffle Dikes, dated 2/16/96
96-094	Incorporate Technical Specification Interpretation, dated 8/29/96
96-095	Incorporate Technical Specification Interpretation, dated 8/29/96
96-096	Incorporate Technical Specification Interpretation, dated 8/30/96
96-104	Revise Emergency Diesel Generator Transfer Pump Logic, dated 9/17/96
96-118	Revise Spent Fuel Pool Rack Information, dated 9/26/96
91-047	Correction to Updated Safety Analysis Report Change Request 90-114, dated 7/10/91

Regulatory Screenings

Number	Title
05622	Revision 0, Motor Operated Valve
05720	Revision 0 and Revision 1, Pressure Locking Modification
05782	Revision 2, Turbine Driven Auxiliary Feedwater Pump Resistor Modifications
05846	Revision 0, NK Battery Replacement
05900	Revision 0, Pressure Locking/Thermal Binding Evaluation
05906	Revision 0, Centrifugal Charging Pump High Temperature Alarm
05927	Revision 0, Low Flow Cavitation Limit Exceeded
06023	Revision 0, Pacific Valve Configuration Change
06025	Revision 0, Drain Holes in Code Relief Valves
06107	Revision 0, Relief From American National Standards Institute Code Hydrostatic Test Requirements
06183	Revision 0, Delete Thermal Relief Valves from Component Cooling Water System
06189	Revision 0, Battery Charger Alarm Setpoint
06252	Revision 0, Turbine Driven Auxiliary Feedwater Pump Valve Stem Replacement
06285	Revision 0, Revised Thermal Design Flow
06304	Revision 0, Load Drop for New Fuel Storage Facility
06394	Revision 0, Safety Injection Pump Rework
06445	Revision 0, New Safety Injection Pump A and Rotating Element B Approval
2121	Revision 5, Flow Element EM FE0928 ALARA Concern
3749	Revision 1, SMB-00 Torque Switch Improvement
4055	Revision 4, Valve EM HV8807A & B Speed Reduction
4139	Revision 6, Motor-operated Valve - Concerns with EM HV8814A/B

4145	Revision 4, Adjust Torque Switch Settings on EM HV8924
4148	Revision 7, Motor-operated Valve - Disposition for EM HV8801A/B & EM HV9903A/B
4150	Revision 5, Valve EM HV8835 Motor-operated Valve Disposition
4385	Revision 2, Main Control Board Switch Engraving Discrepancies
4394	Revision 13, Target Rock Valves Replacement
4537	Revision 4, Boron Injection Tank Recirculation Removal
6424	Revision 1, Fabrication of Thrust Collar Spacer for PEM01A
6457	Revision 0, Safety Injection Pump Motors PEM01B Bolts Modification

Industry Technical Information Program Reports

Number	Title
02102	Liberty Technologies, 10-2-92: 10 Code of Federal Regulations Part 21 Notification, Stem Material Constants And Torque Calibrator Effects Impact Votes Testing Accuracy, Potential For Overthrust
02340	NRC Information Notice 93-37: Eyebolts With Indeterminate Properties Installed in Limitorque Valve Operator Housing Covers
02371	Limitorque Maintenance Update 92-02: Motor Pinion Keys, Motor Performance, Declutch Tips, Torque Switch Repeatability, Actuator Nameplate, Actuator Wiring
02372	Limitorque Maintenance Update 92-02: Motor Pinion Keys, Motor Performance, Declutch Tips, Torque Switch Repeatability, Actuator Nameplate, Actuator Wiring
02373	Limitorque Maintenance Update 92-02: Motor Pinion Keys, Motor Performance, Declutch Tips, Torque Switch Repeatability, Actuator Nameplate, Actuator Wiring

Calculations

Number	Title
C-1989-130	Seismic Reanalysis of Refueling Water Storage Tanks, Revision 2
EF-M-014	Ultimate Heat Sink Thermal Analysis Review for Power Uprate, Revision 1
EF-M-029	Minimum Essential Service Water Temperature Rise, Revision 1
EF-M-030	Determine Required Essential Service Water Warming Line Flow, Revision 0
EF-M-031	Determine Orifice Sizes for Ultimate Heat Sink Outlet, Warming Line Outlets, and FE-3&4 Necessary to Ensure 5000 GPM Essential Service Water Warming Line Flow and the Corresponding Maximum Pressure Downstream of FE-3&4, Revision 0
EF-M-032	Determine Hydraulic Grade Line Elevation Required at the Essential Service Water Warming Line Branch, Revision 0
EF-M-033	Evaluate if 1" Thick Plate is Acceptable for EF-FE-03 & EF-FE-04, Revision 0
EF-M-034	Investigate Design for Ultimate Heat Sink Discharge Orifice Plate on Essential Service Water System, Revision 0
EF-M-035	Investigate Design for Warming Line Discharge Orifice Plate on Essential Service Water System, Revision 0
EF-M-036	Determination of Maximum Lake Temperature for Operation with Warming Flow, Revision 0
EF-M-037	Summary of Document Control Procedure 06349 M-11EF01 Flow Diagram Changes, Revision 0
ECCS-5	Centrifugal Charging Pump "A" Net Positive Suction Head Determination During Cold Leg Recirculation, Revision 0
ECCS-6	Centrifugal Charging Pump "B" Net Positive Suction Head Determination During Cold Leg Recirculation, Revision 0
ECCS-7	Centrifugal Charging Pump "A" Net Positive Suction Head Determination During Hot Leg Recirculation from Residual Heat Removal Sump "A," Revision 0

ECCS-8	Centrifugal Charging Pump "B" Net Positive Suction Head Determination During Hot Leg Recirculation from Residual Heat Removal Sump "A," Revision 0
ECCS-9	Refueling Water Storage Tank to Safety Injection Pump A - Criteria Calc. (1 - 10), Revision 0
ECCS-10	Residual Heat Removal Sump A to Safety Injection Pump A Suction - Mode F, Revision 0
ECCS-11	Residual Heat Removal Sump A to Safety Injection Pump B Suction - Mode F, Revision 0
ECCS-17	Maximum Head Loss from Refueling Water Storage Tank to Either Centrifugal Charging Pump During Injection Phase of SIS, Revision 0
ECCS-32	Containment Sump "B" to Safety Injection Pump "B" Inlet, Mode E, Revision 0
ECCS-36	Refueling Water Storage Tank to Safety Injection Pump "B" Suction Mode A, Revision 0
ECCS-47	Safety Injection Pumps Net Positive Suction Head from Refueling Water Storage Tank, Revision 0
EF-35	ESW Pump Head Requirement, Revision 2
EJ-29	Residual Heat Removal - Flow Orifice Sizing, Revision 0
EJ-30	Residual Heat Removal Pumps A&B Net Positive Suction Head, Revision 1
EJ-35	Residual Heat Remova Pump Minimum Flow Recirculation Line Orifice Sizing, Revision 0
EJ-37	Residual Heat Removal Cold and Hot Leg Recirculation Orifices, Revision 0
EJ-38	Containment Recirculation Sump Screen, Revision 0
EJ-40	Containment Recirculation Sump Screen Fluid Velocity, Revision 0
EJ-M-001	Verification of Relief Valve Capacity for Valves EJ8708A&B, Revision 0
EJ-M-017	Potential Susceptibility for Pressure Locking of Motor-operated Valves EJHV8819A&B, Revision 2

EJ-M-019	Sizing of Expansion Pipe for Valves EJHV8811A&B for Pressure Locking Concerns, Revision 1
EJ-MH-001	Heat Transfer for the Evaluation of Thermal Binding and/or Pressure Locking of Valves EJ-HV8716A&B, Revision 0
EJ-S-003	Min. Wall Thickness Evaluation, Revision 1
1-HBC-W	Essential Service Water Discharge Piping Design Pressure and Minimum Wall Thickness Determination, Revision 1
IMS-01	Missiles, Revision 0
PB-01	Total Pipe Break Summary, Revision 1
BN-20	Refueling Water Storage Tank Level Set-Points, Revision 1

Modifications

Number	Title
03377	Seismic Reanalysis of Refuel Water Tank, Revision 0
03838	EF/EA Cross Tie Piping Modification, Revision 0

Temporary Modification Order

Number	Title
96-018-EJ	Installation of Pressure Gauge Downstream of Valve HV8840
96-024-BB	Eliminate Nuisance Alarm of annunciator D074, Revision 2
96-038-FP	Replace Plant Diesel Fire Pump with Temporary Pump While Fire Pump is Repaired, Revision 1
96-040-SE	Eliminate inadvertent alarm of Control Room Annunciators 82B and 83C, Revision 0
96-020-AB	Install Temperature Monitoring Equipment on the Main Steam Isolation Valve Accumulators, Revision 0
96-021-BB	Protect Vessel Head Seismic Support Plate from Excessive Leakage from the Vessel Head Vent Valves, Revision 0

Self Assessment Reports

Number	Title
95-056	Auxiliary Feedwater System
95-039	System Engineering Self Assessment
96-025	System Engineering Self Assessment Effectiveness Follow-Up
96-033	Licensee Event Report Program

Drawings

Number	Title
M-12BB01	P&ID Reactor Coolant System, Revision 15
M-12BG03	P&ID Chemical & Volume Control System, Revision 16
M-12BN01	P&ID Borated Refueling Water Storage System, Revision 08
M-12EJ01	P&ID Residual Heat Removal System, Revision 15
M-12EM01	P&ID High Pressure Coolant Injection System, Revision 16
M-12EM02	P&ID High Pressure Coolant Injection System, Revision 09
M-12EM03	P&ID High Pressure Coolant Injection System Test Line, Revision 00

Reportability Evaluation Request Form

Number	Title
96-035	Mechanical Position Stops on BG Valves, dated October 23, 1996

Procedures

Number	Title
28D-001	Self Assessment Process, Revision 2
05-004	Specifications, Revision 1
05-003	Design Document Change Notice, Revision 1

05C-002	Engineering Evaluation Requests, Revision 0
05-002	Dispositions and Change Packages, Revision 2
05-001	Change Package Planning and Implementation, Revision 2
21I-001	Temporary Modifications, Revision 1
AP23L-001	Lake Water Systems Corrosion and Fouling Mitigation Programs, Revision 0
SYS EF-205	ESW/Circ Water Cold Weather Operations, Revision 1
STS EF-100A	ESW System Inservice Pump A and ESW A/Service Water Cross Connect Valve Test, Revision 17
STS EF-100B	ESW System Inservice Pump B and ESW B/Service Water Cross Connect Valve Test, Revision 18
STS EF-001	Essential Service Water Valve Check, Revision 7
STS IC-917	Analog Channel Operation Test Essential Service Water To Air Compressor Isolation, Revision 5
STS IC-602A	Slave Relay Test K602 Train A Safety Injection, Revision 8
STS IC-603A	Slave Relay Test K603 Train A Safety Injection, Revision 14
STS IC-608A	Slave Relay Test K608 Train A Safety Injection, Revision 11
STS IC-609A	Slave Relay Test K609 Train A Safety Injection, Revision 10
STS IC-927	ESW to Air Compressor High DP Isolation, Revision 3
STS IC-918	Channel Calibration Essential Service Water to Air Compressor Isolation, Revision 4
STS AL-005	Auxiliary feedwater Auto Pump Start and Valve Actuation, Revision 11
STS KJ-001B	Integrated D/G and Safeguards Actuation Test Train B, Revision 14
AP 14A-003	Scaffold Construction and Use, Revision 3
AP 21G-001	Control of Locked Component Status, Revision 7
STS BG-004	Chemical and Volume Control System Seal Injection and Return Flow Balance, Revision 5

STS EM-001	ECCS Throttle Valve Verification, Revision 11
MGE LT-012	SMB 000 Removal/Replacement, Revision 1
EMG ES-12	Transfer to Cold Leg Recirculation, Revision 7
AP 02-002	Chemistry Surveillance Program, Revision 2
STS EM-003B	ECCS (Safety Injected Pump) Flow Balance, Revision 0
STS EM-003A	ECCS (Centrifugal Charging Pump) Flow Balance, Revision 0
STS CR-001	Shift Logs for Modes 1, 2, & 3, Revision 33
STS BG-002	ECCS Valve Check and System Vent, Revision 8
STS EM-003	ECCS Flow Balance, Revision 8
STS IC-902A	Actuation Logic Test Train A Residual Heat Removal Suction Isolation Valves, Revision 0
STS IC-902B	Actuation Logic Test Train B Residual Heat Removal, Revision 0
STS KJ-001A	Integrated D/G And Safeguards Actuation Test - Train A, Revision 14
STS KJ-001B	Integrated D/G And Safeguards Actuation Test - Train B, Revision 14
STS IC-740A	Residual Heat Removal Switchover to Recirculation Sump Test - Train A, Revision 9
STS IC-740B	Residual Heat Removal Switchover to Recirculation Sump Test - Train B, Revision 9

Work Requests and Work Packages

Number	Title
110110	Motor-operated valve motor insulation found designated incorrectly
104812	Residual Heat Removal Pump Mechanical Seal Leakage
104898	Replacement of Relief Valve EJ8856A
106028	Residual Heat Removal Heat Exchange A Shell to Waterbox Bolting Torque Verification
107013	Valve EJV0053 Needs Lubrication of Stem

107292	Screens require refurbishment due to corrosion
108111	Essential service water pump motor oil level low
109892	Running of Residual Heat Removal Pumps Below 1700 gpm for Extended Periods of Time
109954	Inspect Pump Internals Due to Material Found in Valve ME8956C
110193	Wall thickness due to corrosion
110524	Installation of Temporary Gauge @ EJV0063 Downstream of HV8840
110622	Valve EJHCV0606 Leaks By (open)
110955	Essential service water pump operation below flow limits
110959	Residual Heat Removal Pump A Run at Flow Rates Below 1700 gpm
113208	Essential service water pump casing line leaking
113614	Leaking valve
113731	Valve EJ HCV-8890B Will Not Open
114876	Verify Shell to Waterbox Bolting Torque for EEJ01A (open)
115491	Check Valve EJ8730B Not Fully Seating (open)
108477	Essential service water pump prelube tank level indicator failed
109280	Cross tie valve failed leakage test
111729	Replacement of handle on essential service water tank screen
110136	Valve actuator shaft sheared off

Performance Improvement Requests

Number	Title
96-1488	Drawing change not properly removed from document control file
96-0634	Limit switch rotors not set correctly
96-0500	Drawing not added to vendor manual

96-1617	Questions related to essential service water icing event
96-1542	Non safety-related sealant used
96-1288	Confusion in throttle valve position
96-0659	Multiple failures of actuator shear pins
96-0365	Level indicator problems
96-1684	Inservice Testing stroke time failure
96-1214	Valve exceeded maximum alert stroke time
96-1741	Incorrect stroke time in procedure
96-1836	Corroded bolt holes on essential service water tank basket
96-1395	Difficulties encountered with controlotron operation
96-0737	Severe corrosion on essential service water piping and valves
96-0579	Severe corrosion on essential service water strainer backwash piping
96-2502	Valve failed stroke time test
96-1953	Fuse blocks found swapped
96-1902	Procedure conflict with updated safety analysis report
96-2675	USAR Statement on ECCS Water Hammer
96-2729	Missing Internal Missiles Design Basis Calculation Reference
96-2733	Questionable Use of a Pipe Whip Assumption in a Design Basis Calculation
95-0428	Industry event evaluation regarding SI Pump Runout Potential
96-2710	Mechanical Position Stops on BG Valves
94-0427	Low Flow Cavitation Limit Exceeded
94-0092	Limitorque Maintenance Update 92-02
94-0090	Limitorque Maintenance Update 92-02
94-0089	Limitorque Maintenance Update 92-02

95-0910	CCW Return Thermal Relief Valve Not Reseating
95-2901	Plant Modification Prepared Without Referring to Interim Drawing Changes
96-1014	Excessive Valve Local Leak Rates
94-0825	Potential for Inadvertent Safety Injection Actuation During Surveillance Testing
96-0308	Generic Letter 96-01
95-0625	Mitigation and Evaluation of Pressurizer Thermal Transients Caused by Insurges and Outsurges
95-0336	Lifting of Residual Heat Removal Relief Valves EJ8856A, B & EJ8842
96-0384	Thermal Binding Issue w/ Regard to Motor-operated valve EJ HCV8840

ENCLOSURE 5

COPY OF THE LICENSEE'S PLANT MODIFICATION REQUEST PMR 00903 PRESENTED DURING PREDECISIONAL ENFORCEMENT CONFERENCE EA 96-470, JANUARY 16, 1997

KPN-[illegible] REV. 2/85

71.1A

585073113

PLANT MODIFICATION REQUEST

PMR NO. 00903

CATEGORY 1A [X] 1B [ ] 1C [ ] [ ] 2 [ ] 3

PAGE 1 OF 11 REV 0

REF DOC KNPL8 8[illegible]

SECTION I - INITIATION

PMR TITLE Setpoints for safety related instruments

REASON FOR CHANGE Setpoints requested per Referenced document.

SUGGESTED MODIFICATION Issue safety-related set points on a design documents as requested by referenced document.

A. CLASS

SAFETY [X]
SPECIAL SCOPE [ ]
NONSAFETY [ ]

B. STANDARD PLANT [X] YES [ ] NO

C. COMPLETION DATE 5/2/85

D. COST CODE NUMBER 327.400

E. ALARA REVIEW REQUIRED [ ] YES [X] NO

ORIGINATOR R. Kevin Steinbrock DATE 5/2/85 TELEPHONE 1797

MGR/SUPT [signature] DATE 5-2-85 BRANCH/DIV NPE

SECTION II - CATEGORY 2 OR 3 APPROVAL

PLANT MGR N/A DATE [ ] APPROVED [ ] DISAPPROVED

SECTION III - CATEGORY 2 OR 3 SCOPE APPROVAL

PLANT MGR N/A DATE DNO DATE

INSERVICE DATE -

DANIEL CONSTRUCTION CO.
DOCUMENT CONTROL
MAY 02 1985
TIME

SECTION IV - ENGINEERING APPROVAL

RESP ENGR R. Kevin Steinbrock DATE 5/2/85

SEC MGR [signature] DATE 5/2/85

MGR NPE [signature] DATE 5-2-85

DOCUMENT CONTROL RELEASE FOR IMPLEMENTATION

REV 0

DATE DC38 5-2-85

SECTION V - VERIFICATION The PMR and WP are complete.

WCGS RESULTS ENGR DATE WCGS SUPV DATE

SECTION VI - CLOSEOUT Component data and documentation have been updated.

CM ENGR DATE

FORM KGE-12 REV. 2/85

PLANT MODIFICATION REQUEST

DOCUMENTATION SHORT FORM

PMR NO. 00903

PAGE 2 REV 0

MODIFICATION DOCUMENTS

AFFECTED DOCUMENT NUMBER	SHEET NO.	REV	CHANGE DOCUMENT NUMBER	DOC TYPE	PMR REV
J-121000A (Q)		0	CS-163-W-J-121000A (Q)	PMPD	0
J-121000B (Q)		0	CS-163-W-J-121000B (Q)	PMPD	0

AFFECTED COMPONENTS

COMPONENT NUMBER	COMP TYPE	DISP	MATERIAL CODE FOR SPARES	DATE CMO COMPLETED	PMP REV
See attached sheets					

OTHER AFFECTED DOCUMENTS

AFFECTED DOCUMENT NUMBER	SHEET NO.	REV	CHANGE DOCUMENT NUMBER	DOC TYPE	PMR REV
N/A					

REFERENCE DOCUMENTS/INFORMATION

DOCUMENT NUMBER	REV	DESCRIPTION
KNPLB 84-118		Kansas nuclear project letter to Bechtel (3 pgs) attached
CS-163-W	0	Plant Modification package (2 pgs) attach.

INFORMATION:

7901170317

FORM RGF-7 REV. 1/85

PLANT MODIFICATION REQUEST

AFFECTED COMPONENTS

PMR NO. 00903

PAGE 3 REV 0

COMPONENT NUMBER	COMP TYPE	DISP	MATERIAL CODE FOR SPARES	DATE CMO COMPLETED	PMR REV
EF-PDS-20		MOD			
EF-PDS-19		MOD			0
EF-PDSH-43		MOD			0
EF-PDSH-44		MOD			0
EG-LSLL-1		MOD			0
EG-LSLL-2		MOD			0
EG-FSH-62		MOD			0
EG-PSL-77		MOD			0
EG-PSL-78		MOC			0
EG-FSH-107		MOD			0
EG-FSH-108		MOD			0
EN-LSLL-15		MOD			0
EN-LSLL-16		MOD			0
FC-PSL-25		MOD			0
FC-PSL-26		MOD			0
FC-PSL-125		MOD			0
FC-PSL-126		MOD			0
GD-TSL-1		MOD			0
GD-TSL-11		MOD			0
GG-RSH-27		MOD			0
GG-RSH-28		MOD			0

FORM KGP-7 REV. 1/85

PLANT MODIFICATION REQUEST

AFFECTED COMPONENTS

PMR NO. 00403

PAGE 4 REV 0

COMPONENT NUMBER	COMP TYPE	DISP	MATERIAL CODE FOR SPARES	DATE CMO COMPLETED	PMR REV
GK-ASH-2		MOD			0
GK-ASH-3		MOD			0
GK-RSH-4		MOD			0
GK-RSH-5		MOD			0
GM-TSL-1		MOD			0
GM-TSL-11		MOD			0
GT-RSH-22		MOD			0
GT-RSH-33		MOD			0
GT-RSH-31		MOD			0
GT-RSH-32		MOD			0
JE-LSL-1A		MOD			0
JE-LSL-21A		MOD			0
JE-LSL-21C		MOD			0
JE-LSH-1B		MOD			0
JE-LSH-21B		MOD			0

Pink 903

Pg 5 of 9 RKS
11

REDA No. N-L-808-XX Rev. 0
Initiating Document No. KNLR84-118 Rev. -
PMP No. CS-163-W Rev. 0

CRITERIA AND INSTRUCTIONS FOR
PERFORMING 10CFR50.59
SAFETY EVALUATIONS

(LICENSING REVIEW SUPPLEMENT)
(SHEET 1 of 3)

INSTRUCTIONS:

Complete parts I, II, III, V, and VI for all design changes described in the primary document. Complete Part IV and the UNREVIEWED SAFETY QUESTION DETERMINATION on the SAFETY REVIEW RECORD, if one or more items in Part I are answered YES. Reviews under parts III and IV are considered to constitute the SAFETY EVALUATION required by 10CFR50.59. NRC approval is required prior to making the change or conducting the test or experiment if a change to NUREG 1104 is required or an UNREVIEWED SAFETY QUESTION exists.

I. 10CFR50.59 APPLICABILITY DETERMINATION

A. ✓ YES ___ NO — Does the change described in the primary document involve making changes in the facility as described in the safety analysis report?

B. ___ YES ✓ NO — Does the change described in the primary document involve making changes in the procedures as described in the safety analysis report?

C. ___ YES ✓ NO — Does the change described in the primary document involve conducting tests or experiments not described in the safety analysis report?

II. FSAR CHANGE DETERMINATION

A. ___ YES ✓ NO — Does the change described in the primary document involve a change to the FSAR?

B. If YES, identify the FSAR material subject to change:

SECTION(S) PAGE(S) TABLE(S) FIGURE(S)

C. ___ YES ___ NO ✓ N/A Proposed FSAR material changes are attached.

PMR 403

Pg 6 of Pages 11

(LICENSING REVIEW SUPPLEMENT)
(SHEET 2 of 3)

III. NUREG 1104 CHANGE DETERMINATION

A. __ YES ✓ NO Does the change described in the primary document involve a change to the NUREG 1104 which is incorporated into the Operating License?

B. If YES, (1) Identify the NUREG 1104 material subject to change:

SECTION(S)	PAGE(S)	TABLE(S)	FIGURE(S)

(2) Attach proposed NUREG 1104 changes.

(3) A SAFETY JUSTIFICATION is required for the NUREG 1104 changes.

IV. 10CFR50.59 UNREVIEWED SAFETY QUESTION DETERMINATION - ___ N/A

Complete the UNREVIEWED SAFETY QUESTION DETERMINATION on the SAFETY REVIEW RECORD.

___ YES ✓ NO Does the change described in the primary document involve an UNREVIEWED SAFETY QUESTION, as determined on the SAFETY REVIEW RECORD?

If YES, a SAFETY JUSTIFICATION is required to justify the acceptability of the change.

V. LICENSING CHECKLIST DETERMINATION

For all items in this part, identify where the change should be made and briefly describe the change in the space provided. (Attach additional sheets, if required.)

The following hazards analyses need to be updated as a result of the change described in the primary document.

___ YES ✓ NO II/1

7901070413

PMR 903

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11

(LICENSING REVIEW SUPPLEMENT)
(SHEET 3 of 3)

___ YES ✓ NO Fire Hazards

___ YES ✓ NO Pipe Break

___ YES ✓ NO Missile

___ YES ✓ NO Flooding

___ YES ✓ NO ALARA

The following qualification programs or licensing reviews need to be updated as a result of the change described in the primary document.

___ YES ✓ NO Environmental Qualification

___ YES ✓ NO Seismic Qualification

___ YES ✓ NO Human Factors Review

VI. COORDINATION AND APPROVAL

Coordination		Date	Approval	Date
Mech.	[signature]	4/9/85	Josiah A Long	4/19/85
			Responsible Engineer	
Elec./Cont. Sys.	[signature]	4/19/85		
Civil/Arch	NR			
Hanger	NR		A.J. DiPace	4/19/85
			Group Supervisor	
			[signature]	4/19/85
			Licensing Engineer	

7001 7 . 0 4 1 3 3

REDA No. N-L-808-XX Rev. 0
Initiating Document No. KWLB 84-118 Rev. -
PMP No. CS-163-W Rev. 5

SAFETY REVIEW RECORD
UNREVIEWED SAFETY QUESTION DETERMINATION
(SHEET 1 of 2)

INSTRUCTIONS:

1. Evaluate each of the criteria below for applicability to the change described in the primary document and check YES or NO as appropriate.

2. In the space below each criterion, document the applicability evaluation for each criterion; negative declarations and justifications are required. (Attach additional sheets, if required.)

3. If any criterion is applicable, an UNREVIEWED SAFETY QUESTION exists and a SAFETY JUSTIFICATION must be documented. NRC approval is required prior to making the change or conducting the test or experiment.

A. ___ YES ✓ NO Will the probability of occurrence of an accident previously evaluated in the safety analysis report be increased?

PMP CS-163-W involves issuance of BOP Safety Related Setpoints in a format requested by the client. These setpoints are consistent with FSAR Section 7.3.3 : 27 and do not affect the design basis of the plant.

B. ___ YES ✓ NO Will the consequences of an accident previously evaluated in the safety analysis report be increased?

See A above

C. ___ YES ✓ NO Is there a possibility that an accident of a different type from any evaluated previously in the safety analysis report may be created?

See A above

D. ___ YES ✓ NO Will the probability of occurrence of malfunctions of equipment important to safety, previously evaluated in the safety analysis, be increased?

See A above

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SAFETY REVIEW RECORD
UNREVIEWED SAFETY QUESTION DETERMINATION
(SHEET 2 of 2)

E. ___ YES ✓ NO Will the consequences of a malfunction of equipment, important to safety, previously evaluated in the safety analysis report, be increased?

See A above

F. ___ YES ✓ NO Is there a possibility that a malfunction of equipment, important to safety, may be created which is of a different type than any evaluated previously in the safety analysis report?

See A above

G. ___ YES ✓ NO Will a reduction in the margin of safety, as defined in the bases for any Technical Specification, result?

See A above

H. ___ YES ✓ NO An UNREVIEWED SAFETY QUESTION exists, (i.e., any of the above criteria are applicable).

See A above

I. ___ YES ___ NO ✓ N/A SAFETY JUSTIFICATION is attached.

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REDA No. N-L-808-XX Rev. 0
Initiating Document KNLB 84-118 Rev. NA
PMP No. CS-162-W Rev. 0

ALARA REVIEW RESULTS

PART I - CHANGES NOT INVOLVING RADIATION HAZARDS

It can be concluded that there is reasonable assurance that this proposed design change does not involve a radiation hazard. This proposed design change does not require design provisions or considerations to comply with ALARA guidelines.

Preliminary ALARA Review a. j. DiPerna 4/19/85
Primary Group Supervisor Date

Final ALARA Review [illegible] [illegible]
Licensing Engineer Date

PART II - CHANGES INVOLVING POTENTIAL RADIATION HAZARDS

I hereby verify that this proposed design change does include appropriate design provisions and considerations that comply with ALARA guidelines to the extent practicable. There is assurance that radiation exposures to plant operating personnel will be ALARA.

Preliminary ALARA Review ______________________
Primary Group Supervisor Date

Final ALARA Review ______________________
Licensing Engineer Date

PART III - APPROVALS

[illegible] 4/19/85
Project Engineer or Date
Ass't Project Engineer (Jobsite)

7001 7.0413

PMR00903

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11 of 11

REDA No. N-L-808-XX Rev. 0
Initiating Document No. KNLB84-BB Rev. -
PMP No. CS-163-N Rev. 0

FIRE PROTECTION REVIEW CERTIFICATION

A. PRELIMINARY FIRE PROTECTION REVIEW

I hereby certify that this design change does not impact the fire protection program or there is reasonable assurance that the proposed design change can be developed so that fire protection requirements will be met.

Signed:

Josiah A Long 4/19/85
Responsible Engineer Date
(Origination Discipline)

Reviewed: A final fire protection review (is) (is not) required.

4/19/85
Mechanical Group Supervisor or Date
Senior Mechanical Supervisor (Jobsite)

A. J. DiPenna 4/19/85
Group Supervisor or Date
Senior Supervisor (Jobsite)

B. FINAL FIRE PROTECTION REVIEW

I hereby certify that the documentation prepared in support of the design change incorporates design provisions to assure fire protection requirements.

Signed:

Mechanical Responsible Engineer Date

Signed:

Responsible Engineer Date
(Originating Discipline)

Mechanical Group Supervisor or Date
Senior Mechanical Supervisor (Jobsite)

Group Supervisor or Date
(Originating Discipline)

C. APPROVALS

4/19/85
Project Engineer or Date
Ass't. Project Engineer (Jobsite)

7 9 0 1 7 . 0 4 1 3 J

Attachment to PMR ØØ5Ø3 Pg. 1 of 3

KGE THE ELECTRIC COMPANY

KANSAS GAS AND ELECTRIC COMPANY

December 17, 1984

Mr J H Smith
Project Engineering Manager
Bechtel Power Corporation
15740 Shady Grove Road
Gaithersburg, MD 20877-1454

KNPLB 84-118 TE - 10970 K03
SUB: Setpoint Information
REF: 1) BLKE-1179
2) KNPLB 84-116

Dear Mr Smith:

The attachment should close out all items listed on Reference 1 for Kansas Gas & Electric action, with the exception of Item 12. Item 12 cannot be obtained until after ILRT, however since Item 12 is only required as a check, this should not hold up the setpoint generation. Included in the attachment is a prio[illegible] list, if you have any questions on this please contact Charle[illegible]is at (316) 364-8421, extension 1796.

Sincerely,

Melvin L Johnson
Manager Nuclear Plant Engineering

CRM:dab

cc: J Long w/a
R Ennis w/a
C M Herbst w/a
M [illegible] McMullen w/a

Item Number Per Reference 1

Item 2: See KNPLB 84-116

Item 3 & 4: All these concerns are resolved by the revised setting tolerance table. Please note, based on the revised values, Bechtel should revise the Tech Spec setpoints for instrument loops AL-37, 38, 39 and AC-231, 232, 233.

Item 5 & 6: All calibration periods are shown on the calibration frequency table.

Item 7: See KNPLB 84-116

Item 8: See KNPLB 84-116

Item 9: See KNPLB 84-116

Item 11: The following test data was obtained by KG&E Start Up.

Flow (x10 lbs/hr)	P (WC)
1.6	59.62
1.38	41.60
1.15	33.28
0.95	17.19
0.75	10.26
0.5	2.77

Item 12: This cannot be completed until after ILRT due to the ILRT valve lineups. Please note, Bechtel has stated that this information is not required for generation of the setpoint.

Item 13: Per memo from Chemistry the radiation monitors setpoints (safety limits) are as follows:

GT - 31,32	4.9 E-3 uci/cc
GG - 27,28	2.2 E-3 uci/cc
GK - 4,5	1.1 E-3 uci/cc
GT - 22,33	2.08 E-2 uci/cc

Kr-85 was used for preliminary data.
Please also supply data for the use of Xe-133.

The following is a list of the BOP instrument loops for which Bechtel is calculating setpoints in support of Reg Guide 1.105. The list sets priorities (from 1-6, 1 being the highest priority) in order that Bechtel may complete the calculations in a manner which will support fuel load and power ascension.

Priority*	System	Loops
1	GK	2,3
1	GK	4,5
1	GT	22,31,32,33
2	EF	43,44
2	EG	1,2
2	EG	107,108
3	AL	37,38,39
4	AC	231,232,233
5	GG	27,28
6	BB	17,18,19,20
6	EF	19,20
6	EG	62
6	EG	77,78
6	EN	15,16
6	EN	17,19
6	FC	25,26,125,126
6	GD	1,11
6	GM	1,11
6	JE	1,21(A,C)
6	JE	1,21(B)

*See sheet 2 of 2 for explaination of priority codes

Priority	
1	Required for surveillance testing in modes 1,2,3,4,5,6
2	Required for surveillance testing in modes 1,2,3,4
3	Required for surveillance testing in modes 1,2,3
4	Required for surveillance testing in mode 1
5	Required for surveillance testing in mode 1,2,3,4,5,6 after 1st refueling
6	No surveillance test requirement exists

OPERATIONAL MODES

MODE	REACTIVITY CONDITION, K_{EFF}	% RATED THERMAL POWER*	AVERAGE COOLANT TEMPERATURE
1. POWER OPERATION	$\geq$ 0.99	> 5%	$\geq$ 350 F
2. STARTUP	$\geq$ 0.99	$\leq$ 5%	$\geq$ 350 F
3. HOT STANDBY	< 0.99	0	$\geq$ 350 F
4. HOT SHUTDOWN	< 0.99	0	350 F>T > 200 F
5. COLD SHUTDOWN	< 0.99	0	$\leq$ 200 F
6. REFUELING**	$\leq$ 0.95	0	$\leq$ 140 F

*Excluding decay heat.
**Fuel in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.

Attachment to PMR 85-103

PLANT MODIFICATION PACKAGE

COVER SHEET

JOB 10466

Page 1 of 2

☐ PRELIM. ☒ FINAL

PMP NO: CS-163-W

DATE: 4-19-85 REV 0

RESPONSIBLE DISCIPLINE		RESPONSIBLE ENGINEER			
CONTROL SYSTEMS		J. A. LONG			
SYSTEM AFFECTED	**SAFETY EVALUATION RECOMMENDED**	**TEST OUTLINE RECOMMENDED**	**CHANGES TO FSAR RECOMMENDED**	**CHANGES TO TECH SPECS RECOMMENDED**	**SCHEDULE ISSUE DATE**
Various	☒ YES ☐ NO	☐ YES ☒ NO	☐ YES ☒ NO	☐ YES ☒ NO	4/19/85
					FORECAST DELIVERY DATE
					/ /

SCOPE

Issuance of Safety Related BOP Setpoints in accordance with Calculation J-K-GEN.

BECHTEL REFERENCES

BASIS OF CHANGE ☐ FCR No. ________ ☐ SFR No. ________

☐ SUPPLIER (Name & Date of Letter) ☐ CLIENT (Date & Log No. of Letter)

☒ OTHER REDA ※ N-L-808-XX

CHANGE CATEGORY: ☐ OPERABILITY ☐ MAINTAINABILITY ☐ IMPROVE PLANT EFFICIENCY

☐ REGULATORY ☐ VENDOR/SUBCONTRACTOR ☒ UTILITY REQUEST

APPROVALS

UTILITY APPROVAL

UTILITY DECISION: ☐ OTHER ________

☐ DO NOT PROCEED

☐ CONCEPTUAL ENGINEERING ONLY

☐ DETAILED ENGINEERING ONLY

☐ DETAILED ENGINEERING AND PROCUREMENT

DATE DESIGN REQUIRED ________ DATE MATERIAL REQUIRED ________

APPROVAL FOR PREPARATION OF

(✓)	YES	NO
SAFETY EVALUATION		
TEST OUTLINE		
FSAR		
TECH. SPECS.		

UTILITY REPRESENTIVE(S)

Name ________ Title ________ Date ________

Name ________ Title ________ Date ________

COMPLETED PMP APPROVALS (Bechtel)

RESPONSIBLE ENGINEER	DATE	ENGINEERING SUBCOORDINATOR	DATE
x Joseph A. Long	4/18/85	Jerre K. Olena, Jr	4.19.85
RESPONSIBLE GROUP SUPERVISOR	**DATE**	**PROJECT ENGINEER**	**DATE**
x A. J. DiPerna	4/19/85	x [signature]	4/19/85

Attachment to PMR 00903

PLANT MODIFICATION PACKAGE
DISCIPLINE CONTINUATION SHEET
JOB 10466

DISCIPLINE: CONTROL SYSTEMS
PMP NUMBER: CS-163-W
REV: 0
PAGE 2 OF 2

DOCUMENTS ASSOCIATED WITH PMP

NUMBER	REV	TITLE	SCHED REL DATE	REMARKS
CS-163-W-J-121000A(Q)	0	Control Logic Diagram: BOP Safety Related Setpoints	4/19/85	ISSUED
CS-163-W-J-121000B(Q)	0	Control Logic Diagram: BOP Safety Related Setpoints	4/19/85	ISSUED

Material Required
- NONE -

Item	Proc. Resp.	*Specification Number	*Specification Rev.	Purchase Order Number	Purchase Order Rev.	P.O. Item No	Scheduled Delivery Date

Remarks

Group Supervisor's Approval: A J DiPerna

Date: 4/19/85

* FILLED IN FOR BECHTEL PURCHASE ITEMS ONLY

TABLE 9.2-1

ESSENTIAL SERVICE WATER SYSTEM COMPONENT DATA

Essential Service Water Pump (all data is per pump)

Quantity	2 (100% each)
Type	Vert centrifugal - 2 stg. with packed stuffing boxes
Capacity, gpm	15,000
TDH, ft	361
Submergence required, ft	9
Material	
Case	Carbon steel
Impeller	Aluminum - Bronze
Shaft	Stainless Steel
Design Codes	ASME Section, III Cl. 3
Driver	
Type	Electric motor
Horsepower	1,750
RPM	885
Power Supply	4,000 V 60 Hz, 3-phase, Cl.1E
Design Code	NEMA
Seismic design	Category I

Essential Service Water Pump Prelube Storage Tanks (all data is per tank)

Quantity	2
Type	Vertical
Capacity, gallons	43
Design pressure	Atm.
Design temperature, F	122
Shell material	Carbon steel
Corrosion Allowance	1/16 inch
Design code	ASME Section III, Cl. 3
Seismic design	Category I

Essential Service Water Self Cleaning Strainers (all data is per strainer)

Quantity per unit	2
Capacity, gpm	15,000
Pressure drop, clean	1.1 psi
Pressure drop, dirty*	3.0 psi
Strainer openings	1/16 inch
Design pressure psig	200
Design temperature, F	100

*At start of backwash

TABLE 9.2-1

ESSENTIAL SERVICE WATER SYSTEM COMPONENT DATA

Essential Service Water Pump (all data is per pump)

Quantity	2 (100% each)
Type	Vert centrifugal - 2 stg. with packed stuffing boxes
Capacity, gpm	15,000
TDH, ft	361
Submergence required, ft	9
Material	
Case	Carbon steel
Impeller	Aluminum - Bronze
Shaft	Stainless Steel
Design Codes	ASME Section, III Cl. 3
Driver	
Type	Electric motor
Horsepower	1,750
RPM	885
Power Supply	4,000 V 60 Hz, 3-phase, Cl.1E
Design Code	NEMA
Seismic design	Category I

Essential Service Water Pump Prelube Storage Tanks (all data is per tank)

Quantity	2
Type	Vertical
Capacity, gallons	43
Design pressure	Atm.
Design temperature, F	122
Shell material	Carbon steel
Corrosion Allowance	1/16 inch
Design code	ASME Section III, Cl. 3
Seismic design	Category I

Essential Service Water Self Cleaning Strainers (all data is per strainer)

Quantity per unit	2
Capacity, gpm	15,000
Pressure drop, clean	1.1 psi
Pressure drop, dirty*	3.0 psi
Strainer openings	1/16 inch
Design pressure psig	200
Design temperature, F	100

*At start of backwash

SET POINT CHANGE REQUEST

EF	84	01
SYS DES	YR	NO

PART I

System EF-ESSENTIAL SERVICE WATER
Component No. PDS-19 1/2 & 20 1/2 Safety-Related [X] Yes [ ] No
Component Function SELF CLEANING STRAINER D/P
Computer Point(s) NA

	Present	Requested
Setpoint	LATER	5.5 PSID
Tolerance	LATER	+-0.5%

Reference Drawing JK2 EF03A/2 Manual NA
Requested by BARRY BOYLE Date 3 / 13 / 84
I&C Supervisor Approval N/A RESULTS INITIATED Date / /
Reason for Change ADD PRELIMINARY SETPOINT

PART II

Affected Drawings NA Procedures NA
FSAR or Tech Spec Section(s) NA
FSAR or Tech Spec Limit(s) NA
ALARA Review Date / / NA [X]

Engineering Review:

Recommended [X] Rejected [ ] Date 3 / 13 / 84
Remarks PDS 19 1/2 & 20 1/2 WILL BE ADDED TO WCGS-TPSD AS ATTACHED. PLEASE NOTE THESE ARE PRELIMINARY SETPOINTS WHICH WILL BE FINALIZED AFTER BECHTEL COMPLETES THEIR LOOP UNCERTAINTY CALCULATIONS.

Operations Review: Signature [signature] Date 3-13-84

Recommended [ ] Rejected [ ] Date / /
Remarks

Signature N/A Date

PART III

Set Point Change Request Approved [X] Disapproved [ ]
Plant Support Supervisor [signature] Date 3 / 13 / 84
PSRC Review: Approved [ ] Disapproved [ ] NA [ ]
PSRC Chairman Date / /

PART IV

Actual Setpoint
Completed by [signature] Date 1 / 13 / 86

CC: Originator
Results Engineering
Computer Engineering
I&C
Operations
Training
NPE

RECEIVED
JUL 07 1986
RECORDS ROOM

K05-005

Wolf Creek Generating Station

PSRC REVIEW SHEET

Figure 8

PROCEDURE NUMBER AND REVISION: ADM-05-102 Rev 0

PROCEDURE TITLE: Total Plant Setpoint Document Change Request
Request # EF-84-01

Superintendent of Operations		Date	
Superintendent of Technical Support		Date	
Superintendent of Maintenance		Date	
Chemist	[signature]	Date	3-14-84
Health Physicist		Date	
I&C Supervisor		Date	
Superintendent of Plant Support	[signature]	Date	3-13-84
Reactor Supervisor	[signature]	Date	3/14/84
Results Supervisor	[signature]	Date	3-13-84
Superintendent of Regulatory, Quality and Administrative Services	[signature]	Date	3-14-84
Quality Assurance (ADM, STS, OCP only)		Date	

Approved: [signature] for FT Rhodes Date 3-14-84
PSRC Chairman

RECEIVED
JUL 07 1986
Q.A. RECORDS ROOM

KOS-005

SETPOINT INDEX

INPUT FORM

SYS: EF

FUNCTION CODE: PDS

SEQ. NO.: 0019

SUF: 1/2

TS:

SERVICE DESCRIPTION: ESW SELF-CLN STR A DP

Safety Limit of 6.0 PSID − 0.37 SP [illegible] from BLSE 10.904
.05 [illegible] ed Cd Error
.08 [illegible] to round off
RESULTS .50

INSTRUMENT SETPOINT: 5.50 PSID

[illegible]

TOLERANCE: +-0.5%

[illegible]

RESET: ADJ

[illegible] from J301-11

RANGE/SPAN: 0-10 PSID

Accuracy used in BLSE 10.904
([illegible] estimates)

ACCURACY: +-0.5%

Also J110-338-17 Foxboro TI 2AP-100 SP Repeatability

Safety Limit of [illegible] from JK2 EF03A/2

SOURCE DOCUMENT: RESULTS

COMPUTER ADDRESS: NA

MANUFACTURER/MODEL (RESERVED-2): FI80/2AP+ALM-AR

SG: 1

REV:

REMARKS 1.

REMARKS (Cont.)

REMARKS 2

REMARKS (Cont.)

KCS-00[illegible]

SETPOINT INDEX

INPUT FORM

SYS	FUNCTION CODE	SEQ. NO.	SUF	TS
EF	PDS	0020	1/2	

SERVICE DESCRIPTION

ESW SELF-CLN STR B DP

Safety Limit of 6 PSID − 0.37 SRSS Estimate from SLSE 10,904
.05 Projected Cal Error
.08 Margin to roundoff
.50

INSTRUMENT SETPOINT

5.50 PSID

Accuracy
TOLERANCE

+-0.5%

[illegible] 1% but [illegible] to [illegible]
Conservatively [illegible]

RESET

ADJ

PT [illegible] from J[illegible]

RANGE/SPAN

0-10 PSID

Accuracy used in SLSE 10,90[illegible]
(Less [illegible])

ACCURACY

+-0.5%

Also J110-3SS-17 [illegible] TI
2AP-100 S.P. Repeatability

[illegible] from [illegible]

SOURCE DOCUMENT

RESULTS

COMPUTER ADDRESS

NA

MANUFACTURER / MODEL (RESERVED-2)

F180/2AP+ALM-AR

SG

4

REV.

REMARKS 1

REMARKS (Cont.)

REMARKS 2

REMARKS (Cont.)