In Reply Refer To:
Docket: STN 50-482

JUL 18 1988

Wolf Creek Nuclear Operating Corporation
ATTN: Bart D. Withers
President and Chief Executive Officer
P.O. Box 411
Burlington, Kansas 66839

Gentlemen:

Thank you for your letters of April 8 and April 28, 1988, in response to our the NRC letter of February 8, 1988. This letter acknowledges receipt of your response to our Safety System Outage Modifications Inspection (NRC Report No. 50-482/87-32). We will review the implementation of your corrective actions during a future inspection to determine that full compliance has been achieved and will be maintained.

Sincerely,

Original Signed By:
L. J. CALLAN

L. J. Callan, Director
Division of Reactor Projects

cc:
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P.O. Box 411
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Wolf Creek Nuclear Operating Corporation
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Kansas Radiation Control Program Director

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7/15/88	7/15/88	7/15/88

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WOLF CREEK
NUCLEAR OPERATING CORPORATION

John A. Bailey
Vice President
Engineering and Technical Services

April 8, 1988

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ET 88-0051

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

Reference: Letter dated February 8, 1988 from D. M. Crutchfield, NRC to B. D. Withers, WCNOC
Subject: Docket No. 50-482: Inspection Report 50-482/87032-Safety Systems Outage Modifications Inspection

Gentlemen:

The purpose of this letter is to establish the revised submittal date for the response to Inspection Report 50-482/87032 concerning the Safety Systems Outage Modifications Inspection (SSOMI) conducted at Wolf Creek Generating Station. Based on discussions with the NRC Project Manager for Wolf Creek Generating Station, Wolf Creek Nuclear Operating Corporation (WCNOC) will respond to the subject Inspection Report on or before April 29, 1988.

The Reference transmitted the SSOMI Inspection Report and requested a response to the concerns identified in the Inspection Report. Due to the length and complex nature of the SSOMI Inspection Report, WCNOC verbally requested and received an extension for submitting the repsonse to this Inspection Report.

If you have any questions concerning this matter, please contact me or Mr. O. L. Maynard of my staff.

Very truly yours,

John A. Bailey

John A. Bailey
Vice President
Engineering & Technical Services

JAB/jad

cc: B. L. Bartlett (NRC)
D. M. Crutchfield (NRC)
R. D. Martin (NRC)
P. W. O'Connor (NRC), 2

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WOLF CREEK
NUCLEAR OPERATING CORPORATION

RECEIVED
MAY - 4 1988

Bart D. Withers
President and
Chief Executive Officer

April 28, 1988

WM 88-0113

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

Reference: Letter dated February 8, 1988 from D. M. Crutchfield, NRC to B. D. Withers, WCNOC
Subject: Docket No. 50-482: Response to Safety Systems Outage Modifications Inspection Report 50-482/87032

Gentlemen:

Attached is Wolf Creek Nuclear Operating Corporation's response to the Safety Systems Outage Modifications Inspection Report transmitted in the Reference. The response provides a description of programmatic and organizational enhancements made at Wolf Creek Generating Station subsequent to the inspection as well as detailed responses to the specific findings.

If you have any questions concerning this matter, please contact me or Mr. O. L. Maynard of my staff.

Very truly yours,

BDWithers

Bart D. Withers
President and
Chief Executive Officer

BDW/jad

Attachment

cc: B. L. Bartlett (NRC), w/a
D. M. Crutchfield (NRC), w/o
R. D. Martin (NRC), w/a
P. W. O'Connor (NRC), 2 w/a

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RESPONSE TO
SAFETY SYSTEMS OUTAGE MODIFICATIONS
INSPECTION REPORT

(50-482/87-032)

APRIL 28, 1988

WOLF CREEK
NUCLEAR OPERATING CORPORATION

INDEX TO SSOMI ITEMS

*Since no concerns or weaknesses were identified in the Inspection Report, WCNOC has not responded to these items.

SECTION I

OVERALL CONCLUSIONS

INTRODUCTION

During the second refueling outage at Wolf Creek Generating Station, a Safety Systems Outage Modifications Inspection (SSOMI) was conducted by the NRC's Office of Nuclear Reactor Regulation. The SSOMI was completed in two portions. The first portion dealing with Design and Procurement was conducted on November 2-13, 1987, while the second portion dealing with Installation and Testing was conducted on November 9-20,1987.

The SSOMI identified both strengths and weaknesses in the modification activities that were performed at Wolf Creek Generating Station (WCGS). These findings were documented and transmitted to Wolf Creek Nuclear Operating Corporation (WCNOC) in NRC Inspection Report 50-482/87032. This Inspection Report stated that, "The modification activities inspected by the SSOMI team during the Wolf Creek outage, including procedures, installed equipment and materials, and workmanship by crafts, were generally in accordance with NRC requirements and licensee commitments". In addition, specific strength were noted related to, "...the acquisition and control of equipment and materials, the trend analysis of quality findings and reported deficiencies, and workmanship by maintenance personnel".

The Executive Summary of the Inspection Report also identified weaknesses in the general areas of Management Controls, Engineering Support and Evaluations, and Corrective Actions. Each of these areas is specifically addressed in Section I of WCNOC's response to the Inspection Report. Section II of WCNOC's response presents a detailed response to each of the SSOMI team findings categorized as either a weakness or concern.

It should be noted that findings categorized as strengths have been omitted from Section II. Each finding in Section II is presented with a Restatement of the Finding, a General Discussion of the Finding, a Response to the Finding, Actions Taken, and Conclusions based on the Finding.

MANAGEMENT CONTROLS

As stated in NRC Inspection Report 87-032, Safety Systems Outage Modification Inspection, "modification activities inspected by the SSOMI team during the Wolf Creek outage, including procedures, installed equipment and materials, and workmanship by crafts, were generally in accordance with NRC requirements and licensee commitments". However, as a result of WCNOC management concern with regard to the significant operational events that occurred during WCGS's second refueling outage and individual issues identified by the NRC, WCNOC has instituted several organizational enhancements to alleviate any perceived weaknesses in WCNOC's overall management control.

In order to more closely control day to day work activities during outages, the Outage Manager will report to the Plant Manager rather than the Vice-President Nuclear Operations. The Outage Manager's function and authority will be clearly reflected in Administrative Procedures. The Outage Manager will have a support staff consisting of Scheduling personnel and two Senior Technical personnel.

As an improvement to the Maintenance Organization, Maintenance has been combined with Facilities and Modifications to form an organization titled Maintenance and Modifications. This organizational change combines all maintenance activities under a single manager and creates a large skilled labor pool to be used on any maintenance or modification task. In context with this change, significant management attention has been focused on assuring that unnecessary levels of supervision have been eliminated and establishing more direct management control in all work activities within Maintenance and Modifications.

In addition to these organizational enhancements, WCNOC Management formally restated to all personnel the importance of strict adherence to written procedures during the performance of work activities, testing, equipment repair, and plant modifications. This is being reinforced by mandatory pre-work briefings. Daily management meetings have been established to ensure direct management control and supervision of all ongoing work activities.

To enhance communications and provide more management involvement in day-to-day activities, the daily planning meeting has been restructured to focus more on problems and corrective actions than on work status. The meeting is attended primarily by management personnel to provide the necessary level of authority to resolve problems and take necessary corrective actions. The meeting is normally attended by one or more of the Corporate Officers to provide senior management support.

The above organizational modifications and programmatic enhancements will strengthen the performance of WCNOC and subsequently WCGS. These actions should minimize the potential for operational or management weaknesses during future outages at WCGS. The individual examples identified in this section are fully discussed in other sections of this response to NRC Inspection Report 50-482/87032.

ENGINEERING SUPPORT AND EVALUATIONS

NRC Inspection Report 50-482/87032 made the statement that, 'The engineering support provided for a number of recent modifications and maintenance activities was found to be inaccurate or lacking in thoroughness.'. The Inspection Report also stated that, 'The SSOMI team found that modified designs were traceable to the original design bases and regulatory requirements, that correct design information was utilized, and that applicable design controls were implemented.' It should be noted that the activities discussed as engineering support in the Inspection Report include activities which are performed by different organizations within WCNOC and not just the design engineering group. Engineering support for day-to-day operational and maintenance activities is performed by various groups under the Plant Staff. Engineering support for modifications and other design activities are performed by Nuclear Plant Engineering (NPE).

The organizational enhancements and increased management involvement discussed in the Management Controls section of this response will improve the support activities being performed by the engineering groups. The interaction between organizations in the daily management meetings will help to focus efforts into a single direction so that each organization is aware of what is required to accomplish any given task. In addition, these meetings ensure that individual

work assignments are being performed by the organization with the proper experience and expertise to accomplish the task in the most effective manner. WCNOC believes that with this strong foundation in the design area along with enhancements made because of events during the second refueling outage and items identified by the SSOMI team, the weaknesses identified by the SSOMI inspection in the engineering support and evaluations area will not reoccur at Wolf Creek Generating Station.

CORRECTIVE ACTIONS

NRC Inspection Report 87-032 identified several examples which indicate weakness involving the adequacy of corrective actions for identified deficiencies, including the identification of root causes, evaluation of related areas for similar deficiencies, and actions to prevent recurrence. It should be noted that an earlier NRC Inspection concerning corrective action was conducted during the periods May 18-22, and June 2-5, 1987. Upon receiving this NRC Inspection Report, 87-011, on November 3, 1987 WCNOC took steps to enhance the corrective action program as documented in letter WM 88-0028, dated January 29, 1988. An integrated corrective action program in the form of a WCNOC General Procedure, KGP-1210, "Corrective Action for Programmatic and Implementation Deficiencies", has been developed. The procedure establishes a standardized method for all WCNOC organizations to document and respond to programmatic or implementation quality problems.

During a WCNOC Quality Department audit conducted between December, 1987 and February, 1988, concerning the implementation of this procedure, several inconsistencies were identified in the methods various organizations were using to implement the procedural requirements. This has resulted in a complete review of KGP-1210. The procedure is currently being revised to provide plant personnel with a more usable explanation of determination of root causes. It is expected that the revision to KGP-1210 should be implemented by June 1, 1988. A training program is currently under development to provide direction to the personnel responsible for implementation of the revised KGP-1210. In addition, a seminar by EG&G on "Accident Investigation", is scheduled to be conducted at the Wolf Creek Generating Station during May, 1988. The above noted revision to KGP-1210 and training will serve to strengthen the overall corrective action program of WCNOC.

SUMMARY

WCNOC believes the actions described above in conjunction with the specific actions described in Section II of this response fully address the weaknesses identified by the SSOMI Inspection Report. It is acknowledged that during the SSOMI review of modification activities and significant operational events several specific weaknesses were identified, however, WCNOC believes that the overall organization and program enhancements described above will serve to optimize the present and future operation of WCGS.

SECTION II

2.1.2.1 PMR 2024: BATTERY CHARGER AC ALARM SETPOINT

FINDING:

This PMR changed the battery charger AC input alarm setpoints to eliminate spurious AC alarms and was required because of previous changes to the AC system made in 1985 by PMR 1345 which reduced the AC input voltage at selected safety related and non-safety related battery chargors. PMR 2024 indicated that the cause of spurious AC undervoltage alarms was an incorrect voltage transformation ratio assumed in the original System Relay Setting Calculation (H-12). A review of the latest revision to this calculation indicated that the transformation ratio was corrected in April 1983. The correct voltage transformation ratios were indicated for nine of the eleven battery chargers on Relay Setting Tabulation Drawing E-11028(Q) issued in June 1984, however the new relay settings had not been added at that time.

The SSOMI team reviewed Bechtel Voltage Calculation (B-8), which was the basis of PMR 1345, searching for other possible reasons for the spurious undervoltage alarms. The impedance data used as a calculational input for one of the safety related load center transformers was based upon General Electric test data, however this impedance value did not agree with the transformer's nameplate data. Engineering Department personnel indicated that although the test data used in the calculation was not specific to the equipment installed at Wolf Creek Generating Station (WCGS), the error accounted for a calculated voltage error of less than two percent. The SSOMI team considered that the failure to identify the error in the Bechtel Voltage Calculation is indicative of a weakness in engineering evaluations.

GENERAL DISCUSSION OF FINDING:

During the review of PMR 2024, PMR 1345, system relay calculation H-12, voltage calculation B-8, and relay setting Tabulations (Drawings E-11027 and E-11028(Q)) The SSOMI team identified two distinct concerns:

1. Corrections made in June 1983 to Calculations H-12 and H-13, concerning battery charger low AC input voltage alarms and potential transformer ratios, were only partially incorporated into relay setting tabulations E-11027 and E-11028(Q) during the June, 1984 revision to those documents.

2. The impedance data used in Calculation B-8 for one of the load center transformers was based on the manufacturers test data which does not agree with the transformer nameplate.

RESPONSE TO THE FINDING:

1. The corrections made in the subject calculations identified a lower setpoint value for the battery charger low AC input voltage. It is important to note that the purpose of the alarm is to alert the Control Room operators to low AC input voltage on the battery chargers. The original setting provided for proper monitoring of the electrical distribution system at Wolf Creek Generation Station. It is good engineering practice to not eliminate the conservatism in the system

design setting on the sole basis that a revision to a design calculation identifies additional available margin. Therefore at the time of the calculation revision, the new setpoints calculated did not need to be incorporated into the setpoint documents for Wolf Creek. Only after the alarms became frequent due to the implementation of PMR 1345, which optimized the transformer taps for the Wolf Creek Electrical Distribution System, did a need arise to reduce the alarm setting to the value identified in the previously revised design calculation.

It is true that for two of the eleven battery charger control power transformers, the subject setpoint document listed the wrong transformation ratios. These two chargers were added as design enhancements during 1982. The oversight was corrected promptly when discovered but did not affect the conservatism of the alarm setpoint.

2. The transformer manufacturers test data and certified test data report which were used as a basis for the subject Voltage Calculation B-8, are judged to be accurate as issued. The subject certified test data report reflects the actual results of a performance test on the installed equipment as certified by the manufacturer. It is noteworthy that the equipment serial number recorded on the test data and to which the referenced calculation refers, also matches the serial number stamped on the transformer nameplate. Therefore, the test data and calculation are applicable and specific to the transformer installed at Wolf Creek. In addition, at the time the transformer was procured, the nameplate was not required to reflect the values recorded on the certified test report (reference ANSI C57.12.80 1973) and therefore, only reflected the design impedance value. It was, in more recent revisions, that the applicable industry standards for transformers (501 KVA and larger) required that the actual test data be reflected on the nameplate.

CONCLUSION:

1. The SSOMI team identified that there were two incorrect non-1E battery potential transformer ratios on the applicable Relay Setting Tabulation drawing. However, these errors are not considered reflective of general engineering weakness and were corrected when discovered. The item concerning the lack of incorporation of the revised calculation into the applicable Relay Setting Tabulation for the battery chargers is not considered a deficiency. The fact that the input voltage low alarm setpoint was 94% (on a 480 volt base) instead of the 90% value, determined by the design calculation is considered conservative. These types of margins are not utilized unless necessary in order to maintain the maximum safety margin for protection systems at Wolf Creek Generating Station. This approach demonstrates a strength in engineering evaluations, and therefore, requires no further action.

2. The SSOMI team item concerning engineering's failure to identify an apparent error in Voltage Calculation B-8 due to data inconsistencies between transformer nameplate data and certified vendor test reports does not constitute a deficiency. The subject Calculation, B-8, as well as the cited certified vendor test report data, all considered correct and applicable to the transformer installed. At the time the transformer was procured, the nameplate was not required to reflect the values recorded on the certified test report (reference ANSI C57.12.80 1973) and therefore, only reflected the design impedance value. It was, in more recent revisions, that the applicable industry standards for transformers (501 KVA and larger) required that the actual test data be reflected on the nameplate. Therefore, no further action is required and no revision to these documents or the transformer nameplate is required.

2.1.2.2 PMR 899: ACCUMULATOR LEVEL TRANSMITTERS

FINDING:

This PMR replaced Accumulator Tank Barton level transmitters with Rosemount level transmitters and relocated them to the side of the tank using the existing sensor taps. The SSOMI team identified the following discrepancies with this PMR:

a. The new sensor connections to the reference leg were five feet higher, which provided an additional 41.36 cubic feet of water in the tank at the minimum level setpoint than designed. As a result, a smaller volume of nitrogen gas remained in the Accumulator Tank to provide for water injection into the primary system in the event of a LOCA. The Technical Specifications required that a minimum of 818 cubic feet of water be injected from the tank into the primary system in the event of a LOCA. The actual tank pressure required with the new volume of water was not determined. The design nitrogen gas pressure of 585 psig had some allowance for conservatism, however this allowance was also unknown.

b. A root cause analysis for changing the level transmitters was not provided. Although a significant amount of data existed to justify the change, this data had not been formally communicated within the company. In addition, the Q-list, which lists all safety related equipment in the plant, was not revised as required.

The failure to evaluate the change in nitrogen gas pressure required for Accumulator Tank injection, including calculation of the root cause for the modification is a weakness in the engineering area.

GENERAL DISCUSSION OF FINDING:

a. The design modification stipulated in PMR 899 called for utilization of the same tank taps, standpipe taps, and diaphragm seal locations as the original installation. The replacement Rosemount transmitters were relocated to a lower elevation than the Bartons (approximately 5 feet to 17 feet difference depending on transmitter) and are now placed at the midpoint between the diaphragm seals rather than above the upper seal.

b. Many letters were written as a result of the Barton instrument problems identified at WCGS. The responsible engineer's correspondence file on the subject was turned over to the SSOMI team for review. Although this file did not contain all the project correspondence, it did include the minutes of a meeting held at Wolf Creek on January 25, 1984, with representatives in attendance from KG&E (including the Project Director), Union Electric, Westinghouse, ITT Barton, and Nuclear Projects Incorporated. During this meeting, KG&E/UE failures were discussed, including failure modes, and action plans were established.

   A review of the PMR package indicates that Q-List Change Notices were part of the package and were referenced on the Modification Document Form (KGF-6). Neither the SSOMI team nor the responsible engineer located the Change Notices in the copy of the package provided to the SSOMI team.

RESPONSE TO THE FINDING:

a. Since the tap connections and diaphragm seal locations were unchanged, the original tank level setpoint calculations remain valid. The replacement of a transmitter as well as its relocation from above the upper diaphragm seal to the seal midpoint would result in the need for calibration. This was performed by Instrumentation and Controls personnel for PMR 899 in accordance with STS IC-908A 'Channel Calibration Accumulator Level Transmitters'. These actions result in the tank water content remaining the same as prior to the modification and being within the Technical Specification bounds. No engineering reanalysis of the effect of increased water content was necessary.

b. The January 25, 1984 meeting minutes and other correspondence such as a January 27, 1984 letter from John Bailey of the Wolf Creek project to Kent Brown, KG&E Vice-President (subject-Barton Transmitter problems and intended solutions) clearly show that management and the project as a whole was being informed of the Barton transmitter problem.

During discussions between the SSOMI team and the responsible engineer, it was mentioned that Q-List change documents did not exist for the newly installed transmitters. The responsible engineer could not remember generating such documents and thus acknowledged the concern. A review of the PMR package was undertaken as a result of this finding revealed that the Q-List Change Notices did exist.

CONCLUSION:

a. No changes were made to the system which would require a new analysis. WCNOC does not consider this item to be a deficiency.

b. WCNOC believes that the project, including upper management, was aware of the problem with Barton transmitters, and that existing documents indicate that a root cause determination was sought as a part of resolving the issue. The PMR package does contain an appropriate Q-List Change Notice.

2.1.2.3 PMR 2167: ELECTRICAL EQUIPMENT ROOM NO. 1403 CHILLER

FINDING:

This PMR added additional cooling to Electrical Equipment Room No. 1403 and installed a room temperature indicator controller and an automatic chiller water control valve, TIC 185 and TV 185, respectively. A Field Change Request (FCR) was subsequently issued to delete TIC 185 and TV 185 and to add a manual globe valve, V150. The change required the operator to manually control the room temperature to 75°F ± 5°F by adjusting valve V150. This installation of a manual valve was inadequate because a temperature indicator was not provided to measure the temperature of Room 1403, the TS surveillance procedures did not include this room for periodic surveillance and the room was not required to be monitored for environmental conditions. In addition, the basis for the acceptability of 75°F was not addressed. The failure to evaluate the effect of the FCR change on the PMR and the failure to document the basis for the design change is an example of a general weakness in the engineering area.

GENERAL DISCUSSION OF FINDING:

In Item 2.1.2.3 the SSOMI team determined that the installation of a manual valve (via an FCR) was inadequate because a temperature indicator was not provided to measure the temperature of Room 1403, the surveillance procedures did not include this room for periodic surveillance and the room was not required to be monitored for environmental conditions. In addition the SSOMI team advised that the basis for the acceptability of 75°F room temperature was not addressed in the PMR. The SSOMI Team indicated that failure to evaluate the effect of the FCR change on the PMR and the failure to document the basis for the design change is an example of general weakness in the engineering area.

RESPONSE TO THE FINDING:

PMR 2167 was revised, as requested by FCR 02167-F-009 because of material unavailability, to allow the temporary replacement of the three-way flow control valve GL-TV-185 with a manual globe valve. The disposition of the subject FCR stated that the aforementioned globe valve 'shall' be replaced with the three-way flow control valve upon availability. Although the three-way valve has been received but not yet installed, PMR 2167 has been revised and reissued to provide for the installation of the three-way valve. Additionally, in the interim the disposition stated that the room temperature was to be frequently monitored and the globe valve adjusted to maintain the room design temperature. This was to be accomplished, once the unit had been placed in service, by the Auxiliary Building Operator at least once per shift and documented per ADM 02-030 'Reading Sheets and Shift Round Instructions Checklist'. At the time of the SSOMI, the additional fan coil unit had not been placed in service. This revision to the design package demonstrates not a weakness in engineering, rather, an engineering strength. Additionally, an installed temperature indicator is not warranted for the temporary use of the globe valve.

The Technical Specification surveillance requirements do not address the area temperature monitoring for the room, because this room is considered a mild environment as defined in USAR Page 3.11(B)-29.

The air handling unit (SGL02) for the room and its associated power and control are not safety related as there are no safety design bases associated with the system. The additional fan/coil unit SGL20 was added to operate in parallel with the original SGL02 unit to provide increased cooling capability for the room. The reason that the design change was issued, was to provide for the installation of a room cooler that would provide cool air in the vicinity of the non-safety related, yet temperature sensitive, Full Length Rod Control Cabinet. An evaluation of the cooling load for the room was conducted as well as for the heat sensitive Full Length Rod Control Cabinet. The Engineering evaluation included the review of existing supplier documentation for the cabinet, and the performance of a calculation to determine the cooling capacity required of the new unit. The basis for the design change was documented on the PMR cover sheet form, with further supporting documentation (i.e. calculations, supplier documentation) existing as backup information. Westinghouse documentation provided information that the preferred ambient temperature would be approximately 77°F, but not to exceed 50-104°F. This was the basis for the 75°F ± 5°F design temperature.

CONCLUSION:

The disposition stated that the room temperature was to be frequently monitored and the globe valve adjusted to maintain the room design temperature. At the time of the SSOMI, the additional fan coil unit had not been placed in service. Since the unit has been placed in service the room temperature has been read by the Auxiliary Building Operator at least once per shift and documented per ADM 02-030 'Reading Sheets and Shift Round Instructions Checklist'.

As demonstrated in the discussion above, the effect of the FCR change was evaluated and the basis for the design change was documented. WCNOC does not consider this item to be a deficiency.

2.1.2.4 PMR 1634: REACTOR COOLANT DRAIN TANK ISOLATION VALVE

FINDING:

This PMR installed an isolation valve upstream of relief valve H-7160 to simplify inspection and repair of the relief valve. Previously, repair or replacement of the relief valve required a plant shutdown. The following discrepancies were identified:

a. The Results Engineering Group issued Temporary Modification 86-24-4B in March 1986, to gag Reactor Coolant Drain Tank relief valve HBV-7160. The 10 CFR 50.59 Safety Evaluation performed indicated that this modification did not affect the tank's overpressure protection because the tank was protected by relief valve HBV-7169. Although valve HBV-7169 had a larger spring to accommodate a higher set pressure, the evaluation indicated that the setpoint was below the design pressure of the relief tank and therefore provided adequate overpressure protection. However, the Safety Evaluation did not evaluate the required flow rate, the relative flow rates of the two valves at 110% of the tank design pressure (110 psi) and the differences in configuration. Therefore, the evaluation did not demonstrate the second relief valve provided equivalent or adequate protection for the tank.

b. The new isolation valve added by PMR 1634 had less flow area than the relief valve inlet, contrary to the requirements of Paragraph UG-135, Appendix M, Section VIII, of the ASME Boiler and Pressure Vessel Code Division 1-1974. An analysis to verify that the isolation valve would not reduce the capacity of the relief valve was not performed.

c. In addition, instrumentation was not installed at the isolation valve location to enable appropriate emergency actions if the tank was overpressurized.

The failure of the safety evaluation to demonstrate that the second relief valve provided adequate overpressure protection for the drain tank and to verify that the isolation valve would not reduce the capacity of the installed relief valve is an example of a general weakness in the engineering area.

GENERAL DISCUSSION OF FINDING:

a. The SSOMI team felt that the 10CFR50.59 Safety Evaluation did not demonstrate that the second relief valve off of the Reactor Coolant Drain Tank (RCDT) provided equivalent or adequate protection for the tank, as the relief valve that was gagged. The RCDT is an ASME Code Section VIII tank with a working pressure of 100 psig. This tank has two relief valves off of it, one set at 100 psig and the other at 25 psig. The 25 psig set relief valve is the only relief valve which is required by design for providing overpressure protection. The 25 psig relief valve was gagged by the subject Temporary Modification Order. The associated 10CFR50.59 Safety Evaluation lacked sufficient documentation that the system bases for the two relief valves with different setpoints was understood. The system consequences of gagging the 25 psig relief was not discussed. The evaluation did not document how relief valves

considered identical but not exactly alike provided the same overpressure protection and relief capacity for the RCDT.

b. The SSOMI team determined that the new isolation valve added by PMR 1634 had less flow area than the relief valve inlet, contrary to the requirements of Paragraph UG-135, Appendix M, Section VIII, of the ASME Boiler and Pressure Vessel Code Division I-1974. The SSOMI team indicated that failure to verify that the isolation valve would not reduce the capacity of the installed relief valve is an example of a general weakness in the engineering area.

c. In addition, the SSOMI team indicated that instrumentation was not installed at the isolation valve location to enable appropriate emergency actions if the tank was overpressurized.

RESPONSE TO THE FINDING:

a. The Safety Evaluation clearly stated that the two relief valves were supplied by the same manufacturer and are identical in size, style, type, assembly number, and material specifications. Additionally, it stated that the ASME code relief requirements for the tank were maintained. The evaluation did not state that after full accumulation of the relief valve, the relief capacity is a function of the valves' orifice size and that the orifice sizes of the two valves were identical. The root cause of this finding is an inadequately documented review to support the subsequent evaluation conclusions.

b. During the design process of PMR 1634, a review of ASME B&PV Code, Section VIII, Paragraph UG-135 and Appendix M was made. Engineering made the interpretation that "full area stop valve" meant and was intended to mean that the isolation valve between a pressure vessel and its pressure relieving device was to be of the identical line size as the piping and relief valve. Subsequent to the concern expressed by the NRC, Engineering discovered that an interpretation had been made by the ASME (Interpretation VIII-1-83-338) for the definition of a "full area stop valve" that confirmed that the minimum flow area within the stop valve be at least equal the inlet area of the pressure relief device. The intended stop valve had a minimum flow area of 2.64 square inches versus 3.14 square inches for the pressure relief valve. A technical evaluation of the stop valve to be utilized indicated that no actual reduction in capacity of the pressure reducing device would occur (i.e., design function would not have been effected).

c. Engineering recognizes that isolation of HBV-7160 is not a recommended mode of operation with the RCDT in service, but that maintenance activities for HBV-7160, if necessary, are not possible without a means of isolation. The RCDT is an unfired pressure vessel; thus, it is pressurized only from external sources. If HBV-7160 is removed from the RCDT as a means of overpressure protection, then administrative action would be required to; remove the sources of pressure from entering the RCDT, or; ensure that HBV-7169 is available to the RCDT, and; comply with the Code requirement that an authorized person monitor and restore the

stop valve to a locked open status. Permanent local instrumentation for monitoring an activity as infrequent as postulated above is not considered warranted.

ACTIONS WHICH HAVE BEEN OR WILL BE TAKEN:

a. In the two years since this evaluation, significant experience and program development has occurred. The Results Engineering 10CFR50.59 guidance has been augmented to include in each Safety Evaluation the design bases or function of a component and fully describe how this component and its system are affected by the change or modification. Relief valves which appear indentical and have different setpoints have different springs, but at full accumulation are, in fact, identical. This should be documented when identical relief valves are discussed.

b.,c.
Subsequent to the review of the ASME Code Interpretation VIII-1-83-338, PMR 1634 was withdrawn from implementation status. Based on current system performance without the modification, this de ign change is not presently deemed necessary.

CONCLUSION:

a. The system was restored on December 9, 1986. WCNOC considers this item closed.

b.,c.
Even if the proposed modification had been implemented, the modification would not have effected the original relief capacity of the valve. Subsequent to the review of the ASME Code Interpretation VIII-1-83-338, PMR 1634 was withdrawn from implementation status. Based on current system performance without the modification, this design change is not presently deemed necessary.

2.1.2.5 PMR 1613: VALVE LEAKOFF CONFIGURATIONS

FINDING:

This PMR redesigned the leakoff configuration for valves BG-HV8146, BB-8074 A,B,C, and D, BB-8055, BG-LC459 and BG-LCV460. The PMR added flexible hoses and a shutoff valve on the leakoff lines of each of these valves. Since these valves were subject to Reactor Coolant System (RCS) pressure and temperature, the flexible hoses could have been pressurized if the valve packing leaked with the leakoff isolation valves closed. The manufacturer's rated pressure for the flexible hoses is less than RCS pressure, therefore the flexible hoses are subject to failure. The failure to adequately evaluate this modification is an example of a general weakness in the engineering area.

GENERAL DISCUSSION OF FINDING:

PMR 1613 redesigned the valve stem packing leakoff configurations for valves BG-HV8146, BB-8074A,B,C and D, BB-8085 (not 8055), BG-LCV459 and BG-LCV460. This PMR added flexible metal hose and a shutoff valve on the valve stem packing leakoff lines of each of these valves.

Since these valves are subject to Reactor Coolant System (RCS) pressure and temperature, the flexible metal hoses can be pressurized to the same RCS pressure if the valve packing leaked and the leakoff isolation valves are closed. The area of concern identified by the SSOMI team in the subject PMR is the fact that the manufacturer's rated pressure for the flexible hoses is less than RCS pressure, therefore the flexible hoses are subject to failure.

RESPONSE TO THE FINDING:

Valves BB-8074A,B,C and D are 3 inch, manually operated, normally open, gate valves located in the RTD bypass loops. Valve BB-8085 is a 3 inch, manually operated, locked open gate valve located in the RCS letdown line (reference USAR Figure 5.1-1). Valves BG-LCV-459 and BG-LCV-460 are 3 inch air operated level control valves also in the RCS letdown line Valve BG-HV-8146 is a 3" air operated globe valve in the centrifugal pump discharge line to RCS loop 1 cold leg (reference USAR Figure 9.3-8).

These valves are on lines which have a design pressure of 2485 psig and a design temperature of 650 F. The normal operating pressure and temperature are approximately 2235 psig and 588OF. The flexible metal hoses selected for use are Swagelok models SS-6HO-6-L6 or SS-6HO-1-6-L6 with a SS 3/8" T X 3/8" NPT adaptor fitting.

The Swagelok catalog sheet for flexible metal hose connector identifies that the maximum working pressure rating at 70 F for these models of hoses is 1610 psig. Using the recommended derating factor of 0.74 at 600 F, the maximum working pressure is approximately 1190 psig. If the shutoff valve downstream of a given flexible metal hose is closed, the flexible metal hose would be subject to the same operating pressure(~ 2235 psig) and temperature (~ 588 F) as the main valves, assuming leakage of the primary packing and no leakage from the secondary packing to containment atmosphere.

WCNOC does not consider this item to be a deficiency. The basis for the WCNOC position is as follows:

1. The purpose of the shutoff valve is to preclude packing leakoff from other valves connected to the common header from back-flowing into the packing chamber of the valve during maintenance. The subject valves (BB-8074AD,BB-8085, BG-HV-8146, BG-LCV459 and BG-LCV460) have backseats. The maintenance manuals for these valves require that prior to working on the packing chamber, the packing chamber be isolated from the main pressure by the use of backseat. Therefore, during the maintenance of the subject valves when the leakoff shutoff valves are isolated the flexible metal hoses will not be subject to RCS pressure.

2. The subject valve packing leakoff connections tie into a 2" header which discharges to Reactor Coolant Drain Tank THB09 via line HB-056-HCD-3" (reference USAR Figure 11.2-1). The reactor Coolant Drain Tank has an internal design pressure of 100 psig and a design temperature of 250 F (reference USAR Table 11.2-1). Line HB-060-HCD-2" connected to line HB-056-HCD-3" is provided with a relief valve HB-7160 set at 25 psig.

   The disposition to EER 86-XX-46 stated that the shutoff valves on the valve leakoff connections are to be maintained normally in the open position. Therefore, during the normal plant operation when the shutoff valves are open the flexible metal hoses will not be subject to RCS pressure. In fact, during normal plant operation, the pressure in these flexible metal hoses is expected to be less than 25 psig.

3. It should be noted that the nominal burst pressure for these flexible metal hoses at 70°F is 6440 psig. Using a recommended derating factor of 0.74 at 60°F, the nominal burst pressure is 4765 psig (reference Swagelok Catalog Sheet). At the normal RCS pressure of 2235 psig, there is still a safety margin of ~ 2 available.

CONCLUSION:

Based on the above discussion, WCNOC does not consider this item to be a deficiency.

2.1.2.7 PMR 2206: AUXILIARY BUILDING FIRE DETECTION SYSTEM

FINDING:

This PMR revised the fire detection system in the Auxiliary Building and installed 3-hour fire resistant material on a hatch. The revisions were issued in response to additional combustible material loading as a result of the installation of a tool storage area and an anti-contamination clothing storage area in the basement of the Auxiliary Building. The increased combustible loading deviates from the commitments of the USAR, as indicated below.

a. USAR Page 9.4-2, indicated that Fixed Water Suppression Systems were installed in areas with a high fire or loss potential. No fixed system was installed in the areas in question.

b. USAR Table 9.5.1-2 stated that an automatic pre-action sprinkler system was installed to protect cable trays in the Auxiliary Building at elevation 1974'-0" and that vertical cable chases were protected with an automatic wet pipe system. Uncovered cable trays containing the power cables for the "A" and "B" Auxiliary Feed Pumps pass vertically through one of the areas in question with less than 20 feet separation between the combustible material and no sprinkler system provided.

c. USAR Table 9.5A-1 indicates that safety related systems are isolated or separated from combustible materials; the USAR also indicates that these systems are separated when practical. The installation of the two storage areas in close proximity to safety related systems violates this guideline.

The SSOMI team was concerned that combustible loading in the Auxiliary Building was increased without implementation of the commitments of the USAR.

GENERAL DISCUSSION OF FINDING:

PMR 2206 revised the fire detection system in the Auxiliary Building (Area 5) and installed 3-hour resistant material on a hatch that connected areas at elevations 1974' and 1988'. The revisions were issued in response to additional combustible material loading in the vicinity. Engineering Evaluation Request 87-21-04 requested an evaluation of the additional fire loading as the amount and location of these combustible materials were deemed necessary to support plant operation and maintenance activities. The SSOMI team was concerned that combustible loading in the Auxiliary Building was increased without implementation of the commitments of the USAR.

RESPONSE TO THE FINDING:

a. Page 9.5-2 of Rev. 0 of the USAR includes Power Generation Design Basis Four, which states that fixed water suppression systems are installed as required in areas with a high fire or loss potential. Further, criteria for determining the need for these systems is in substantial compliance

with the American Nuclear Insurers (ANI) "Basic Fire Protection for Nuclear Power Plants" (March 1976). It should be noted that this design basis is not a safety design basis.

The area in question, Rooms 1128 and 1129, maintain a relatively low fire or risk potential as the increase in the fire loading from the additional combustible materials does not substantially increase the risk or loss potential from an exposure fire. This conclusion was based, in part, on the fact that proper storage of combustibles (e.g.,NFPA-30 rated containers, metal shelving, and bins) and housekeeping controls would be maintained.

b. The subject area is relatively uncongested which promotes effective fire fighting ability. In general, sprinklers are located to provide corridor bay coverage where concentrations of cable trays occur. Typically, sprays are provided where two or more stacks of trays are present with two or more trays in each stack. Also considered are the congestion in the area and the height of the trays above the floor. In the case of the subject area, the cable tray concentration and loading is minimal, and as such does not warrant sprinkler coverage.

In Appendix 9.5B, Fire Hazards Analyses of Rev. 0 of the USAR, paragraph two of Section A.1.7.2 Safe Shutdown Capability addresses the issue of both motor driven auxiliary feedwater pumps' power cables being located in Room 1128. The principal rationale set forth in the analysis is that the turbine driven auxiliary feedwater pump is not affected by a fire in this area and will be available to bring the plant to a safe shutdown. Other supportive information is provided in USAR section A.1.7.2 (second paragraph), such as the distance between the cables being nineteen feet, no other safe shutdown equipment, trays, or exposed conduits are located in the room, traffic to other areas of the plant cannot pass through the room, and transient combustibles are limited to those required for pump maintenance.

c. The reference to USAR Table 9.5A-1 is taken out of context. The section that is being referred to is under Section D, General Guidelines for Plant Protection. The WCGS Summary of Compliance with Appendix A of NRC Branch Technical Position (BTP) APCSB 9.5-1 for item D.2.a is that safety related systems are isolated or separated from combustible materials, where practical. Where this is not practical, special protection is provided to prevent failure of both safe shutdown trains by a single fire. Also, reference is made to the Fire Hazards Analysis, Appendix 9.5B.

The need for fixed water suppression systems is considered not required for the subject plant modification. Since, the Turbine Driven Auxiliary Feedwater pump is unaffected by a fire in this area and the PMR provides additional fire protection afforded by the installation of 3-hour-rated fire resistant material to the floor hatch between Rooms 1207 and 1129 and the installation of fire detectors, therefore a fixed water suppression system is not warranted.

The establishment of a separate storage facility outside of the

Auxiliary Building was not considered practical. As discussed in Item b above, the turbine driven auxiliary feedwater pump is not affected by a fire in this area. The turbine driven auxiliary feedwater pump is considered to be the redundant safety related train for a fire in this area, and thus protection is afforded in the event of a single fire. The subject storage area is actually not in close proximity to the required turbine driven auxiliary feedwater pump train. The storage area is basically twice removed by two separate three-hour barriers from the turbine driven auxiliary feedwater pump train, and thus safe shutdown is assured.

CONCLUSION:

Although the initial placement of combustibles in this area was not fully in accordance with USAR discussions, the design modification meets or exceeds all USAR commitments. The PMR design ensures that the turbine driven auxiliary feedwater pump train is protected and isolated from a fire in this area. WCNOC does not consider this item to be a deficiency.

2.1.2.8 PMR 2222: CONTAINMENT COOLING FAN DAMAGE

FINDING:

This PMR implemented corrective actions following the failure of a fan blade in Containment Fan Cooler (CFC) SGN01B. The failure damaged the cooling unit and required entering a TS Limiting Condition for Operation (LCO) Action Statement. The cause of the failure was determined to be loosening of the nut which held the fan blade to the hub of the fan rotor. Subsequent inspection revealed four other loose nuts on the CFC SGN01B. In addition, the licensee determined that a similar failure which was attributed to loosening of the blade nuts had previously occurred at another facility. The licensee instituted an inspection procedure to verify the tightness of the nuts during each refueling outage, but did not investigate and determine the root cause of the failure. Possible causes for these failures are insufficient torque on the nuts to produce the required preload, excessive torque causing overstressing of the blade shafts, inadequate blade shaft size or strength and excessive vibration. The failure to investigate the root cause of these failures is an example of a general weakness in engineering evaluations.

GENERAL DISCUSSION OF FINDING:

In Item 2.1.2.8 the SSOMI team determined that the root cause investigation of the fan blade failure in Containment Cooler SGN01B was insufficient and that the failure to fully investigate the root cause of the event to be a general weakness in the engineering evaluation.

RESPONSE TO THE FINDING:

In the investigation performed of the fan blade failure, visual examinations of the blades and examinations of the blade threaded connections by the magnetic particle method were conducted. Due to the nature of the failure of the broken blade, which was broken transversely across the threaded root portion, and the fact that no cracks were found in like areas of the remaining fan blades, it was concluded that inherent material deficiencies or the existence of overstressed conditions were unlikely root causes. The nature of the blade failure and the subsequent damage incurred to the remainder of the fan made it difficult, if not impossible, to distinguish if the contact of the fan blade tip to the shroud housing occurred prior to or after the fracture of the blade root. This key aspect prevented determining whether the fan blade root nut loosened or deficient fan blade material was the root cause of the fan blade failure.

Based on the above and the isolated instance of a fan failure at another facility of similar design (Hydrogen Mixing Fan manufactured by Joy Manufacturing) from which very little information could be derived, it was concluded that the root cause of the event could not conclusively be determined. Effective prevention of another such type of failure by inspections of blade tip angles, torque checks or the blade attachment nuts, and lubrication of the motor bearings of other like vanaxial fans and continued monitoring was deemed most appropriate.

ACTIONS WHICH HAVE BEEN OR WILL BE TAKEN:

Lubrication of the motor bearings, inspection of the blade tip angles and torque checks of the blade attachment nuts is completed at each refueling outage as required by the Preventive Maintenance Data Base Program. Verification of acceptable vibration levels are also conducted in like fashion.

CONCLUSION:

Based on the above discussion, WCNOC does not consider this item to be a deficiency.

2.1.2.9 APPENDIX J LEAK TEST REQUIREMENTS

FINDING:

PMR's 1143 and 2109 required the encapsulation of hinge pins for feedwater valves due to leakage. The SSOMI team determined that 10 CFR 50, Appendix J required leak testing had not been performed on these valves. The licensee indicated that the Steam Generators and attached secondary systems inside containment were closed systems and that for all accident conditions, the secondary pressure was higher than containment pressure. Therefore leak testing of the valves was not required. The SSOMI team questioned whether the Steam Generator System was required to be maintained pressurized in the event of a design basis accident and whether the instruments, instrument lines and appurtenances on the Steam Generators and other secondary systems were adequately protected in the event of a high energy line break. Further action is necessary to clarify the containment boundary and leak testing requirements with respect to the secondary systems, the assumed condition of the secondary systems during a design basis accident, and the design of the secondary systems with respect to high energy line breaks. This general concern has been previously identified at other plants and is under review by NRR. There were no further concerns identified with respect to these PMRs.

GENERAL DISCUSSION OF FINDING:

During the review of PMRs 1143 and 2109, the SSOMI Team identified that 10CFR50, Appendix J required leak testing had not been performed on the associated valves. The team questioned the containment boundary and leak testing requirements with respect to the secondary systems, the assumed condition of the secondary systems during a design basis accident and the design of the secondary systems with respect to high energy line breaks.

RESPONSE TO THE FINDING:

The valves associated with the above identified PMR's are Feedwater Isolation Check Valves AE-V-120, AE-V-121, AE-V-122 and AE-V-123 and Feedwater Chemical Addition Isolation Valves AE-V-128, AE-V-129, AE-V-130 and AE-V-131. These valves are associated with the Main Feedwater Line Penetrations P5, P6, P7 and P8.

As discussed in Safety Evaluation Seven, in USAR Section 6.2.4.3, the containment penetrations associated with the steam generators are not subject to GDC-57, since the containment barrier integrity is not breached. The boundary or barrier against fission product leakage to the environment is the inside of the steam generator tubes, the outside of the steam generator shell, and the outside of the lines emanating from the steam generator shell side. USAR Figure 6.2.4-2 shows the steam generator and associated secondary systems that serve as a barrier to the release of radioactivity post-LOCA. USAR Figure 6.2.4-1, Sheets 5, 6, 7 and 8 show the configuration of the subject penetrations.

As indicated in USAR Section 6.2.6.3, Containment Isolation Valve Leakage Rate Tests (Type C tests), the valves associated with the piping systems connected to the secondary side of the steam generators isolate the steam generators and are not considered containment isolation valves and are, therefore, not leak

tested. All portions of the secondary side of the steam generators are considered an extension of the containment. The above identified USAR Section further elaborates and specifically excludes penetrations P5, P6, P7 and P8 from Type C testing. As shown on USAR Figure 6.2.4-2, the water level in all steam generators is maintained above the tubes following a LOCA to preclude the entrance of the containment atmosphere into the secondary side of the steam generators. As discussed in USAR Section 6.2.6.1.1, following the containment stabilization period of the Type A Test (ILRT), the secondary side of the steam generators are vented outside of the containment to ensure the most conservative test configuration during the ILRT.

The high energy line pipe break locations and types of breaks are identified in USAR Figures 3.6-1 and 3.6-3. The stress results that were utilized to determine the break types and locations are provided in USAR Table 3.6-3. The high-energy pipe break effects analyses criteria and results are provided in USAR Table 3.6-4. The results of the analysis demonstrate that the for all postulated breaks, including the design basis LOCA, containment integrity is maintained and no essential systems are impacted.

CONCLUSION:

As discussed above, the subject valves and associated penetrations have previously been identified in the USAR as being exempt from local leak rate testing. In addition, their configuration and compliance with 10CFR50, Appendix J requirements, relative to Type A Testing, (ILRT) have also been previously provided in the above identified USAR Sections.

The high energy line break evaluation and supportive existing USAR information has been identified for clarification as requested. The Safety Evaluation Reports, NUREG-0830 and NUREG-0881, describe, in Sections 3.6.1, 3.6.2, 6.2.3, and 6.2.5, the staffs recognition and acceptance of the above identified information.

2.1.2.10 BATTERY DISCHARGE OF OCTOBER 15, 1987

FINDING:

During maintenance of the Division B Vital Bus NB02 on October 15, 1987, both Station Batteries were subjected to a deep discharge resulting in a loss of both vital buses. The root causes of the discharge are being reviewed by Region IV and will be addressed in separate correspondence. As part of this inspection, the SSOMI team reviewed the adequacy of the battery capacity and associated design calculations.

The maintenance which deenergized the bus was expected to last less than 30 hours. Based upon the nominal ampere-hour capacity of battery NK14, Operations personnel performed a calculation and estimated that the battery would provide 50 hours of service with a 35 ampere load. The SSOMI team found that the basis for the estimate that the batteries would provide 50 hours of service was inadequate. The licensee indicated that the estimate was made by dividing the battery rating of 1650 ampere-hours by 50 hours which yielded a permissible discharge rate of 35 amps However, the actual load on the battery during the discharge was recorded in the control room each shift as 70 amperes. Battery Sizing Calculation E3, indicates that steady state loads of 220 and 100 amperes would result in discharge rates of 20 amperes per positive plate (APPP) and 16 APPP for Batteries NK12 and NK14 respectively. The battery cell characteristic curve indicates that the battery capacity would provide 10-hours of service at these discharge rates. Based on Calculation E-3 and the cell characteristic curve, the SSOMI team calculated that a 70 ampere load would have resulted in a discharge time of 35 hours.

In addition, the SSOMI team noted that during removal of the NB02 Vital Bus from service, system operating procedures SYS NB331/4 and SYS NB331/5 were not utilized. These procedures specified the requirements and precautions for system operation and isolation and specified a maximum discharge time of 200 minutes. The licensee subsequently indicated that operational procedures are not routinely used for removing and returning equipment to service. The failure to utilize the appropriate operational procedures and incorporate the precautions and requirements of these procedures for the removal and return of equipment from service in accordance with the requirements of the Technical Specifications and 10 CFR 50, Appendix A, is considered to be a weakness. The SSOMI team considers that, if the Operations Department had complied with the procedural requirements, or had used the battery sizing calculations provided by Engineering, or had consulted with Engineering, the deep discharge and resulting loss of both vital buses would not have occurred.

The battery discharges occurred on October 15, 1987. Following these events, the batteries' conditions were not recorded prior to recharging. Based upon the results of the earlier October 7, 1987, performance discharge test on battery NK12 discussed in Section 2.1.2.12 of this report, and the fact that the unplanned discharges went deeper than the performance test, at least Cell 32, and probably other cells, reversed polarity during the unplanned discharges.

The failure to provide adequate controls for the removal from service of a safety system which resulted in a loss of the Station Batteries is a weakness.

In addition, the failure to involve the Engineering Department, either through the use of system procedures or by consultation during the battery discharges, is a significant weakness.

GENERAL DISCUSSION OF FINDING:

Station procedures were not utilized to remove Vital Bus NB02 from service and there was a failure to provide temporary power to Station Batteries NK12 and NK14. Additionally, engineering was not utilized to perform discharge calculations.

RESPONSE TO THE FINDING:

The root cause of these events has been attributed to cognitive personnel error by Operations, Maintenance and Outage management personnel in failing to plan for the capability to provide temporary power supplies to the batteries if the NB02 outage was extended. Preparations to supply the batteries with alternate power were in progress when the Low Voltage Alarms were received, however, the task was not completed prior to discharge of the batteries. This situation existed because prior to this event, station policy was to utilize work request procedures and tag out procedures to control plant equipment during outage work. Also, implementation of the temporary modification to bring in alternate power to the bus was too time consuming.

As a result of the battery discharges, an evaluation of the batteries was performed. This evaluation concluded that all of the batteries were capable of performing their design function based on a review of the specific gravity, voltage, electrolyte level, and electrolyte temperature for each cell.

ACTIONS WHICH HAVE BEEN OR WILL BE TAKEN:

Programmatic changes have been incorporated which require either the use of existing procedures or development of a new procedure to take equipment out of service and to restore equipment to service. Prior to this time, the work request procedure and the clearance order procedure were used to control these work activities.

By September 1, 1988, a specific procedure will be prepared which will provide instructions for deenergizing a vital bus with requirements for powering Station Batteries from an alternate source. The preparation of this procedure and its use will preclude future similar occurrences.

Plant and Nuclear Plant Engineering procedures have been changed to enhance the use of engineering talent and skills. These procedure modifications changes should result in proper personnel involvement in a specific problem area.

The events described above were addressed in Programmatic Deficiency Report (PDR) OP-87-81 and Licensee Event Report (LER) 87-049. The implementation of the corrective action will resolve the concern identified by the SSOMI team.

2.1.2.11 BATTERY SIZING CALCULATION E-3

FINDING:

Bechtel Calculation E-3, "Class 1E Battery System," Rev. 0, dated January 12, 1987, was reviewed by the SSOMI team and the following discrepancies were identified:

a. The calculation assumed a constant load on the DC System based upon a full load rating of the 7.5 KVA for the Battery Inverter. However, in calculating the DC current, the DC voltage was taken at the nominal 125 VDC level. Because inverters try to deliver a constant KVA load, as the battery voltage decreases during a discharge the current drawn by the inverter from the battery will increase. These errors would add approximately 10% to 15% to the required battery size.

b. The input data did not document the cell characteristics used by the computerized calculation.

c. The calculation did not consider the minimum cell temperature permitted by the Technical Specification.

The battery sizes selected for batteries NK11 and NK13 are 25% and 42% larger than the calculated required size. The errors noted above can be enveloped by the existing battery and there are no safety concerns with the installed size of the batteries. The inadequacies identified in the battery sizing calculation are symptomatic of a general weakness noted by the team in engineering calculations.

GENERAL DISCUSSION OF FINDING:

Station batteries NK12 and NK14 were subjected to a deep discharge due to extended loss of power to their chargers. As a part of their investigation, the SSOMI Team examined the adequacy of the battery capacity and associated Bechtel design Calculation E-3, Rev. 0, dated January 12, 1981. (The subject SSOMI report refers to Bechtel Calculation E-3, Rev. 0, dated January 12, 1987. This is in error. The referenced calculation revision is dated January 12, 1981).

RESPONSE TO THE FINDING:

a. The load profile used in calculation E-3 included continuous and momentary loads that are conservatively assumed to be constant during the design basis duty cycle. Since the total load on the subject batteries consists of mostly constant resistance types of loads (e.g., control panel indication lights, control circuits, instrumentation and emergency lights), the above assumption is considered conservative as the resistive load currents decrease under low voltage conditions.

   It should also be noted that, as verified by Westinghouse, the inverter vendor, the subject inverter draws 58 amps at 135VDC, approximately 62 amps at 125VDC and 70 amps at 105VDC. Therefore, the use of 68 amps as a steady state load over the entire 200 minute profile is conservative and acceptable since this value approximately equals the maximum current draw

which occurs only at 105VDC, for a short duration near the extreme end of the discharge profile.

b. The computer program software was itself an independent and controlled standard program of the Architect Engineer which contains the subject data for the batteries used at Wolf Creek Generating Station. The standard computer program inherently contains the applicable battery cell characteristics, therefore these characteristics are not provided as user entered data every time the computer program is used. It is indicated on page 10, of the subject calculation, that the correct battery type and manufacturer was, in fact, selected and recorded on the calculation.

c. The capacities of the subject batteries have been evaluated and previously discussed with the NRC (Safety Evaluation Reports, NUREG-0881 (Wolf Creek), April. 1982 and NUREG-0830, October, 1981 (Callaway), Section 8.3.2.2) with regard to temperature performance and other factors. The subject batteries are sized in excess of 50 percent of the initial battery capacity (i.e., are 50% oversized) thus enveloping the requirements for temperature, voltage and specific gravity fluctuation, as well as the replacement criterion of 80 percent. The subject batteries meet IEEE Standards for battery sizing and are acceptable. This item was specifically discussed with the SSOMI team in order to document that minimum cell temperatures were evaluated by WCNOC prior to fuel load for Cycle III.

CONCLUSION:

The battery sizes installed are adequately designed and provide more than the required capacity. It is not felt that the above observations are indicative or representative of a general weakness in engineering calculations.

2.1.2.12 BATTERY PERFORMANCE TEST

FINDING:

In order to evaluate the battery discharge of October 15, 1987, the SSOMI team reviewed the performance test performed for Battery NK12. The purpose of the performance test was to demonstrate the actual capacity of the battery compared to the manufacturer's published rating.

Performance Test STS MS 022, performed on Battery NK12 on October 7, 1987, was interrupted prior to the completion of the test to jumper out Battery Cell No. 32. This cell had dropped 80 millivolts in 10 minutes to 1.692 volts. The SSOMI team estimated that the battery would have reached its test limit in less than 1/2 hour if the cell had not been jumped. The test was restarted approximately 6 1/2 hours later. When the test was restarted the battery voltage had increased to a voltage which existed two hours before the test had been stopped. As a result of the failure to recognize that the battery would recover lost capacity during the extended outage, the performance test incorrectly showed that the battery had a capacity of 138.75%. The SSOMI team calculated that if the performance test had been left to continue until completion, the battery capacity would have been calculated at 125%. While the battery has less capacity than the assumed by the licensee as a result of the performance test, the battery has 50% more capacity than that required by the test acceptance criteria. The failure to adequately test and evaluate the results of the battery performance testing is considered to be indicative of a general weakness in engineering evaluations.

GENERAL DISCUSSION OF FINDING:

This item concerns the jumpering of a weak cell (#32) during the October 2, 1987 performance discharge test of battery NK12. The finding concerns the time taken to jumper the weak cell, approximately 6-1/2 hours, and the battery capacity obtained from the test.

The procedure used in performance of this test, STS MT-022, was written to incorporate the requirements in the Technical Specifications, and the recommendations in the Gould Manual and IEEE 450. Step 6.4(4) of IEEE 450 states "if an individual cell is approaching reversal of its polarity (plus 1 volt or less), but the terminal voltage has not yet reached its test limit, the test should continue with a jumper across the weak cell".

RESPONSE TO THE FINDING:

There are two reasons which can be attributed to the performance test being stopped for approximately 6-1/2 hours. The procedure did not specify a maximum allowable time for interrupting the test. In addition, although jumper cables had been prepared prior to the test, the availability of the jumper cables was not checked immediately prior to the start of the test. Maintenance personnel did not recognize the time sensitivity of this while new jumper cables were being made.

The Gould Manual does not address performance or duty cycle testing. IEEE 450 lists recommendations for the performance and duty cycle tests, but does not identify the maximum allowable time for interrupting a test. As

addressed in IEEE 450 Step 6.4(4), the jumper could be placed directly across the cell, which would short the cell and completely discharge it, or the test could be stopped and that cell disconnected from the battery replacing it with a jumper. Since IEEE 450 states a cell is approaching reversal at plus 1 volt or less, a determination was made when writing the procedure to disconnect the cell from the battery to keep it from being completely discharged.

Gould was contacted after the test was performed as to their recommendation for jumpering a cell, and a maximum test interruption time. Gould indicated they would accept either method for jumpering a cell, and if we choose to jumper a cell that a test interruption of approximately 15 minutes should not effect the battery capacity value obtained from the test.

ACTIONS WHICH HAVE BEEN OR WILL BE TAKEN:

Procedure STS MT-022 will be revised to incorporate the Gould recommendation of 15 minutes, as the maximum time allowed to interrupt the test. Due to this 15 minute time limit, another step will be added to the STS to have the jumper cables cleaned and ready to install prior to starting the test. These changes will be completed prior to July 1, 1988.

The battery capacity, listed in Step 5.2.5 of STS MT-022 performed on October 2, 1987 will be corrected to show a capacity of 120.7%. This value was the proven capacity at the time the test was stopped. A Fluke meter had been installed to monitor cell #32, and Maintenance Engineering was present during the test interruption. The voltage of cell #32 dropped quickly, with only a few minutes between the 1.692 volt reading and the 1.080 volt reading when the test breaker was opened. If the test had continued after cell #32 was jumpered (in the recommended 15 minutes), the test would have continued for a very short period before it would have been stopped again to jumper cells #38 and #50.

CONCLUSION:

A revision will be made to the applicable procedure by July 1, 1988 to incorporate the resolution of the finding and prevent recurrence. Training for personnel in conducting and evaluating battery capacity testing will be developed and implemented for personnel prior to the next performance of STS MT-022.

2.1.2.13 DC SYSTEM LOW VOLTAGE ALARMS

FINDING:

In order to evaluate the battery discharge of October 15, 1987, the SSOMI team reviewed the alarms associated with the DC System to ensure that the design of the alarm setpoints were adequate. A review of the DC System alarms detailed on Relay Setting Drawing E-11028(Q) indicated that four DC undervoltage alarms existed in each Battery System. These alarms should have provided sufficient warning to prevent the DC System low voltages experienced on October 15, 1987.

GENERAL DISCUSSION OF FINDING:

Station procedures were not utilized to remove Vital Buss NB02 from service and there was a failure to provide temporary power to Station Batteries NK12 and NK14.

RESPONSE TO THE FINDING:

There are four DC undervoltage alarms for each battery system. Each battery system consists of a 125 VDC battery bank, a battery charger and associated equipment. The charger DC undervoltage alarm indicates the loss of a battery charger at 123 VDC. The DC switchboard bus undervoltage alarm indicates low DC voltage at 112.5 VDC. The distribution switchboard undervoltage alarm indicates a loss of the DC distribution panel at approximately 86 VDC. Each of these alarms displays in the control room through an NK system trouble window which will reflash for each alarm. The inverter DC undervoltage alarm indicates the loss of DC input to the inverter at 82.5 VDC. This alarm displays in the control room through an inverter undervoltage window. These alarms provide the control room with indication of a loss of DC system function, but are not generally anticipatory in nature. During the October 15, 1987 event, undervoltage alarms came in at approximately the same time as the initial Engineered Safety Features Actuations (i.e., Control Room Ventilation Isolation Signal, Containment Purge Isolation Signal and Fuel Building Isolation Signal).

ACTIONS WHICH HAVE BEEN OR WILL BE TAKEN:

By September 1, 1988, a procedure will be prepared which will provide instructions for deenergizing a electrical safety related division with requirements for powering Station Batteries from an alternate source.

CONCLUSION:

Implementation of the corrective action will resolve the concern identified by the SSOMI team.

2.1.2.14 DIESEL GENERATOR BREAKER OPERATION

FINDING:

The SSOMI team reviewed the design of the closing circuit for the Emergency Diesel Generator (EDG) output breakers, NB0111 and NB0211, and concluded that the design is inadequate, in that the "anti-pumping" logic prevents the breakers from closing onto a cleared, deenergized bus. In the event that both the normal and alternate 4160 VAC Vital Bus feeder breakers, NB0109 and NB0112, are open and the EDG mode switch is in the "auto" position, the "anti-pumping" relay will be energized continuously, thus preventing the EDG output breaker from closing. Operator action is required to cycle the EDG mode switch at the local station in order to clear the "anti-pumping" logic and close the EDG output breaker to reenergize the 4160 VAC Vital Bus.

The requirement for manual operator action appears to represent an unanalyzed condition, in that the deenergized 4160 VAC Vital Bus cannot be energized automatically by the EDG in this condition. This design deficiency was discussed in a conference call between the licensee and NRC Region IV management on December 8, 1987, and will be reviewed by NRC Region IV. The inability of the EDG output breaker to automatically close onto a cleared, deenergized bus is considered to be a design weakness.

GENERAL DISCUSSION OF FINDING:

The SSOMI team review of the design of the closing circuit for the Emergency Diesel Generator (EDG) output breakers concluded that the "anti-pumping" logic of the diesel breaker prevents the breaker from reclosing onto a deenergized bus after a manual trip is initiated from the Main Control Room. Operator action is required to cycle the EDG mode switch at the local station in order to reset the "anti-pumping" logic and permit reclosure of the EDG output breaker. The inability of the EDG output breaker to directly reclose on to a cleared, deenergized bus was considered by SSOMI to be a design weakness.

RESPONSE TO THE FINDING:

The above concern is related to the reclosing feature (manual or auto) of the subject breaker after a manual trip when offsite power is not available and when power is being supplied by the diesel generator prior to the subject manual trip. It was found that the stated condition resulted from manual interruption, via the Control Room breaker hand switch, of the as designed fully automatic sequence for diesel start, loading and restoration of essential AC power under loss of offsite power conditions.

As originally designed, the diesel generator is normally in the standby mode whereby, on loss of offsite power, it will automatically start and its output breaker will automatically close on to a cleared, deenergized bus, consistent with its design intent. No manual actuation or intervention is required for performance of this function. Tripping the output breaker and subsequently attempting to manually reclose it from the Main Control Room was not the original design intent, nor is it intended under future designs, that such interruption be entertained under the stated loss of offsite power conditions. A meeting was held with NRC, WCNOC and representatives of Bechtel (WCGS A/E) to discuss the existing design and operation of the breaker. The relationship

of the design to Regulatory requirements and considerations for modification to the existing design were also discussed.

The design was in compliance with relevant Regulatory requirements and that the existing design operates correctly under the relevant normal and design basis accident scenarios. However, after discussion with the NRC, it was concluded that it would be beneficial to have suitable provisions to allow reset of the circuit and allow resultant breaker reclosure from the Main Control Room. This capability would allow for more timely recovery should intentional or inadvertent manual trip of the diesel output breaker occur under loss of offsite power conditions.

ACTIONS WHICH HAVE BEEN OR WILL BE TAKEN:

Based on review of the subject control circuit and its design features, it was concluded that the subject diesel generator breaker was not designed to be tripped and reclosed manually from the Main Control Room under the stated conditions. The entire diesel generator start and loading sequence (including breaker closure) and power restoration is designed for automatic operation without manual intervention. However, upon subsequent discussion with the NRC, it was agreed that a design change providing the operator, in the Main Control Room, with a capability to reclose the diesel generator breaker on a deenergized bus would be a beneficial enhancement. Wolf Creek has procured the necessary materials and is ready to implement the modification.

CONCLUSION:

The proposed design change will enhance the capability for manual control from the Main Control Room to reclose the diesel generator breaker if manually tripped during a loss of offsite power event.

3.1.2.2 PMR 1722: VALVE MOTOR-OPERATOR TESTING

FINDING:

This PMR performed valve motor operator testing and torque and limit switch settings in response to NRC Inspection and Enforcement (I&E) Bulletin 85-03, regarding torque and limit switch settings in valve motor operators. In addition internal operator wiring was modified to distribute the limit switch and indication contacts to different rotors.

The PMR, PMR changes, drawings, calculations, and WR packages for several affected valves were reviewed. Operators for the five motor operated valves (MOVs) identified below were inspected for conformance to design requirements and proper workmanship.

Valve Number	Function
BB-HV8000A	Pressurizer Block Valve
BB-HV8000B	Pressurizer Block Valve
AL-HV031	ESW to Motor Driven AFW Pump A
AL-HV030	ESW to Motor Driven AFW Pump B
EM-HV8923B	SI Pump B Suction Isolation

The following discrepancies were identified:

a. Deficiencies were identified in two of the valve operators. The spare conductors of the control cable in each Pressurizer block valve, which are located inside containment, were not properly protected to prevent possible interference with the operation of the valve. Bechtel Detail Drawing E-11013, required spare conductors of safety related cables in the Reactor Building to be protected with Raychem end caps. Contrary to this requirement, the spare conductor in Valve BB-HV8000A was unprotected, and the spare conductor in Valve BB-HV8000B had been originally protected with tape which subsequently became unraveled. Drawing E-11013 allows the use of tape for protection of conductors outside, but not inside, the Reactor Building. In addition, insulation damage was noted on several conductors of the control cable in Valve BB-HV8000A. Based on these findings, WR 4954-87 was written for Valve BB-HV8000A and WR 4953-87 was written for Valve BB-HV8000B to correct the deficiencies and to evaluate the insulation damage.

b. A concern was identified in the complexity of drawings required to identify the wiring configuration for conduct and verification of maintenance and modifications of the motor operated valves (MOVs). As many as seven different drawings and wiring lists were required to fully inspect the wiring configuration of each MOV reviewed. For example, the drawings required to inspect Valve AL-HV030 included a design schematic, a vendor wiring diagram which also included a partial schematic, two field wiring lists, and three design change documents. Other valves had wiring changes required by the vendor that were specified in writing by vendor change requests, but were not indicated on the vendor wiring drawings. Differing conventions for identifying the wire termination points on wiring diagrams were also identified. In some instances, wiring diagrams identified field wire terminations, and in others the internal wire designations duplicated field wire designations.

The difficulty in using the large number of inconsistent drawings to make changes to wiring had been previously identified by the station Electrical Maintenance Department. Engineering Evaluation Request (EER) 86-EM-03 described the problem of multiple drawings and inadequate cross referencing between plant and vendor drawings. The SSOMI team was concerned that, although the EER was approved and submitted on July 18, 1986, an evaluation or disposition for the EER had not been made.

EER 87-KC-08 was written to correct a vendor wiring drawing that used the same wire numbers twice. The engineering evaluation rejected the recommended drawing change on the basis that a review of four related design drawings indicated the wiring was inadequate. The EER response referenced Bechtel Specification F-01016, "Electrical As-Built Drawing Criteria," Note N, as the reason for not correcting the wiring drawing. Note N states that internal vendor wiring inconsistencies which do not affect the circuit electrically or functionally will not be incorporated into the as-built drawings. The SSOMI team considered that this resolution was inadequate because as-built wiring drawings are routinely used for maintenance and modification activities.

Except for the discrepancies in the Pressurizer Relief Valves discussed in Section 3.1.2.2.a, the workmanship observed was good. The fact that few problems have been identified in motor operators is attributed to the diligence of the maintenance craft and engineers and is considered a strength.

GENERAL DISCUSSION OF FINDING:

a. During field walkdown with the inspectors to check the actuator wiring installed in accordance with PMR 01842 (not 01722 as referenced), it was found that spare conductors were not isolated correctly. It was also noted that several conductors of cable 11BBG39AG were abraded and needed to be checked for insulation damage within valve actuator BB-HV8000A.

b. EER 87-KC-08 identified an anomaly on a vendor wiring diagram, involving the duplication of vendor wire numbers. It was recommended that one set of the duplicated wire numbers be revised to preclude confusion.

The SSOMI team concern dealt with what was termed "the complexity of drawings required to identify the wiring configuration for conduct and verification of maintenance and modifications of the motor operated valves (MOVs)". The design documents used on PMR 1722 were of the standard format and consisted of:

a) Schematic drawings (E-03's) - drawings used to show the valve operator's internal electrical configuration and its interface with other systems and components.

b) Internal wiring diagrams - vendor supplied internal wiring configuration for each valve. Represents vendor provided or vendor side wiring.

c) External wiring diagrams - drawings or listed information that shows the configuration of external cables, their associated wires and termination points. Wires installed external to the equipment which are needed to fit or configure that equipment into the overall scheme are considered as field wires or jumpers. The termination list and jumper list are included in this category.

When any electrical modification is made, the possibility exists that at least one, and in many cases all three of the above documents will be revised by either issuing a document revision on a document change notice. Due to time constraints, a needed revision of the modification documents associated with valve AL-HV030 was accomplished by issuing three document change notices.

RESPONSE TO THE FINDING:

a. The use of tape within the Reactor Building is not a design approved means to isolate or protect spare conductors. The root cause is recognized to be a lack of attention to the specific details by electricians, QC, and Maintenance Engineering personnel. The abrasion on conductors was due to original installation methods.

b. The general description, specifications and details for vendor equipment are provided in vendor instruction manuals. It is necessary that this information be reviewed prior to routine maintenance or modification activities.

The instruction manual for the equipment shown on the subject vendor wiring diagram includes information that eliminates the confusion regarding the duplicate wire numbers. Therefore, note N on Bechtel Drawing E-01016, "Electrical As-Built Drawing Criteria", applies to the identified anomaly and no drawing revision is required. Therefore, the disposition is adequate and WCNOC does not consider this item to be a deficiency.

The electrical design documents used for plant modifications are based on the system utilized by the Architect/Engineer (A/E) for the original plant design. Each document type is used to convey its portion of the total picture. Once an individual becomes familiar with the scheme used, the system is not "difficult". With regard to what the inspector described as "inconsistent drawings", it must be recognized that the above three document types were consistent with respect to one another for any particular operator. However, due to the fact that each vendor's internal wiring diagrams are different and site installation characteristics for each operator may be different (e.g. slack in field run cables,, it is possible that the set of electrical design documents for a particular operator may be different (or "not consistent") with those of another operator.

With regard to the SSOMI team's comments on dispositioning Engineering Evaluation Request (EER) 86-EM-03, Nuclear Plant Engineering is addressing the concerns reflected in the EER as part of the work scope of PMR's 1722, 1842, 2071, 2073 and 2076.

ACTIONS WHICH HAVE BEEN OR WILL BE TAKEN:

a. Work Request #4953-87 installed an appropriate end cap on the spare conductors in BB-HV8000B. Work Request #4954-87 installed approved end caps on the spares in valve BB-HV8000A. Work Request #4954-87 also reworked the abraded conductor insulation in accordance with the criteria defined in E-11013.

   Maintenance Engineers were reminded of the Raychem end cap criteria for spare conductors within the containment, and measures were taken to prevent further conductor degradation by installing protective sleeving on the abraded conductors.

CONCLUSION:

a. This item does not point to a significant program breakdown. The improper isolation of spare conductors is not a major defect. The abrasion noted is a result of previous incorrect installation and the problems were resolved expediently when identified.

b. WCNOC realizes that the electrical design document format may be somewhat confusing to an individual who is unfamiliar with the A/E's system. However, the system does work and provides the information needed to support plant operations. Although a "standard" approach is desirable and is used where possible, differences in vendor drawings and field installation configurations do cause variations in the method used to achieve the same end results.

3.1.2.4 PMR 2018: ASCO SOLENOID VALVE REPLACEMENT

FINDING:

This PMR replaced twelve safety related, seismic Category I, environmentally qualified, Asco solenoid valves on air operated valves in the Reactor Coolant System, Chemical and Volume Control System, Residual Heat Removal System, High Pressure Core Injection System, and Liquid Radiological Waste System. The modification was initiated as a result of shorting of electrical wiring within the solenoid housing caused by limited space within the housing, splicing of pigtail leads, and Raychem sleeving of wires with damaged insulation. The PMR replaced solenoid valves using electrical pigtails with solenoid valves using terminals.

The PMR, WRs, procedures and documentation associated with the installation were reviewed. A field inspection was conducted on three solenoid installations that had been completed at the time of the inspection. The installations were inspected against the assembly and mounting detail drawings and work instructions in WRs that implemented the PMR. Solenoid valve serial numbers were verified against material requirements for the valves that were inspected. The following concerns were identified:

a. The 10 CFR 50.59 Safety Evaluation was inadequate, in that the seismic and environmental equivalency of the solenoid valves being installed was not documented. The written evaluation described only the improvement in safety because the leads would not short out in the new model solenoids.

   The safety evaluation for changes involving substitution of components would normally be expected to reference equivalency or superiority in form, fit, function, materials, mounting, and qualification, which would equate the change to a quality level at least consistent with the originally analyzed design.

b. The air supply line for the ASCO solenoid valve EJ-HCV8890B had inadequate seismic support between the solenoid valve and the polar crane wall. The supply line had approximately eight feet of rigid and hard copper tubing that did not have support. The unsupported tubing contained both an unsupported drain valve and the solenoid air isolation valve.

c. The 3/8-inch diameter stainless steel air tubing for the Asco solenoid valve serving Valve EJ-HCV8890B had one loose tubing support between the Asco solenoid valve and the air operator.

d. Work Request 1042-87 which replaced Asco solenoid Valve EM-HV8881, had one instance of unclear work instructions. Step 26 required recording the valve serial number without reference to which valve. Consultation with the personnel who had written the work instructions and performed the work was required to ascertain that the serial number recorded was for the valve which had been removed. In this case, the unclear instructions did not affect safety.

GENERAL DISCUSSION OF FINDING:

a. SSOMI team review of PMR 02018 Asco Solenoid Valve Replacement identified a concern with the 10CFR50.59 Safety Evaluation. The identified concern is an isolated case that was an omission in the preparation of the 10CFR50.59 Safety Evaluation because the solenoids utilized in the design were procured earlier as replacement parts to the appropriate technical and quality requirements.

b. The air supply line for ASCO Solenoid Valve EJ-HCV-8890B appeared to be installed with inadequate seismic supports. Approximately eight feet of rigid and hard copper tubing were not seismically supported, as well as a drain valve and the solenoid air isolation valve.

c. During field walkdown of valve EJ-HCV8890B a loose 3/8" air line was identified.

d. Work Request #1042-87 written for valve EM-HV8881 contained a confusing step for the recording of a valve serial number.

RESPONSE TO THE FINDING:

a. The new model solenoids were originally procured as replacements for the solenoid valves. The spare parts had been purchased in accordance with the original Westinghouse Purchase Order which included seismic and environmental qualification requirements. In addition, the Material Received Report (MRR) includes a certificate of conformance documenting the compliance with Westinghouse Purchase Order and the applicable Codes and Standards. Based on this discussion, the seismic and environmental equivalency of the replacement parts were considered during the procurement process; therefore, no hardware deficiency has been identified. However, reference to this equivalency should have been documented on the 10CFR50.59 Safety Evaluation.

b. The Instrument Air System at WCGS, of which the subject tubing is a part, is non-safety related and non-seismic. The copper tubing in question is designed to ANSI B31.1, and the branch line feeding the solenoid on Valve EJ-HCV-8890B is field routed and supported in accordance with Specification 10466-M-205, Appendix X. In the event of the failure of this air supply tubing, the valve will return to its fail-safe mode.

   From the standpoint of II/I considerations the subject tubing was reexamined by an engineering walkdown subsequent to the SSOMI Review. The walkdown confirmed that no adverse II/I condition exists with the subject installation and no II/I considerations need be made for the subject tubing installation.

c. The cause of the loose air line support is unknown.

d. The root cause of the confusing work instruction on valve EM-HV8881 was the lack of adequate review of work instructions by the maintenance lead electrical engineer.

ACTIONS WHICH HAVE BEEN OR WILL BE TAKEN:

c. The loose air line support was tightened in accordance with Work Request #4802-87.

d. Step 36 was an information only item for maintenance. It was not a requirement of the installation, therefore no corrective action to correct the intent of the step was taken. Work instructions are currently being reviewed in a much more attentive manner prior to issue in attempt to minimize confusing items in the future.

CONCLUSION:

a. WCNOC acknowledges that the 10CFR50.59 Safety Evaluation for PMR 2018, Asco Solenoid Valve Replacement should have identified the reference to equivalency and suitability of the replacement solenoids. In order to preclude further omissions of this nature, the subject deficiency will be brought to the attention of those personnel performing safety evaluations through review of the SSOMI team audit findings and the WCNOC response. No further action is warranted at this time.

b. As discussed above, the subject installation does not require additional seismic or II/I consideration, and as such WCNOC does not consider this item to be a deficiency.

c. The identification of the loose air line support was corrected on WR #4802-87.

d. WCNOC acknowledges that confusing work instructions existed. Current plans for more detailed reviews, word processing capability, and the normal recognition of experience will minimize these type of problems. These improvements will be implemented by September 1, 1988.

3.1.2.5 PMR 2329: RAYCHEM SPLICES

FINDING:

This PMR did not require field work and was used to document the disposition of 38 deficient Raychem splices identified in WR 4443-87 that were to be dispositioned 'use-as-is'. The Raychem splices were not installed in accordance with the Raychem instructions in that overlaps were less than the required two inch minimum and bend radii were less than the required five times the outside diameter. The PMR had been approved based on Wyle Nuclear Environmental Qualification Test Report No. 17859-02P, Revision A. The Wyle test report qualified seven Byron and Braidwood Generating Station specific configurations and thirteen Zion Generating Station specific configurations with overlaps as little as 1/2 inch and bend radii of 1.2 times the outside diameter.

The PMR, the WR which documented the 38 splices and the Wyle test report were reviewed. In addition, the Nuclear Plant Engineering (NPE) personnel who conducted the engineering evaluation and the Instrument Maintenance (IM) personnel who had participated in the original walkdown of the splices were interviewed. The following concerns were identified:

a. The PMR did not contain an evaluation or documentation which indicated that the WCGS design LOCA environment is equivalent to or less than severe than the design LOCA environment that was used as a basis for the Wyle testing. Evaluation and documentation of the LOCA conditions is required to establish a basis for use of the Wyle test report at WCGS.

b. WR 4443-87 documented the recommended disposition of 'use-as-is' for the 38 splices. The WR listed the splices and stated that all splices, as a group, had seal lengths of greater than 1/2 inch but less than two inches, and that the minimum bend radius was less than five times the shrink tube outer diameter. The WR did not document the configuration of each individual splice or provide measured bend radii or overlaps. The EER disposition that accepted the splices for 'use-as-is,' based on the Wyle test report, stated that the splices with bend radii as little as 1.2 times the outside diameter of the splice were tested. It also stated that the tested splices had bend radii more severe than the splices identified at WCGS. However, documentation that each splice has a bend radius greater than the bend radius tested by Wyle (stated as 1.2 times the outside diameter) was not available.

   The Wyle test report covered tests of twenty Raychem splices with different configurations, tubing sizes and wire types. Only one splice tested was of a configuration similar to the 38 deficient splices at WCGS. The SSOMI team considered that the confirmation of each of the deficient splices at WCGS should be documented to confirm that the results of the Wyle tests are applicable.

c. The Wyle test report documented that the Raychem tubing was bent while it was heated. The WCGS splices were performed in accordance with vendor instructions which allowed the splices to cool before bending. Bending the cooled Raychem tubing is less conservative because of the reduced pliability of the tubing at lower temperatures.

During a previous NRC Environmental Qualification (EQ) team inspection, questions were raised regarding the lack of documentation for sizes and lengths of machine screws used in Raychem splices. The maximum working diameter of Raychem tubing would be exceeded if the machine screw used to connect the terminal lugs were too long. Following the EQ inspection, an Engineering Evaluation Request was initiated by the licensee to establish the acceptance criteria for conducting field measurements of tubing diameter to insure that maximum Raychem working diameters were not exceeded. At the time of the SSOMI inspection, WR 4943-87 and other similar WRs were in process to inspect all suspect splices. The SSOMI team concluded that the WR appeared to adequately verify and correct Raychem problems associated with exceeding the maximum working diameter, but did not address the concerns discussed above regarding the applicability of the Wyle test report. The failure to adequately document and evaluate the engineering disposition of the nonconforming splices is a weakness in the engineering area.

GENERAL DISCUSSION OF FINDING:

a. The concern was that PMR 2329 did not contain an evaluation or documentation to show that the parameters of Wyle test report No. 17859-02P enveloped the WCGS environmental conditions.

b. The NRC was concerned that Work Request (WR) #4443-87 gave a "use-as-is" disposition for 38 Raychem splices with seal lengths of greater than 1/2 inch but less than 2 inches and bend radii less than five times the splice outer diameter without specific documentation of the configuration of each splice.

c. The Raychem splice was bent while heated in the Wyle Labs test report (17859-02P) versus bending while cool for the WCGS splices. The NRC considers the bending of cooled splices to be less conservative.

RESPONSE TO THE FINDING:

a. PMR 2329 did not contain the evaluation of Wyle Test Report No. 17859-02P because Environmental Qualification Work Packages (EQWP's), not MR's, are used at WCGS to document environmental qualification analysis. An applicability review of the test report was performed prior to the release of the PMR as stated in the Engineering Disposition to WR #4443-87 which was attached to the PMR. However, this review was not formally documented prior to the release of the PMR and included in EQWP-E-01013. The documented review is required to be included in the appropriate EQWP by Nuclear Plant Engineering Procedure KPN-D-319.

b. Only two of the 38 splices had seal length less than two inches. WR #4214-87, which is referenced in WR #4443-87, documents the seal length for each splice. Instrumentation and Controls personnel verified verbally to Nuclear Plant Engineering (NPE) that the bend radii of the splices were not less than 1.2 times the outside diameter of the splice, however, this phone conversation was not documented. While documentation of the bend radius of each splice should have been provided, the lack of documentation would not impact the "use-as-is" disposition. Confirmation that the bend

radii were greater than 1.2 times O.D. allowed the use of only one test report (No. 17859-02P) to address the deficiencies.

It should be noted that another Wyle Labs test report (No. 17859-02B), which was reviewed and determined applicable to WCGS, successfully tested a Raychem splice which was bent back on itself with essentially zero bend radius. The test parameters of this report bounds the WCGS conditions. This report was subsequently included in the Equipment Qualification Work Package.

c. The vendor instructions state that the tubing is not to be flexed until after cooling to the point it is comfortable to touch. This would prevent thinning of the tubing at the point of bending. Since the splices tested by Wyle were bent while heated, the testing indicates that acceptable qualification exists even if thinning may have occurred. In addition, the splice which was successfully tested with essentially zero bend radius in Wyle test No. 17869-02B was bent while cooled.

ACTIONS WHICH HAVE BEEN OR WILL BE TAKEN:

Verbal information utilized for dispositions, will be subsequently documented and included with the appropriate documents prior to formal closeout of the document.

CONCLUSION:

a. While it is true that the evaluation is not in the PMR, the performance and documentation of qualification test evaluations is addressed by NPE procedures. The results of these evaluations are in the EQWP's and are not required to be included in PMR's. This is not considered to be a deficiency.

b. Subsequent to this disposition, it was decided that all 38 splices should be replaced with splices completed strictly in accordance with the Raychem installation instructions. This replacement has been completed.

The NRC concern is partially addressed in that all the specific splice seal lengths were provided in WR #4214-87. While the bend radius for each splice should have been provided, inclusion of this information would not have changed the original "use-as-is" disposition.

c. The concern regarding bending of the splices while heated or cooled is not considered significant in that successful qualification tests have been performed for splices bent both while cool and while heated.

3.1.2.6 PMR 1828: ESW BUILDING CABLE REPLACEMENT

FINDING:

This PMR replaced cables routed to the Emergency Service Water (ESW) building based upon the need for additional circuits at the ESW structure and the identification of several failures in existing cables. Cable failures were discovered during the performance of surveillance testing in which circuit breakers failed to trip open on a non-safety related load shed signal during Safety Injection actuation. Investigation by the licensee resulted in the identification of grounded and open circuit conditions in a number of cables which had been pulled through the duct bank system from the Main Power Block to the ESW building. Subsequent evaluation of the failed cables identified damage in the form of cuts and nicks in the insulation and jackets of these cables. This damage was assumed to have occurred during the initial cable pull. Consequently, this PMR was issued to pull new cables to the ESW building.

Although the licensee performed an evaluation of the damaged cables, an additional evaluation to determine the root cause of the failures in the original safety related cables routed to the ESW building was not performed. The SSOMI team noted that banding material had been found in a duct bank associated with some of the damaged cables and was considered by the licensee to be the cause of the failures to the cables in that duct bank. However, the root cause for cable failures in redundant trains and cables associated with other duct banks has not been determined. The SSOMI team was concerned that the conditions associated with the original cable pulls were not evaluated to provide assurance that the cable failures were not the result of a generic condition.

GENERAL DISCUSSION OF FINDING:

The SSOMI team identified a concern with PMR 1828, ESW Building Cable Replacement, that the root cause for control cable failures had not been determined. The SSOMI team was concerned that an evaluation was not performed to eliminate the possibility of a generic condition affecting the opposite train.

RESPONSE TO THE FINDING:

The original revision of PMR 1828 provided a design for replacement of control cables in both safety related trains. After removal of the damaged train A control cables information was available to justify revision of the PMR to exclude the other train of control cables. The justification for concluding that the failures were not indicative of a generic condition involved the following facets:

(1) During the design development of PMR 1828, detailed cable pulling calculations were performed in order to add the maxisum number of spare control cables possible. These calculations represented a limiting case (more severe than the original design) which established that the original ductbank design (e.g. distance between manholes, slope of the ducts, etc.) provided for a safe pulling

length for the original control cable installed. Therefore, the ductbank design was eliminated as a potential root cause for the control cable failures.

(2) A review of the number of control cable failures and their physical routing within the ductbank was accomplished. This review revealed that seven conductors had failed in the A train and one conductor had failed in the B train. In addition, the A train conductors had the same physical routing within the ductbank.

(3) An evaluation of the damaged cables removed during the train A cable pull revealed damage to the outer jackets as well as conductor insulation damage at various locations. Based on discussions as well as visual observations of the cable (exposed copper was not corroded), some of the damage to the cable occurred during removal from the ductbank. However, some damage occurred prior to removal based on visual observation of copper corrosion found on the exposed conductor (in one case the conductor had completely corroded away). This damage is judged to be the leading contributor to the control cable failures that probably occurred during the initial installation.

(4) The available spare control cables were assessed for future contingencies. In order to maximize the available spare conductors, auxiliary relays were added to train B control circuits to provide two additional train B control conductors.

(5) The fact that only one control cable has failed in train B provides evidence that the control cables were not damaged during installation as indicated by the large number of failures in train A. This conclusion is based on the premise that cables that have a common failure mode should have approximately the same average time to failure under the same conditions. Therefore, if any other control cables in train B are affected, additional failures in train B should have been identified.

Based on these considerations, adequate justification existed to conclude that the failure mechanisms were isolated to train A only. Based on this conclusion, PMR 1828 was revised to eliminate the train B control cable replacement.

CONCLUSION:

WCNOC believes that adequate justification exists to conclude that the control cable failures were not indicative of a generic condition affecting the opposite train. It is true that the justification was not assembled as a part of the subject PMR and, therefore, not readily available for the SSOMI team involved with the installation and testing inspection.

3.1.2.7 CONTAINMENT PRESSURE TRANSMITTERS

FINDING:

During plant inspections associated with other modifications, it was noted that the connection box cover for Containment Pressure Transmitter PT-934, was missing one of two screws. Maintenance or modifications were not in progress on the transmitters. A field inspection of the other pressure transmitters identified the following discrepancies.

PT-934 - Missing 1 of 2 connection box screws.

PT-935 - No discrepancies noted.

PT-936 - Missing 1 of 2 connection box screws.

PT-937 - Missing 1 of 2 connection box screws.

PT-938 - No discrepancies noted.

PT-939 - Missing 1 of 2 connection box screws.

The licensee indicated that the missing cover screws may have been left out during a Raychem splice inspection. These discrepancies indicated a lack of attention to detail in maintenance and modification activities but were considered to be minor in nature.

GENERAL DISCUSSION OF FINDING:

The inspection team found one of two connection box cover screws on several of the containment pressure transmitters missing. Although the inspection report admitted this to be minor in nature, it concludes that the missing screws indicate an overall lack of attention to detail in performing maintenance or modifications.

RESPONSE TO THE FINDING:

Although the actual reason for the missing screws could not be determined, it is most likely that at some time during construction, startup or commercial operation the cover screws were lost. It is possible that because personnel were aware that the covers served no direct safety function such as environmental sealing or steam impingement protection and because one screw adequately holds the cover in place, efforts were not made to obtain replacement screws.

ACTIONS WHICH HAVE BEEN OR WILL BE TAKEN:

When Instrumentation and Controls was notified of the missing screws, replacement screws were immediately obtained and installed. The SSOMI Inspection report has been discussed in a monthly shop meeting with emphasis placed on workmanship requiring attention to detail and has been routed as required reading.

CONCLUSION:

The impact of the missing screws is considered minor in nature, however, workmanship and attention to detail has been emphasized in monthly shop meetings. No further specific action is considered necessary.

3.1.2.9 TECHNICAL SPECIFICATION TESTS

FINDING:

The SSOMI team monitored the performance of Technical Specification Surveillance Tests (STS) STS IC-280A, 'Analog Channel OP Test Ctrl Rm Detection Train A,' and STS IC-433, 'Channel Calibration NIS Post Accident Monitoring N61,' to ensure that they were performed in accordance with the requirements of the listed test procedures and were administratively controlled by procedure ADM 02-300, 'Surveillance Testing.' STS IC-280A verified the operability of the Chlorine Detection Control Room Ventilation Isolation System and STS IC-433 calibrated the Post Accident Monitoring Nuclear Instrumentation System (NIS).

The performance of STS IC-280A was in accordance with requirements. Test personnel were knowledgeable and appeared qualified to accomplish test objectives. However, several weaknesses were observed in the test instructions of STS IC-433. A lack of detail in some sections of the procedure resulted in confusion on the part of the Test Technician. For example, a note to Section 6.3.1.3.1 incorrectly referenced 'zero power' as a prerequisite for bypassing certain steps of the procedure. This note should have referenced a Nuclear Instrumentation System (NIS) Source Range level. In addition, Sections 6.3.2.2.2 and 6.3.2.3.2.1 specified test equipment connections which could not be accomplished without removal of the appropriate solid state circuit card. The removal and reinstallation of the circuit card and any subsequent requirement for equipment warm up were not detailed in the procedure. Consequently, interpretation on the part of test personnel was required in order to accomplish the test objectives. As a result of these discrepancies, the test performance was suspended until the procedure was revised.

The SSOMI team was unable to determine the extent of this concern because of the limited sample of tests and test procedures available for review. The test personnel were knowledgeable and this test could have been accomplished through application of the skills and experience which they possessed. However, the SSOMI team was concerned that detailed and accurate test instructions were not provided to ensure that test objectives and applicable TS requirements are fulfilled in approved surveillance procedures.

GENERAL DISCUSSION OF FINDING:

During the performance of STS-IC-433 the inspector identified what he considered several weaknesses in the test instructions. The term 'Zero Power' was considered inadequate because nuclear instrumentation system was not specifically mentioned as the source reading to determine 'Zero Power'. Additionally, two paragraphs identified test point connections that required circuit card removal to gain access to them. The procedure did not specify removal and reinstallation of the circuit card.

RESPONSE TO THE FINDING:

The procedure was initially developed directly from instructions contained in the controlled technical manual by personnel very familiar with the equipment. The manual instructions assume that the performer has the drawings in the manual readily available. It is possible that when the procedure was previously performed the technicians had the manual with them to aid in physically locating specified test points.

ACTIONS WHICH HAVE BEEN OR WILL BE TAKEN:

The test was suspended to allow the test performer time to research and change the procedure. Temporary procedure change (TPC) MA 87-481 was written to clarify the steps in question. Following TPC approval the test was satisfactorily completed. An ongoing review of procedures is providing a high level of confidence that test objectives and Technical Specifications are met.

CONCLUSION:

The inspection report states that "The test personnel were knowledgeable and this test could have been accomplished through application of the skills and experience which they possessed." The actions taken by the test performers when questions were encountered were proper. The test was stopped, clarification was obtained, and the procedure was enhanced prior to continuing. This is considered to be a normal part of improving procedure instructions as a result of experience. The weaknesses found in STS IC-433 did not compromise the valid performance of the test. No further action is considered necessary.

3.2.2.1 TEMPORARY MODIFICATION TMO 87-120 GK: CLAMPED OPEN CRVIS DAMPER

FINDING:

This modification clamped the Train A Control Room Emergency Ventilation System supply damper in the open, actuated position in response to several failures of the actuating linkage. The SSOMI team determined that the licensee had not adequately determined whether the system remained operable and capable of pressurizing the control room as required by TS 3/4.7.6.

TS 3/4.7.6, "Plant Systems - Control Room Emergency Ventilation System," requires that the Control Room Ventilation Isolation System (CRVIS) be operable in all modes and capable of pressurizing the control room to 0.25 inches of water (gauge) upon detection of radiation or toxic gas in the air intakes. In addition, Section 3.1 of the USAR and 10 CFR 50, Appendix A, require that safety systems be designed to automatically protect against single failures of passive and active components. The system configuration established by the temporary modification provided a bypass path for supply air to the control room if the Train A fan failed to automatically start on a CRVIS initiation signal. Upon CRVIS actuation, the Train B CRVIS fan would start and discharge air to the fan discharge plenum, however the clamped open Train A damper would permit backflow through the idle Train A CRVIS fan and bypass the control room. In this configuration, the CRVIS would not be able to provide the required positive pressure in the control room, assuming the single failure of Train A CRVIS fan. Additionally, the licensee had not performed a calculation or a functional test to demonstrate the ability of the Train B CRVIS to maintain the required control room pressure in this degraded mode.

Additional concerns associated with this temporary modification were noted as follows:

a. The licensee failed to recognize the requirement for the safety system to remain operable with a single failure without operator action. In order to ensure the operability of the CRVIS, the temporary modification required an operator to remove the clamp from the failed CRVIS damper. Although these actions would permit Train A to be isolated upon fan failure and therefore satisfy the single failure design of the system, the need for operator action to meet the single failure design of a safety system does not conform to the requirements of 10 CFR 50, Appendix A.

b. Even though the operator actions did not meet single failure design requirements, the specified operator actions would not be sufficiently responsive when considering the design requirement of the CRVIS to maintain a positive pressure in the control room in the event of radiation or gas in the air intakes. The emergency instructions require the operator to don a self-contained breathing apparatus, go to the damper, climb a ladder to the blocked damper, remove the clamp and manually ensure that the damper has closed. Furthermore, the SSOMI inspector noted that these instructions would have been difficult to implement in an emergency because they were not found in the Alarm Response Procedure as normally expected, but were included as an addendum to the temporary modification.

c. Despite having performed repeated Safety Evaluations on this and other CRVIS dampers which were similarly clamped open, the licensee failed to recognize that additional testing or calculations were necessary to verify that the system remained operable with the temporary modification implemented.

d. Appropriate corrective action in preventing repeated damper failures was not taken Five CRVIS damper failures were experienced during the period of June 25 to November 3, 1987. As a result, replacement operating gear to repair the broken train A CRVIS supply damper was not available. Appropriate corrective actions require a consideration of past defects and noncompliances in basic components important to safety. WCGS Procedure ADM 01-033, "Instructions Describing Reportability Review and Documentation of Licensee Event Reports and Defect Deficiencies," Rev. 16, specified evaluation and reporting requirements pursuant to 10 CFR 21. The licensee had not identified the above failures as potentially reportable nor evaluated the failures per the above procedure.

GENERAL DISCUSSION OF FINDING:

a.,b.,c.

The SSOMI team stated that the 10CFR50.59 Safety Evaluation did not adequately determine if the CRVIS damper blocked in its safeguards position by the subject modification impaired the operability of the CRVIS function in fulfillment of the systems safety function. Additionally, they stated that the temporary change introduced a stated compromise of the single failure criteria which is not in accordance with the general design criteria for nuclear power plants.

The Technical Specifications require two independent Control Room Emergency Ventilation Systems to be operable in all MODES. The ACTION statement during MODES 5 and 6 states that with both Control Room Emergency Ventilation Systems inoperable suspend all operations involving core alterations or positive reactivity changes.

RESPONSE TO THE FINDING:

a.,b.,c.

In the USAR Chapter 15 analysis, the bounding worst single failure has been ascertained to be the failure of the filtration fan in one of the two filtration system trains. Operator action is required to isolate the train with the failed filtration fan. At the same time, one train of the control room pressurization system will also be isolated. Prior to isolation, a potential pathway exists allowing air from the control building to enter the control room, bypassing the control room filtration filters. After isolation, one control room pressurization fan and one control room filtration fan operate for the duration of the accident. No bypass pathways then exist for unfiltered air to enter the control room.

Blocking the subject damper open in its safeguards position did not put the system in a condition outside of previous analysis nor create a different type of accident nor increase the probability or possibility of a malfunction from previous analysis. Safety Evaluation Three of USAR Section 6.4.4 for the Control Room Ventilation System states that "no single failure will compromise the systems safety functions." In view of the inherent design aspects of the system, the denial of the damper to go closed upon occurrence of an active failure does not put the system outside of previous analyzed system failures since other limiting failures exist.

The subject dampers are closed by their motor operators when the air conditioning units fan motor is deenergized. They do not have a spring return; hence, the damper failure mode is "as is". Clamping these dampers open would place them in their safeguards position where they could fulfill their intended CRVIS safety function. A single failure per 10CFR50 Appendix A means an occurrence which results in the loss of the capability of a component to perform its intended safety function. Clamping the damper(s) open does not compromise the CRVIS safety function. The designated operator action was not essential, but would have placed the system in a conservative tested configuration.

d. Defect Deficiency Report 87-122 was initiated on November 17, 1987, as a result the SSOMI team concern about performing a 10 CFR Part 21 reportability evaluation of mechanical problems associated with damper GK-D-084 and its actuator GK-HZ-40A. A review of the maintenance work requests for dampers/actuators GK-D-080/GK-HZ-29A, GK-D-081/GK-HZ-029D, GK-D-084/GK-HZ-40A, GK-D-085/GK-HZ-40B was conducted. This review identified that the actuators were reworked during 1984 with some machining being conducted on the couplings. In 1985, GK-HZ-40A and B were replaced due to the actuators being jammed. In 1987, failures of GK-HZ-29A, 40A and 40B, occurred which required replacement of the actuators. Discussions with a vendor representative indicated that these failures could be attributed to a misalignment of the coupling and the saddle because of previous maintenance and the method of clamping the dampers when blocking them open or closed. The vendor has subsequently provided appropriate tolerances and coupling clearances to ensure proper coupling alignment. WCNOC concluded that this deficiency was not reportable pursuant to 10 CFR Part 21.

The appropriate tolerances and coupling clearances were obtained and an alignment jig was fabricated to assist in alignment of the saddle and the coupling. Three of the above dampers were realigned. The fourth damper was inspected and determined to be within the tolerances provided.

CONCLUSION:

a.,b.,c.

The 10CFR50.59 Safety Evaluation was not deficient in its analysis of the subject change. The condition described in the evaluation was bounded by the analysis presented in Chapter 15 of the Updated Safety Analysis Report. Damper replacement parts were installed and the CRVIS system was restored on November 26, 1987.

d. As discussed above, WCNOC believes the cause of the repetitive damper problems during 1987 have been identified and corrected

3.2.2.2 PMR 2106: PRESSURIZER SPRAY VALVE BONNET REPAIR

FINDING:

This PMR involved the installation of gland plugs or set screws in four injection holes as permanent modification to the Pressurizer Spray Valve. The four injection holes were drilled in the bonnet of the Pressurizer Spray Valve in order to provide an injection path for a liquid sealant (Furmanite) which was used to repair the body-to-bonnet steam leak.

The engineering disposition of Engineering Evaluation Request (EER) 85-XX-37, which requested that NPE approve the use of certain sealing compounds in temporary repair procedures at the discretion of the Maintenance Superintendent, previously approved the repair of leaking mechanical joints in piping components such as flanges, valve packing, and bolted valve bonnets by the use of sealant injection, provided that the sealant chemistry requirements, application procedure, and system limitations specified in the engineering disposition were followed. The disposition approved the generic use of liquid sealing compounds such as Furmanite and authorized drilling holes into pressure retaining parts of ASME Code Class I components in order to facilitate the repair process.

WR 00101-87 and associated revisions, the vendor work request and procedure, the engineering disposition to PMR 2106 and other associated documentation which were a part of the work package were reviewed. The following concerns were identified:

a. Section 3.2, "Bolted Connection," of the application procedure, which was detailed in the engineering disposition to EER 85-XX-37 and used to repair the Pressurizer Spray Valve, was not verified to meet the ASME Code requirements. The Justification of Engineering Resolution for EER 85-XX-37 indicated that the disposition ensured that Code requirements were not violated. NPE subsequently indicated that the requirements cited in the Justification of Engineering Resolution were obtained from the vendor and not from the ASME Code as indicated. When requested by the SSOMI team, the licensee could not demonstrate that the requirements provided by the vendor met the Code requirements.

b. The SSOMI team considered that the 10 CFR 50.59 Safety Evaluation of EER 85-XX-37 was inadequate, in that it did not identify that drilling holes into pressure retaining parts of ASME Class I components involved changes to the facility. Although the Safety evaluation performed for WR 00101-87, Rev. 3, correctly determined that the holes drilled in the pressurized bonnet were a change to the facility, the explanation given for determining whether this instance involved an unreviewed safety question was inadequate because it failed to address the safety significance of the modification to the Pressurizer Spray Valve using EER 85-XX-37.

c. The engineering disposition to EER 85-XX-37 required the sealing compound injection pressure to be calculated so as to limit the injection pressure and thereby limit the stress on the flange bolts. Documentation of these calculations was not available. Additionally, the maximum pressure used to inject the sealant compound in the body-to-bonnet area of the

Pressurizer Spray Valve was not recorded. Because the maximum pressure was not calculated and records did not exist to demonstrate the pressure used, the flange bolt stresses could have been exceeded for the Pressurizer Spray Valve during the injection process.

d. A Safety Evaluation was not performed for the vendor work procedure to repair the Pressurizer Spray Valve by sealant injection as required. ADM 07-100 requires that a 10 CFR 50.59 Safety Evaluation be completed for all procedures reviewed by the Plant Safety Review Committee (PSRC).

GENERAL DISCUSSION OF FINDING:

a. During the review of PMR 2106, the SSOMI team determined that the Engineering Disposition to EER 85-XX-37 was not verified to meet the ASME Code requirements.

b. During the review of PMR 2106, the SSOMI team determined that the Safety Evaluation associated with EER 85-XX-37 was inadequate in that it did not identify that drilling holes into pressure retaining parts of ASME Class 1 components involved changes in the facility. Also, the SSOMI team determined that the Safety Evaluation performed for WR 00101-87, Rev. 3 was inadequate in that it failed to address the safety significance of the modification to the Pressurizer Spray Valve using EER 85-XX-37.

c.,d.
The repair of the pressurizer spray valve (PMR 2106) was a SSOMI team audit item. The content of the PMR was a repair instruction on holes drilled into the packing box of the pressurizer spray valve to stop a body-to-bonnet leak by the use of "Furmanite". The PMR issued was not implemented and was cancelled. A replacement of the valve packing box was performed instead.

During the investigation, two concerns were expressed in the procedures for allowing Furmanite to inject the valve.

1) Documenting the injection pressure of the Furmanite compound.

2) No Safety Evaluation was performed on the Vendor Procedure used to implement the vendor work plan.

RESPONSE TO THE FINDING:

a. The disposition to EER 85-XX-37 provided generic guidelines to facilitate temporary on-line sealing of leaking components. This disposition was used for the temporary on-line sealing of the packing box flange on valve BB-PCV-455B and applied the guidelines set forth in Section 3.2, "Bolted Connection", of the subject disposition.

The disposition to EER 85-XX-37 also identified systems which come into contact with primary fluid and required approval from WCNOC Management prior to performing sealant injection on these systems. However, the disposition does not clearly identify the Code(s) to which the requirements for drilling holes into "Bolted Connections" comply with and

did not include provisions to verify code compliance for specific components to which these generic guidelines would be applied.

Work Request #00101-87, Rev. 3 requested an engineering evaluation of a proposed permanent repair to the temporary injection holes. The engineering disposition to WR #00101-87, Rev. 3 required that the injection holes made on valve BB-PCV-455B during the temporary on-line sealing process be permanently weld repaired in accordance with the Section XI repair and replacement program.

b. 1. The Safety Evaluation issued with EER 85-XX-37 did not identify a change in the facility as described in the safety analysis report. As indicated on the Safety Evaluation, this evaluation "addresses only the suitability of the sealant materials, the injection procedures and the effect of the procedure on the sealed components". As previously discussed in the response to SSOMI item 3.2.2.2.a, the disposition to EER 85-XX-37 provided generic guidelines to facilitate temporary on-line sealing of leaking components and did not apply to any specific component.

2. Work Request 00101-87, Rev. 3, requested an engineering evaluation of a proposed permanent repair to valve BB-PCV-455B which had previously been sealed on-line by Furmanite. The proposed repair involved using gland plugs or set screws to seal the injection holes made during the on-line sealing process.

After review of the specific request to provide a permanent repair using gland plugs or set screws, Engineering determined that the injection holes would need to be welded per ASME Section XI Repair/Replacement Program. The engineering disposition and associated Safety Evaluations performed address only the specific repair request in WR 00101-87, Rev. 3. Therefore, the Safety Evaluation provided for the disposition to WR 00101-87 Rev. 3, which, as discussed above, has as its scope only the permanent repair of the temporary injection holes, and is considered adequate within the confines of the scope of the final permanent repair. The subject Safety Evaluation was not intended to address the alteration (temporary) made to the pressurizer spray valve to facilitate on line sealing.

c. An Engineering Evaluation Request, (EER 85-XX-37), disposition was used in preparing and planning the injection of the pressurizer spray valve. The disposition gave generic criteria to be met when using Furmanite to inject a component to stop leakage. The disposition required that the injection pressure should be limited such that when the injection pressure times the bearing area of the sealant plus the system operating pressure times the area subjected to system pressure shall not exceed 1.1 times the flange bolt allowable stress. This formula is used to figure injection pressures. This value has been recalculated and the injection pressure specified (4500 psi) was a conservative number.

A second part of this concern was that the injection pressure was not recorded. There was no requirement in the work package Instruction Checklist to record the injection pressure.

d. At the time of the injection of the valve, there were no procedural requirements to process a Safety Evaluation (10 CFR 50.59) for 'Vendor Procedures'. At this time there are no procedural requirements to process a Safety Evaluation for 'Vendor Procedures'. However, PSRC review of these procedures is required.

ACTIONS WHICH HAVE BEEN OR WILL BE TAKEN:

a. The guidelines provided in the disposition to EER 85-XX-37 for temporary on line sealing were intended to be generic guidelines only and did not apply to any specific component(s). This disposition will be withdrawn and modified to mandate provisions for verification of Code compliance prior to applying the generic guidelines for temporary on-line sealing of specific components. This corrective action shall be accomplished prior to June 1, 1988.

b. As indicated in the response to SSOMI item 3.2.2.2.a, EER 85-XX-37 is being withdrawn and will be modified to mandate provisions for Code compliance verification prior to performing any temporary on-line sealing of specific components. The associated Safety Evaluation will be provided with the modified EER disposition.

CONCLUSION:

a. WCNOC agrees with the SSOMI teams observation that the guidelines outlined in the disposition to EER 85-XX-37 were not verified to meet ASME Code requirements for all components.

As indicated above, these generic guidelines were not applicable to specific components. Therefore, the previous disposition is being withdrawn for revision to indicate that application of the guidelines needs to be verified for compliance with the applicable design Code of the specific application prior to implementation.

It should be noted that the ASME code neither allows nor disallows temporary drilling of components for the purpose of on-line sealing. Verification of Code Compliance in these cases needs to consider the effect of this type of temporary condition, i.e. drilling holes into component parts, on the original design (structural) basis of the subject component.

In a letter dated January 25, 1988, from L. C. Shao, Director Division of Engineering & Systems Technology, USNRC, to J. L. Milhoan, Director Division of Reactor Safety, Region IV, USNRC, concerning the use of sealing fluids on primary pressure boundary components, the NRC evaluated the concern identified at WCGS. The NRC stated that, 'In conclusion, temporary on-line leak sealing of components is an acceptable alternative to an unscheduled plant shutdown to effect a permanent repair provided

that the applicable guidelines are met and the Quality Assurance measures are followed for proper selection, procurement and application of sealants".

b. As discussed above, WCNOC acknowledges the SSOMI team concern regarding the Safety Evaluation associated with EER 85-XX-37.

The SSOMI team concern regarding the adequacy of the Safety Evaluation provided with work request WR 00101-87, Rev. 3, is not considered to be a deficiency. As discussed above the subject work request and associated safety evaluation was limited in scope to only address a final permanent repair.

c.,d.
Although not documented, the injection pressure specified, 4500 psi, was proven to be a conservative number. Also, there was no requirement to document the actual injection pressure but QC accepted the pressure that was used. Based on these two facts, no corrective action is necessary for item 4.2.2.3.c.

A Safety Evaluation is presently not required for Vendor Procedures. The Work Controls Task Force is reviewing the Vendor Work Plan Procedure (ADM 01-043). This procedure will be revised by August 31, 1988, to require safety evaluations to be performed for vendor procedures. Due to the procedural requirements and actions being taken, no further corrective action is required.

3.2.2.3 PMR 7084: CCW PIPE WALL THINNING

FINDING:

This PMR involved application of a weld overlay on a Component Cooling Water (CCW) line servicing the Train A Residual Heat Removal (RHR) Heat Exchanger in order to repair a pipe section downstream of Valve EJ-V033. The licensee's ultrasonic examination of the piping section determined that the pipe section was below minimum wall thickness. Although subsequent ultrasonic examinations determined that these results were erroneous because laminar inclusions (acceptable conditions), rather than the wall inner diameter, were being identified, the SSOMI team identified the following concerns:

a. The licensee did not declare Train A of the CCW and RHR systems inoperable per TS after identifying that the piping was below minimum wall thickness. The licensee subsequently issued a guidance memorandum on July 29, 1987, directing the plant operators to consider system as inoperable if requirements for minimum piping wall thickness were violated. The general matter of the licensee's handling of these and other similar operability matters was subject of an NRC:NRR-licensee meeting in NRC headquarters on November 17, 1987.

b. The piping was repaired without being isolated and drained of water, even though the test results showed wall thinning down to nearly 1/16 inch (about 22 percent nominal). Although Welding Procedures WPS1-0000, "ASME/ANSI General Requirements," Rev. 2, permitted welding under these conditions, the SSOMI team considered that the procedure represented a high risk of wall burn through, threatened system pressure boundary integrity, and therefore represented a unconservative repair procedure.

c. Major changes in the work instructions for weld overlay repairs on the EJ-V033 piping were implemented by Revision 2 to WR 0702-87 but were not incorporated into the revised ASME Section XI work instructions as required by Section 9.3.6.1 of ADM 01-036, "WCGS ASME Section XI Repair and Replacement Program," Rev. 2. ADM 01-036 required that work packages contain complete and concise work instructions to accomplish repairs and provided guidance for content. Revision 2 to the WR implemented changes in the basic repair procedure and included requirements for in-service leak testing and radiography. The existing work instruction was annotated to delete all existing steps by Revision 2, but replacement work steps were not provided. Additionally, the repairs were compiled without craft or QC signoff of the revised work instructions.

d. The post modification leak testing of the piping was not accomplished as required by procedure until two months following completion of the work.

e. During the review of this PMR, the SSOMI team also noted that ASME required radiography was incorrectly deferred by the engineering disposition of Corrective Work Request (CWR) 0702-87, Rev. 1, for about six months. 10 CFR 50.55a(g)(5)(iii) requires NRC notification when conformance with certain code requirements is impractical. The licensee did not initially notify the NRC when the radiography was deferred.

Following identification of similar oversights by the NRC Resident Inspectors, the licensee submitted a Code Relief Request for the radiography of PMR 2084 repairs on August 24, 1987.

GENERAL DISCUSSION OF FINDING:

a. The NRC states that at the time the ultrasonic testing determined that there was a below minimum wall condition on a pipe section downstream of valve EJ-V033, plant operators should have declared Train A of the CCW and RHR systems inoperable. WCNOC did not initially declare Train A of the CCW and RHR systems inoperable, since when it was initially determined that the CCW and RHR piping was below the code minimum wall thickness allowed for new pipe, the Plant Staff was not aware that this condition represented a potential noncompliance that needed to be expeditiously evaluated by Engineering to determine whether or not the as-found pipe thickness was sufficient to maintain maximum design basis stress below the code allowable stresses.

b. The piping section, SA-106GrB Carbon Steel Pipe, 0.375" nominal wall thickness, was repaired utilizing an external weld overlay which extended around the entire circumference of the pipe. This weld repair was performed without the piping section being drained of water, even though the ultrasonic examination indicated that the piping had localized areas of wall thinning as thin as .086". Although the applicable welding procedures permitted welding under these conditions, the SSOMI team considered that the procedure represented a high risk of wall burn through, threatened system pressure boundary integrity, and therefore represented a unconservative repair procedure.

c. Revision 2 to Work Request 00702-87 deleted all previous work instructions, implemented changes to the basic repair procedure, included requirements for inservice leak testing, and deferred the required radiographic examination until the next refueling outage. Complete and concise work instructions were not developed and incorporated into the revised ASME Section XI work instructions as required by the applicable WCNOC procedure. Subsequently the repair was completed without craft signoffs of the revised work instructions.

d. The required post modification leak testing of the piping was not accomplished until two months following completion of the work.

e. Because the intent of the weld overlay was to restore the nominal pipe wall thickness to .375 inches and this repair exceeded the lessor of 3/8 inch or 10% of section wall thickness, radiography of the repaired area in accordance with the ASME B&PV Code, Section III, Subsection ND was required. However, further review of the repair process revealed that:

1. the area of repair would contain an irregular surface on the interior of the pipe due to postulated erosion.

2. the repair was to be conducted with the pipe filled with water.

These two factors were judged to make examination by the radiography method difficult to interpret with meaningful results because of the

irregular surface and that the water would induce diminished sensitivity. In lieu of radiography, Engineering determined that surface examination of each weld deposit layer and adjacent heat affected zone by the liquid penetrant or magnetic particle method would provide adequate assurance of the suitability of service by the repair weldment until the Code required examination or suitable material replacement could be performed.

RESPONSE TO THE FINDING:

a. When it was initially determined that the CCW and RHR piping was below the code minimum wall thickness allowed for new pipe, the Plant Staff was not aware that this condition represented a potential noncompliance that needed to be evaluated by Engineering to determine whether or not the as-found pipe thickness was sufficient to maintain maximum design basis stress below the code allowable stresses. Therefore, the time between initially discovery and subsequent declaration of the EJ piping as inoperable was inappropriately long.

Although the November 17, 1987, meeting discussed in the finding is not a major issue for this item, it should be noted that the meeting was held at the request of WCNOC and the primary subject of the meeting was the WCNOC pipe wall inspection program and the results obtained to that date.

b. WCNOC personnel considered the following factors when developing this repair procedure and concluded that this repair could be performed with an acceptable level of risk.

1) Pressure/temperature of portion of system being repaired. This portion of the CCW/RHR system had been removed from service and allowed to cool to ambient temperature. The system head pressure, not system operating pressure, was present during the repair operation.

2) To minimize the possibility of wall burn through, the first weld layer was deposited utilizing the Gas Tungsten Arc Welding process, maintaining as low amperage as possible, while remaining within the Welding Procedure Specification limitations, to localize and reduce heat input. Additional weld metal was deposited utilizing the Shielded Metal Arc Welding process, increasing amperage and/or electrode diameter with subsequent layers.

3) Previous experience of WCNOC personnel had been that had the welder 'burned through' the pipe wall it would create minor leakage, and not a serious pipe failure.

4) WCNOC personnel performing the repair, were in constant contact with the control room to initiate system isolation and drain down if the welder 'burned through' the pipe.

c. The applicable procedures for work requests and ASME Section XI Packages contain different requirements for the development and revision of work instructions. Personnel failed to follow correct procedural requirements for development and revision of the ASME Section XI Package work instructions, per ADM 01-036.

d. Post modification testing for this repair consisted of an ultrasonic thickness examination, radiographic examination and a leak test at normal operating temperature and pressure. However, due to the assumed internal damage to the pipe from cavitation erosion, it was determined that obtaining code acceptable radiographs, and their subsequent interpretation, would be extremely difficult, if not impossible. Based on this assumption, the radiographic examination was deferred until the next refueling outage.

The work instructions for post modification testing did not clearly define the timeliness of the required testing. Subsequently, upon completion of the ultrasonic thickness examinations the work package was incorrectly placed in a "hold" file for completion of testing during an upcoming refueling outage prior to performing the required leak testing.

e. The reason for deferring that radiographic examination without prior NRC notification was a failure to recognize the applicability of 10 CFR 50.55a(g)(5)(iii) to post maintenance testing. The requirements were interpreted to be applicable only to inservice testing and not post maintenance testing.

ACTIONS WHICH HAVE BEEN OR WILL BE TAKEN:

a. Procedure requirements ensure that engineering is promptly notified of wall thinning concerns, and that engineering evaluations are completed promptly to support operability determinations. These procedures will minimize the time period for identifying equipment which is in an inoperable condition.

c. WCNOC Maintenance/Modification personnel, are developing a standardized work package preparation program which will place all work instructions in a word processing program for producing step by step work instructions on a standardized format. This program will be developed and implemented by August 31, 1988.

d. The required post modification leak test has been performed. Maintenance Engineers have been cautioned to follow up with timely review and completion of NDE requirements and subsequent post modification tests.

e. Upon notification from the NRC that deferring post maintenance testing requirements of ASME Code Class 1, 2, and 3 components was governed by the requirements of 10 CFR 50.55a(g)(5)(iii), WCNOC requested and received, on June 23, 1987, temporary relief from ASME Section XI in order to defer the radiography required by the Code. On June 30, 1987, (not August 24, 1987 as stated in the report) a formal request for relief along with the Technical Justification for the pipe section downstream of valve EJ-V033 was submitted to the NRC.

CONCLUSION:

a. Subsequent to the determination that the EJ piping could potentially exceed Code allowable stress for a worst case design basis event, it was declared inoperable. The timeliness in requesting the evaluation and making that determination was inappropriately long. Procedure requirements ensure that engineering is promptly notified of wall thinning concerns, and that engineering evaluations are completed promptly to support operability determinations. These procedures will minimize the time period for identifying equipment which is in an inoperable condition.

b. In conclusion, WCNOC personnel determined that performing this weld repair with 'stagnant' water in the system could be satisfactorily accomplished without putting the system or plant in jeopardy.

c. The completion of the standardized work package preparation program will resolve this concern.

d. Typically the required post modification testing is performed as soon as possible upon completion of work, therefore timeliness of testing is not normally specified. However, in this case where a deferment of the radiographic examination was being performed, the leak test should have been specified to be performed when the system was returned to normal operating temperature and pressure.

   Clearly, deferment of a required post modification radiographic examination is an unusual deviation from normal operating procedures. In future instances when requirements are deferred, the timeliness of all testing will be clearly specified in the work instructions

e. At the time of the deferment of the radiography for piping downstream of valve EJ-V033, WCNOC did not recognize the requirement of 10 CFR 50.55a to notify the Commission for this case. Since WCNOC is now aware of this requirement, the Commission is being notified of cases in which post maintenance testing for ASME components is going to be deferred.

3.2.2.4 PMR 2116: VALVES EF-V090, EF-V058, EF-HV47 AND EF-HV48 HARD SURFACING

FINDING:

This PMR involved the final repairs of Essential Service Water (ESW) system piping, including EF-V090 piping downstream of Valve EF-V058 on the opposite train. Additionally, this modification involved replacement of a carbon steel 24" X 16" bell reducer with a stainless steel replacement more resistant to erosion and corrosion. Adjacent piping was also hard surfaced with either stainless steel or stellite weld overlays. The PMR, associated WR packages and in progress welding and fitup of a new piping bell reducer at EF-V058 were reviewed. On November 12, 1987, during the SSOMI inspection, work on the above WRs and all safety related pipe fitting and welding activity conducted by the Maintenance Department were suspended by QA Work Hold Agreement #23. The Work Hold Agreement, issued jointly by the QA and Maintenance Departments, cited for fourteen Conditions Adverse to Quality involving PMR 2116 as the basis for the work suspension. These included improper piping cutouts, marginal or inadequate piping fitups, failure to temporarily support piping for spool removal, out of tolerance spool fabrication, bypassed QC hold points, improper weld buildups and overlays, and others. This work hold followed a previous QC requested work stoppage.

Maintenance and QC Department management indicated that the problems associated with PMR 2216 were caused by the Maintenance Department's lack of experience in major pipe fitting modifications. Previously, such work was performed by contractors under the direction of the Facilities and Modifications (F&M) Department. This modification was transferred to F&M on November 13, 1987, and work resumed following review and revision of work instructions on November 14, 1987.

No concerns were identified with respect to this work. However, the SSOMI team's review of testing requirements identified that the hydrostatic test instructions for replacement of ESW piping downstream of Valve EF-V58, did not consider possible overpressurization of adjacent systems. The testing pressurized a portion of the ESW System to 220 psig and required a test relief valve set at 239 psig. The procedure provided a single valve isolation for thirteen heat exchangers served by the piping. No provisions were included to ensure that the adjacent components were aligned such that test boundary valve leakage will be vented to atmosphere or the untested portion of the system.

The SSOMI team was concerned that the test boundary valve leakage without component protection could result in pressurization of the heat exchangers in excess of the pressures allowable under ASME Section XI.

GENERAL DISCUSSION OF FINDING:

The SSOMI team finding concerning the Hydrostatic Test performed downstream of valve EF-V058 per Work Request #2931-87 is outlined in 3 items:

1) Instructions issued with the test package did not consider the possibility of overpressurizing adjacent ASME systems.

2) The valve lineup for the test provided only single valve isolation for thirteen ASME heat exchangers. Test instructions did not provide a means of venting possible leakage past single isolation boundary valves to atmosphere or to the untested portion of the system.

3) Boundary valve leakage may have subjected several ASME heat exchangers to pressures exceeding ASME Section XI allowances.

RESPONSE TO THE FINDING:

Subsequent to the finding, Revision 5 to Work Request 2931-87 added several open valves to the lineup to permit venting boundary valve leakage into the untested portion of the EF System.

Valves EF-V332 and EF-V227, open to the atmosphere, provided overpressure protection to heat exchangers in some adjacent systems as follows:

SGL11A was isolated from the hydrostatic test by boundary valve EF-V057. On the inlet side of this heat exchanger, valve EF-V056 was open with valve EF-V227 vented to atmosphere.

SGK04A was isolated from the hydrostatic test by boundary valve EF-V040. On the inlet side valve EF-V039 was open with valve EF-V227 vented to atmosphere.

SGG04A was isolated from the hydrostatic test by boundary valve EF-V147. On the inlet side of this unit, valve EF-V146 was open and EF-V227 vented to atmosphere.

SGL15A was isolated from the hydrostatic test by valve EF-V042. On the unit inlet side, valve EF-V041 was open. Valve EF-V227 was vented to the atmosphere.

SGF02A was isolated from the hydrostatic test by valve EF-V048. On its inlet side, valve EF-V047 was open with EF-V227 vented to the atmosphere.

SGL12A was isolated from the hydrostatic test by valve EF-V030. Upstream from this unit valve EF-V029 was open with EF-V227 vented to the atmosphere.

SGL09A was isolated from the hydrostatic test by valve EF-V033. On the inlet side, Valve EF-V032 was open, and EF-V227 vented to atmosphere.

SGL13A was isolated from the hydrostatic test by valve EF-V036. On the inlet side of this unit, valve EF-V035 was open, EF-V227 vented to atmosphere.

SGL10A was isolated from the hydrostatic test by valve EF-V038. Upstream, Valve EF-V037 was open with valve EF-V227 vented to atmosphere.

Clearance Order 87-1228-EF shows valve EF-V272 open and valve EF-V332 open to atmosphere, providing added protection against overpressurization for the heat exchangers listed above.

The outlet from heat exchanger EEG01A was isolated by valve EF-HV59. Also, valve EF-V058 was removed during the test. Note that with EF-V058 removed, line EG-208HBC-16 is open at the flanged end of pipe spool EF03-S006. This provided a vent to atmosphere preventing accidental pressurization of the EEG01A shell side in case of boundary valve leakage.

Containment cooler heat exchangers SGN01A and SGN01C were isolated by valves EF-HV45, EF-HV47, and EF-HV49. Thus these heat exchangers, were 'double' isolated from the test boundary after that portion of the line was filled and vented.

SGK05A and the diesel generator coolers on 'A' train were double isolated from the hydrostatic test by valves EF-V053, EF-V055, and EF-V273.

ACTIONS WHICH HAVE BEEN OR WILL BE TAKEN:

WCNOC believes the finding as stated was correct. Prior to the hydrostatic test Work Request 2931-87 was revised accordingly, thus ensuring that the heat exchangers mentioned above would not be pressurized in excess of ASME Section XI limits.

A revision to the hydrostatic and pneumatic test procedure MGM-M00C-02 will be made to add precaution 7.2.4.1, 'The test instruction must include provisions for over pressure protection of systems or components adjacent to single isolation test boundary valves. Leakage past these single isolation boundary valves must not subject other systems or components to pressure exceeding ASME Section XI allowances'. This procedure revision will be completed by July 1, 1988.

CONCLUSION:

WCNOC agrees that the subject finding is correct and the corrective actions taken in response to this finding should preclude the pressurization of heat exchangers in excess of the pressures allowed by ASME Section XI.

3.2.2.6 PMR 0904: ESSENTIAL SERVICE WATER CHECK VALVES

FINDING:

This PMR installed isolation valves for Essential Service Water (ESW) Check Valves EF-V0046 and EF-V0076 to improve the maintenance and testing of the check valves. The PMR added two new gate valves, EF-V0345 and EF-V0346. The review of the PMR and the associated work packages found that Clearance Order No. 871061EF, which was used to establish the clearance boundaries for the work, did not provide correlation between the initials and signatures of the individuals establishing and verifying the boundary as required by 10 CFR 50, Appendix B, Criterion XVII, 'Records'. Criterion XVII requires that quality records shall, as a minimum, identify the inspector or data recorder. Other licensee procedures, such as surveillance procedures, typically include a tabulation which correlates the individual's initials and signatures to permit positive identification. The SSOMI team considered that the personnel performing valve lineups should be clearly identified.

GENERAL DISCUSSION OF FINDING:

During an NRC review of Clearance Order 87-1061-EF, which was used to establish clearance boundaries for implementation of PMR 0904, it was found that there was no correlation between the initials and signatures of the individuals establishing and verifying the boundary. The NRC's basis for this Finding, as stated, is that 10 CFR 50, Appendix B, Criterion XVII 'Records' requires that Quality Records shall, as a minimum, identify the inspector or data recorder.

RESPONSE TO THE FINDING:

10 CFR 50, Appendix B, Criterion XVII specifically states Inspection and Test Records shall, as a minimum, identify the inspector or data recorder, the type of observation, the results, the acceptability, and the action taken in conjunction with any deficiencies noted.'

The Clearance Order form is used by operations to establish and document safe working boundaries for maintenance/modification activities and is not considered an inspection or test record.

CONCLUSION:

WCNOC does not consider this item to be a deficiency based on the requirements of 10 CFR 50, Appendix B, Criterion XVII. It should be noted that WCNOC management desires and is capable of identifying the individuals establishing clearance boundaries. This can be accomplished through a process of determining the shift in which the clearance was established and reviewing Operations Special Order #7, 'Safety Tagging for Personnel Authorized to Perform Tagging Activities'. This Special Order lists tagging Authorities as well as individuals authorized to hang and remove tags. In addition, this Special Order specifies personnel authorized to sign on and off of Clearances. Additional methods are being reviewed to improve the capability to identify a specific individual involved in tagging under the safety tagging program. This review will be completed by June 1, 1988.

3.2.2.7 PMR 1363: CHARGING PUMP CONTROL VALVE CAVITATION DAMAGE

FINDING:

This PMR replaced the trim in Charging Pump Flow Control Valve BG-FCV121 with a new design. The former valve trim experienced cavitation damage because of the high flow and high pressure drop while in service.

WR 4430-86, used to implement PMR 1363 for Valve FCV-121, did not note that PMRs 1613 and 1635 were performed concurrently. Contrary to Note 9 on Copes Vulcan Drawing D-283137, which required a total of nine packing rings to be installed, Step 7 of the work package was annotated to indicate that twelve rings of packing were installed. The licensee indicated that the reason there were more packing rings installed was because the licensee had implemented PMRs 1613 and 1635 concurrently with PMR 1363. PMRs 1613 and 1635 removed packing leakoff piping and lantern rings and replaced braided asbestos packing with graphite packing. More rings of replacement packing had to be used to replace the originally installed packing. A review of PMRs 1613 and 1635 by the SSOMI team found that the installation of the new packing material was acceptable, and the team noted that a supplemental valve repacking instruction was attached to the WRs which implemented the requirements of PMR 1635 for packing replacement. The failure to annotate WR 4430-86 to cross-reference the above PMRs for documentation completeness is considered a weakness.

GENERAL DISCUSSION OF FINDING:

The concern was that in the work instructions for implementing PMR 1363, reference Work Request #04430-86, states to repack the valve in accordance with the technical manual, which states that nine (9) rings of packing needs to be installed. According to Work Request #04430-86, twelve (12) rings of packing were installed. PMR 1635, changing valve packing to a non-asbestos material, also allowed the removal of the lantern ring, which in turn, made more room for packing in the stuffing box. The concern expressed by the SSOMI team was that PMR 1635 was not referenced on Work Request #04430-86.

RESPONSE TO THE FINDING:

During the implementation process of PMR 1635, packing is fabricated by the mechanics using dies and ribbon type packing. Numerous valves have been repacked in the plant with only one known failure. The generic instruction sheet used for repacking gives a stuffing box dimension. The number of rings is not considered as important as the inside and outside diameter of the packing ring. The quantity, as specified on drawings, is considered more of a purchasing number than application number. For example: if a valve is repacked and there is room for more packing or not enough room for the packing as stated by the drawing, the mechanic is going to pack the valve using good maintenance practices.

It should also be noted that packing is a consumable item and as the service life of the valve increases, more packing will be installed depending on valve performance, not the number of packing rings specified on a drawing.

It should also be recognized that PMR 1635 changed the thought process of repacking a valve. Asbestos type packing, as specified on most drawings, is very seldom if ever used.

CONCLUSION:

Valve BG-FCV121 was repacked on November 11, 1986 on Work Request #52181-86. The leak off line was removed, (PMR 1613 on October 5, 1987). This PMR has a requirement that before the leak off line can be removed, the valve must be packed with graphoil packing; which was completed by work Request #52181-86. The valve trim change out was completed on October 9, 1987 per Work Request #04430-86. All work was performed per the direction given in their respective PMR's and by using good maintenance practice. Therefore, it is felt that no corrective actions or changes to the program need to be performed.

3.2.2.9 PRESSURIZER SAFETY VALVE TESTING

FINDING:

The SSOMI team reviewed the ASME Code Section XI testing of the Pressurizer safety valves. The test procedure and the implementation of the test procedure were inadequate because the test procedure did not use the representative temperature of the valve when installed in the system as required by TS 3/4.4.2.2. Additionally, because the "as-left" valve temperature was not recorded for valves which were tested and reset, the TS requirements could not be substantiated.

Three direct acting Code Safety valves made by the Crosby Valve and Gage Company were installed on the Pressurizer. ASME Section XI, Article IWV-3500, required that the valves be tested on a rotating basis so that each valve is tested at least once every five years. This five year cycle for valve testing began with facility commercial operation.

As a result of NRC Region IV concerns with the adequacy of the testing of the safety valves, the licensee conducted setpoint testing of the three installed valves (BB-8010A, BB-8010B, and BB-8010C) during the current outage. The licensee elected to test the valves by the use of a local bench test because of dissatisfaction with the contractor which had previously tested the Main Steam safety valves. The pressurizer safety valves were previously tested by the valve vendor between 1977-1978 and had not exceeded their ASME Code testing periodicity requirement. Additionally, three spare valves had been tested in March 1987. The inspector reviewed the detailed test procedures and data and identified the following concerns.

a. Five of the six valves tested had setpoints well below the minimum TS limit of 2461 psig. Additionally, the "C" valve setpoint was found to be below the Pressurizer Power Operated Relief Valves (PORVs) setpoint, although the licensee indicated that this safety valve had not lifted during a prior plant transient which had caused the PORVs to lift.

   The valves were bench tested in accordance with a method acceptable by the ASME Code, using a low volume, high pressure test rig. The valves were variously heated to simulate ambient installed conditions using either heat blankets or a "hot box" equipped with electrical heating elements. As corrective action the "B" and "C" valves were replaced with spare safety valves and the "A" valve was reset and reinstalled.

b. The SSOMI team also noted that the safety valves exhibited greater than expected setpoint deviation with respect to temperature variations. Diagnostic testing of the safety valves, performed as a result of discrepancies identified in the "B" and "C" valve testing, indicated that the valve setpoint dropped about 0.87 psig per degree Fahrenheit (F) increase. The diagnostic testing was performed at various ambient temperature conditions ranging from 80 to 175 degrees F and various nozzle ring settings. The nozzle ring adjusts the valve lift sensitivity. The licensee performed an informal statistical evaluation of this data and found the temperature setpoint relationship to be linear. A review of the

Electric Power Research Institute (EPRI) test data for similar valves found the temperature setpoint shift observed at WCGS to be two or three times and the maximum inferred from the EPRI data.

c. The temperature conditions used to test the installed valves were different from the temperature conditions used to test the spare valves. The installed valves were tested per procedure STS MT-005, "Pressurizer Code Safety Valve Operability," Revision 1, which specified that the valves be heated to 120 to 200 degrees Fahrenheit (F) to simulate the ambient condition as required by the TS. The spare safety valves were tested by detailed test instructions based on the Crosby Valve Manual which specified that the valves be heated to 120 to 140 F. Additionally, test temperature data was not required nor collected for the valves that were tested and reset and the licensee could not substantiate that the test temperatures represented the "as-installed" valve ambient conditions as required by TS 3/4.4.2.2.

d. NPE had not been involved in the evaluation of data nor the determination of the correct test temperatures. As a result of the SSOMI team concerns, the Maintenance Department issued EER No. 87-BB-21 on November 16, 1987, requesting specification of a valve test temperature range which would satisfy the TS requirements. The disposition to this EER, issued on November 18, 1987, provided a temperature range of 70 to 120 degrees F and direction for obtaining temperature measurements. The SSOMI team considered that the EER disposition was unsatisfactory because the temperature recommended by NPE did not have a correlation with the actual installed conditions.

e. Pressurizer safety valves are equipped with guide and adjusting rings which control the valve blowdown. The ASME Code and associated test requirements assume that guide ring settings established during vendor certification testing result in repeatable valve configurations which do not require periodic testing. Therefore bench testing, which does not change the ring settings, is allowed even though it does not verify actual blowdown performance. During the inspection, the SSOMI team found that the ring settings for the spare Pressurizer safety valves were not set as required by the manufacturer to ensure a proper valve blowdown characteristic. Furthermore, the guide ring position had not been verified on the installed Pressurizer safety valves, and the testing procedures did not include steps to verify guide ring position whenever the Pressurizer safety valves were tested.

f. The licensee failed to evaluate the information contained in I&E Information Notice 86-05, "Main Steam Safety Valve Test Failures and Ring Setting Adjustments," and I&E Information Notice 86-05, Supplement 1, which identified valve performance problems resulting from improperly established guide ring settings. Incorrect ring settings had been found to result in insufficient relief capacity and have possibly contributed to premature valve operation and/or reseating failures. Although the I&E Information Notice addressed Main Steam safety valves and not the Pressurizer safety valves, the Pressurizer safety valves at WCGS are essentially identical in configuration to the Main Steam safety valves and should have been evaluated.

GENERAL DISCUSSION OF FINDING:

a., b., c., e.

Four maintenance items were identified concerning the pressurizer safety valve testing.

1) 3.2.2.9.a - 5 of 6 valves tested below the setpoint.

2) 3.2.2.9.b - Temperature seemed to have more of an impact on setpoint than would be expected.

3) 3.2.2.9.c - Test temperatures used had no correlation with ambient temperatures at the pressurizer.

4) 3.2.2.9.e - The guide ring and blowdown ring settings were not verified.

d. As a result of questions concerning the ambient temperature conditions of the Pressurizer Code safety valves during power plant operation and the temperature influence with regard to setpoint acceptance testing, Engineering was requested to provide the temperatures to be utilized and where physically on the valve to monitor that temperature during testing. EER 87-BB-21 was dispositioned providing an ambient temperature range of 70° to 120°F. This temperature range was based on an evaluation of normal operating containment conditions judged to be reasonably accurate in the immediate area of the Pressurizer Code Safety valves. Locations for temperature monitoring on the valves during setpoint testing were also provided.

f. Although the Pressurizer Code safety valves were not the subject of I&E Information Notice 86-05, "Main Steam Safety Valve Test Failures and Ring Setting Adjustments" and I&E Information Notice 86-05 Supplement 1, the pressurizer code safety valves are similar and should have been evaluated.

RESPONSE TO THE FINDING:

a. b., c., e.

1) The three pressurizer code safety valves removed from the system, were tested using a low volume, high pressure rig, utilizing a heat blanket to simulate operational ambient temperatures stated in the Westinghouse equipment specification. The test required the valve to lift (pop) at 2485 ± 1% psi. Of the three valves removed from the pressurizer, the first tested within tolerance and the other two tested below the Technical Specification limit. Three replacement valves were tested in a similar fashion with additional criteria, and two of these valves tested out of tolerance.

Of the two valves removed from the pressurizer that tested low, it was found that the mechanic who tested these valves believed that when the valves hissed or chattered indicated the setpoint, as found when testing relief valves. The mechanic did not know that a safety valve

would fully lift when test pressure reached its setpoint. Additionally the test equipment was being improperly used. Further testing of one of these valves proved that the valve would operate as designed.

The replacement valves were tested with the same low pressure rig, but were heated with radiant heat in a hot box to a lower temperature more representative of actual ambient operating temperature. Each valve was then required to lift two consecutive times in the test range. All three pressurizer code safety valves were replaced with successfully tested valves.

2) WCNOC believes that the 0.87 pound decrease per degree of temperature is invalid based on the inconsistencies in test parameters, (i.e. method of temperature control and length of time to reach equilibrium).

3) The test temperature of 120° - 200°F was taken from the Westinghouse design specification. The procedure was written with these values, during Start-up of the plant. The spare valves were tested between 120° - 140°F. This value was not from the technical manual but a value chosen which fits within the criteria of 120° - 200°F.

The three valves installed were installed utilizing detailed work instructions rather than procedure, allowing more specific details.

4) If the guide rings are not adjusted during testing, verifications were not needed. In the case of the spare valves, the guide rings were adjusted and placed back to what the Crosby Testing Report specified.

d. The temperature range of 70°-120°F, provided for setpoint valve testing, is considered adequate for the following reasons:

1) Containment atmospheric conditions are designed to be maintained less than 120°F. The location of the Pressurizer Code Safety valves from the pressurizer (11 feet) is sufficient to preclude temperatures of the valve to be excessively greater than that of the general area.

2) The dominant factor that may vary the relieving pressure setpoint in regard to temperature is the relief valve spring. The location of the spring in respect to the process fluid makes the temperature of the spring more closely resemble the general area temperature.

3) Subsequent temperature monitoring of the Pressurizer Code Safety valves during power operation during March 1988 revealed valve surface and spring temperatures ranging from 77° to 88°F. This data confirms the temperature range provided in the Engineering Disposition to EER 87-BB-21.

4) Adequate compensation should be given to operating temperature conditions during the set pressure testing of safety relief valves. The Engineering Disposition to EER 87-BB-21 addressed ambient temperature conditions. Inlet pipe temperature conditions should also be considered as to its applicability or influence on the test results and the acceptance of the test. Engineering concluded that

the contribution that the inlet pipe temperature has in relation to that the spring imposes is negligible.

f. The pressurizer safety relief valves are designed to limit primary system pressure excursions following anticipated operational and accident transients. Operability of the safety valves was demonstrated by prototypical testing and appropriate analysis in accordance with NUREG-0731 II.D.1. The type of safety valve used at Wolf Creek, Crosby Model HB-BP-86 6M6, was tested in an EPRI test program.

The ring settings used for the Crosby 6M6 valves at Wolf Creek approximate those used in a series of EPRI tests on the Crosby 6M6 valve, where "reference" settings were used. Results from this series of tests are considered applicable to the plant valves.

The safety valves are required to operate over a range of full pressure steam, steam-to-water transition, and subcooled water fluid conditions. The valves were tested for the range of required conditions in the EPRI test program. The acceptance of this test program and plant specific results by the NRC was documented in a letter from P. O'Connor to B. Withers, dated August 6, 1987.

ACTIONS WHICH HAVE BEEN OR WILL BE TAKEN:

a., b., c., e.

1) The operability test procedure for the pressurizer code safety valves was revised January 26, 1988 to reflect the following:

The procedure now includes a statement about the lifting characteristics of a safety valve, minimizing test performance error.

Heating requirements of the valve for testing have been changed to reflect more accurate representation of actual ambient temperatures. Using radiant heat, a more even heat soak is obtained, and lessens thermal gradients throughout the valve.

The test procedure now requires two successful consecutive pops within the allowed tolerance. This should verify the accuracy of the set pressure.

2) None deemed necessary

3) Procedure STS MT-005, has been changed for a heat box to be used rather than heat blankets. The box supplies a uniform heating source for the valve, making a consistent temperature across the entire valve. A step was added to the procedure to record the temperature of the valve at the time the valve is tested.

4) Three new safety valves were installed on the pressurizer on Work Requests 91050-87, 03999-87, and 04300-87. The guide ring settings were verified to be correct per Crosby Test Data Sheet. The three valves removed will be disassembled, inspected, reassembled and tested on Work Requests 70654-87, 70705-87 and 70655-87. At that time the guide rings

will be verified as being correct per the Crosby Test Data Sheets. The only time the guide rings need to be verified as being set correctly is when the rings need changed during the course of work.

CONCLUSION:

a., b., c., e.

Procedure STS MT-005 was first used during Refuel II. Though the procedure satisfied the requirements of the Technical Specifications, the methodology was lacking in direction. There has been a complete procedure rewrite, incorporated on a temporary procedure change, and a permanent procedure change is being written.

The items of concern expressed on the testing of the pressurizer safety valve were resolved on Work Requests #91050-87, #04300-87, and #03999-87, with detailed work instruction. The work instructions were then used to develop a new procedure and work instructions for performing inspection and testing of the three removed valves on Work Requests #70654-87, #70705-87, and #70655-87.

Temperature data is being taken in the plant at normal operating pressure and temperature to further define the actual ambient temperatures that are experienced by the safety valves. This data, when completed and evaluated, will be incorporated into the procedure.

d.,f.

Based on the above discussions, WCNOC does not consider this item to be a deficiency.