EXPANDED AUGMENTED SYSTEM REVIEW AND TEST PROGRAM
(EXPANDED ASRTP)

EVALUATION

OF THE

REACTOR COOLANT

SYSTEM

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1.0 INTRODUCTION

The Rancho Seco Expanded Augmented System Review and Test Program [ASRTP] evaluation effort involves an assessment of the effectiveness of the System Review and Test Program [SRTP] and an analysis of the adequacy of ongoing programs to ensure that systems will continue to function properly after restart. The Expanded ASRTP is a detailed system by system review of the SRTP as implemented on 33 selected systems and an in-depth review of the engineering, modification, maintenance, operations, surveillance, inservice testing, and quality programs. It also conducts a review, on a sampling basis, of many of the numerous ongoing verification and review programs at Rancho Seco.

Six multi-disciplined teams composed of knowledgeable and experienced personnel are tasked with performing the Expanded ASRTP. Each multi-disciplined team consists of dedicated personnel with appropriate backgrounds to evaluate the operations, maintenance, engineering, and design functional areas. Independence, perspective, and industry standards provided by team members with consultants, architect engineer and vendor backgrounds are joined with the specific plant knowledge of SMUD team members.

Each team performs an evaluation on a selected system using the same fundamental evaluation techniques employed by the NRC in the ASRTP inspection. System Status Reports are used as the primary source of leads for the teams. They are augmented with refererces to available source and design bases documents as needeo. Team synergism and communication is emphasized during the process in order to enhance the evaluation. Each team prepares a report for each completed selected system evaluated. This report is for the Reactor Coolant system.

2.0 PURPOSE

The objectives of the Expanded ASRTP evaluation are to (1) assess the adequacy of activities and systems in support of restart and (2) evaluate the effectiveness of established programs for ensuring safety during plant operation after restart.

3.0 SCOPE

To accomplish the first objective, the Reactor Plant System team evaluated the Reactor Coolant System to determine whether:

1. The system was capable of performing the safety functions required by its design bases.

2. Testing was adequate to demonstrate that the system would perform all of the safety functions required.

3. System maintenance (with emphasis on pumps and valves) was adequate to ensure system operability under postulated accident conditions.

4. Operator and maintenance technician training was adequate to ensure proper operations and maintenance of the system.

5. Human factors relative to the system and the system's supporting procedures were adequate to ensure proper system operations under normal and accident conditions.

To accomplish the second objective, the team reviewed the programs as implemented for the system in the following functional areas:

1. Systems Design and Change Control

2. Maintenance

3. Operations and Training

4. Surveillance and Inservice Testing

5. Quality Assurance

6. Engineering Programs

The team reviewed a number of documents in preparation for and during the Expanded ASRTP evaluation. This list of documents is found in Attachment 6.1.

The primary source of leads for the team were the problems identified in the Reactor Coolant System Status Report. Various source documents such as the USAR and Technical Specifications and available design bases documents were reviewed as needed to augment the information needed by the team.

The evaluation of the Reactor Coolant System included a review of pertinent portions of support systems that must be functional in order for the Reactor Coolant System to meet its design objectives.

4.0 OVERALL RESULTS AND CONCLUSIONS

The more significant issues identified pertaining to the adequacy of the SRTP and the effectiveness of programs to ensure continued safe operations after restart are summarized below. The summary focuses on the weaknesses identified during the evaluation. Attachment 6.3 provides detailed findings by providing the Request for Information (RI) forms that are used by the Expanded ASRTP teams to identify potential concerns during the evaluation. Section 5.0 lists the specific concerns identified by the teams. The numbers in brackets after each individual summary or concern refer to the corresponding RIs in Attachment 6.3.

4.1 Six of the concerns that generated RIs in the evaluation of the RCS were concerns about the Reactor Coolant Pumps (RCPs), RCP motors, or support components such as RCP Oil Lift Pumps. These concerns addressed RCP seals, overcurrent protection, intermittent short to ground, failure to pass a periodic test, and instantaneous trip settings for the RCPs. These six concerns were in addition to the thirteen problems already identified in the RCS SSR that pertained to the RCPs. The large number of problems and concerns associated with RCPs raises questions about their ability to function.

The team feels, however, that the functionality of the RCPs is not jeopardized. No one problem or combination of problems, to the extent known, appears sufficient to cause the RCPs to fail. The one concern that appears to have the strongest potential to cause the plant to shutdown after re-start is the decision not to change the RCP seals.

Rancho Seco has had good performance from RCP seals over the past several years, especially since changing from the 850 seals to the 875 seal package on three of the four RCPs. The potential problem, however, is that Bingham has no performance data on seals that have been in service for the long time period that two of the seals have been installed at Rancho Seco. The age of the seals coupled with the plant extended outage over the last 21 months cause their reliability to be unpredictable. Consequently, Bingham recommends seal replacement. (RI-222) (RI-232) (RI-245) (RI-259) (RI-274) (RI-277)

4.2 The Loose Parts Monitoring (LPM) Program does not appear to address NRC Reg. Guide 1.133 or incorporate applicable industry operating experience. Reg. Guide 1.133 requests the licensee to appropriately upgrade their equipment and program, to document their program, and to respond to the NRC on the program within six months. We could not determine that the District responded to this request. Also, there is some concern that the sensors for the LPM are not "radiation hardened" as specified by Rancho Seco (also addressed by Reg. Guide 1.133) and may not hold up in the system. (RI-228)

OVERALL RESULTS AND CONCLUSIONS (Continued)

4.3 Training of Quality Control (QC) Inspectors is not administered and documented in accordance with governing procedures and the training program. Several examples indicate that the QC program for ensuring their inspectors are current with training, required reading, etc., may be weak. These include:

- No evidence found that inspectors were trained to required procedures.
- Required reading completed after due date established by management.
- QC training records not forwarded to Training Department in time frame required by procedure.
- The responsibilities of the QC Training Coordinator are not defined. (RI-251)

5.0 SPECIFIC CONCERNS

A list of the specific concerns the Expanded ASRTP team believes are new concerns not previously identified for resolution follows:

5.1 Acknowledged (Valid) Concerns

5.1.1 The loose parts monitoring program does not appear to adequately address NRC Reg. Guide 1.133 or take advantage of lessons learned from industry operating experience. (RI-228)

5.1.2 Actions taken to rectify some identified System Status Report problems with respect to wiring and conduit concerns have not provided adequate solutions. (RI-237)

5.1.3 It appears that some QC inspectors are not adhering to procedures and some are not attending scheduled training. (RI-251).

5.1.4 Calculations for overcurrent protection for Reactor Coolant Pumps could not be located. (RI-259)

5.1.5 High Point Vent Valve fuses are not readily available if needed in the event of an emergency. (RI-272)

5.1.6 Instantaneous trip setting for "D" Reactor Coolant Pump is higher than the other three pumps. (RI-274)

5.2 Open (Potential) Concerns

5.2.1 There are no plans to inspect or change RCP seals prior to plant re-start even though changeout of the seals was recommended by the RCP Vendor and supported by Babcock and Wilcox. (RI-222)

5.2.2 Emergency Procedure E.07 (Inadequate Core Cooling) calls for operator action that could be dangerous for the operator unless the operator is properly trained. (RI-246)

6.0 ATTACHMENTS

6.1 List of Documents Reviewed

6.2 Status of RIs

6.3 Detailed Observations - Requests for Information

LIST OF DOCUMENTS REVIEWED

1) Updated Safety Analysis Report:
   - Chapter 8, Section 8.2
   - Section 1.4.9 criterion 9, Reactor Coolant Pressure Boundary
   - Sections 1.4, 3.2, 4.1, 4.2, 4.3, 14.1, and 14.2

2) Design Bases Document:
   - Rancho Seco RCS Design Basis Document: 51-1159639

3) System Status Report:
   - RCS-System Status Report Rev. 1; WP2930B D0154B

4) Technical Specifications:
   - 3.1.6, 2.1, 2.2, 3.1, 3.2, 3.7, 4.4, 4.9
   - 6.5.2.8 "Audits"

5) Plant Operating Procedures:
   - A.1, Reactor Coolant System
   - A.2, Rev. 16, Reactor Coolant Pumps
   - E.07
   - H2X A
   - TP-600-10
   - E 209 Sh 59 and 59A
   - A-4 Core Flooding System
   - A-3 Rev. 21, Pressurizer and Pressurizer Relief System
   - B.2 Plant Heatup and Startup
   - B.4 Plant Shutdown and Cooldown

6) System Training Manual:
   - Chapter 2, Reactor Coolant System
   - Chapter 3, Pressurizer and Pressurizer Relief System

7) Training Course Outline:
   - Reactor Coolant System; OD 21 I0600, Rev. 1

8) Administrative Procedures:
   - AP.20 Reactor Building Close-out
   - RSAP-0803, Work Request

9) Nuclear Engineering Procedures:
   - NEPM Index No. 5108.2; Application of ASME Codes
   - NEPM Index No. 5101.11; Mandatory Codes and Standards

10) Nuclear Regulatory Commission Documents:
   - I&E Information Notice 86-87
   - I&E Information Notice 86-108 and Supplement #1

ATTACHMENT 6.1

LIST OF DOCUMENTS REVIEWED (Continued)

11) Other Documents/Letters:

Bechtel Dwg C-368 Rev. 7, Reactor Support
Bechtel Dwg C-367 Rev. 8, Steam Generator Support
ITE Dwgs E5.02.11, 12, 14 and 15; E7.02-198 Rev. 4
SMUD Dwgs E-201 Sh 1, 1A, 1B and 1C
E-203 Sh 47 Rev. 2
SMUD Memo dated 6/15/87 Subject: R.C. Press. Boundary Classification
QAIP 18-Rev. 1 "Inspector Training and Qualification Requirements"
TDAP-7100 Rev. 0 "Nuclear Training Records Control"
TDAP-7200 Rev. 0 "Training Information Management System"
SMUD Memo GVC-234 dated 2/23/87 "Reactor Coolant Pump"
SD 330A Vendor Tech Manual "Spectrascope Real Time Analyzer"
Letter: John Mattimoe to NRC dated 9/4/80
B&W Letter: ESC-512 dated 6/22/87 "Control Rod Drive Mechanism"
Dresser Industries Dwg: 4CP 2398
Elementary Diagrams: E-203, Sh 13 Rev. 9; Sh 18 Rev. 7; Sh 14 Rev. 4; Sh 47 Rev. 2
NPRDS Report, Rev. 10; Allis Chalmers Larger Than 3000HP
Bingham Instruction Manual, N.6.02
CIDR #C87R
QAP 17, Rev. 5 "Nonconforming Material Control"
NCR 5864, Rev. 0
QA NCR Log
CIDR MC288
WPS M.308, "Mandatory Welding Practices"
ANSI B31.1-1980 Edition "Power Piping"
Dwg No. 21582-6" ED
Dwg M-853 Line Designation Schedule
NCR 6633
Bingham International Letter: Reactor Coolant Pump Seals, April 13, 1987
Oconee Evaluation Report for Reactor Coolant Pump B2 Vibration, December 17, 1987
Oconee Reactor Coolant Pump Events 3B2, August 1985
Bently Nevada, Field Balancing of RCP P-210C, August 26, 1983

12) Maintenance Procedures:

N1.01-IM01 (Allis Chalmers Ins't Manual)
Electrical Maintenance History File
MAP-006, Work Request Planning

LIST OF DOCUMENTS REVIEWED (Continued)

13) Surveillance Procedures:
SP.214.03 (Rev. 33) Locked Valve List
SP.17 Refueling Interval RCS High Point Vents
SP.16
SP.52 (draft)
SP.206.06 A/B
SP.15 and STP.1042, STP.771, STP.1086, STP.1093 and STP.1041
SP 209.06

14) Engineering Change Notices (ECN):

A-4615 DBR	A2592	A4352
A0777	A2932C	A4583
A0933	A2934	A4691
A1506	A3217	A4822B
A1510	A3231	A4887
A1650	A3778	A4943
A1763	A3886	R0001
A2162	A3989	R0317
R0380		

15) Pipe and Instrument Diagrams (P&ID):

M-520 Sh 1, 2 and 3	Reactor Coolant System
M-522	Decay Heat System
M-521	Purification and Letdown System

16) Work Requests (WRs):
115702, 116987, 130815, 130817, 64948, 35865, 117299, 128398, 130816, 130818, 56448, 129153, 133748, 135590, 119691, 132826

17) List of Calculations:

Z-RCS-EO 642	Z-RCS-M1315	Z-RCS-M2141
Z-RCS-IO 059	Z-RCS-M1625	Z-RCS-M2239
Z-RCS-IO 140	Z-RCS-M1740	Z-RCS-M2282
Z-RCS-IO 141	Z-RCS-M1753	Z-RCS-N0002
Z-RCS-M0497	Z-RCS-M1947	Z-RCS-N0004
Z-RCS-M0498	Z-RCS-M1996	Z-RCS-N0012
Z-RCS-N0016	Z-RCS-M2276	Z-RCS-M2277
Z-RCS-M2278	Z-RCS-M2279	Z-RCS-M1965
Z-PRL-E0048	Z-PRL-E0043	Z-PRL-E0044
Z-PRL-E0074		

18) QA Interval Audit Reports:

87-03	0-785	0-877
0-781	0-805	0-770

ATTACHMENT 6.1

19) Casualty Procedures:

C.3	Small Reactor Coolant Leak
C.7	High Activity IN Reactor Coolant
C.8	Reactor Coolant Pump Malfunctions
C.11	Pressurizer System Failure
C.14	Partial Loss of NNI
C.15	Loss of NNI 'X' Power
C.16	Loss of NNI 'Y' Power
C.17	Complete Loss of NNI
C.20	Loss of CCW
C.95	Loss of OFF Site Power

STATUS OF RIs

Attachment 6.2 provides RI status as of this report date. An RI is considered closed if the Team Leader was convinced a potential concern was not valid or not significant enough to be an RI. An RI would also be closed if requested information was provided. All other RIs are open. Acknowledged RIs are open RIs that have been accepted as valid by the responsible organization and have been stated as concerns in Section 5.0.

RI NUMBER	STATUS
222	OPEN
228	ACKNOWLEDGED
230	CLOSED
232	CLOSED
237	ACKNOWLEDGED
245	CLOSED
246	OPEN
250	CLOSED
251	ACKNOWLEDGED
257	CLOSED
259	ACKNOWLEDGED
272	ACKNOWLEDGED
273	CLOSED
274	ACKNOWLEDGED
277	CLOSED

ATTACHMENT 6.2

DETAILED OBSERVATIONS - REQUEST FOR INFORMATION

During an evaluation, all potential concerns are documented on Request for Information sheets (RIs) that are sent to the responsible organization to receive their input concerning the potential concern. RIs are also used to request information that the EASRTP team is having difficulty obtaining.

These RIs are considered drafts throughout the entire evaluation until they become part of the report. Responsible organizations can accept the potential concern as valid or they may disagree with the potential concern. If they disagree, they can submit information that convinces the EASRTP team members that the potential concern is not valid, or they may redirect the EASRTP members to better focus the concern. RIs developed during the system evaluation comprise this section of the report.

ATTACHMENT 6.3

REQUEST FOR INFORMATION (RI)

RI NO: 222 SYSTEM CODE: RCS ISSUE DATE: 9-14-87

SUBJECT: REACTOR COOLANT PUMP (RCP) SEALS

DEPARTMENT: MAINTENANCE COORDINATOR/EXT: J. DARKE/4817

TEAM LEADER/EXT: K. PRINCE/3851

POTENTIAL CONCERN/QUESTION:

There are no plans to inspect or change RCP seals prior to plant start up even though the plant has been off line for 21 months. Changeout of the seals was recommended by the RCP Vendor, and supported by Babcock and Wilcox.

Since late December 1985, the RCP seals have been subjected to long periods of dryness and may have taken a "cold set". Correspondence between the RCP Vendor, Bingham International, and Rancho Seco staff precipitated a recommendation from the Vendor to refurbish the "C" RCP seals and to upgrade "A" RCP to the 875 seal package. The recommendation was made in a letter from Bingham to Rancho Seco, April 8, 1987. We understand that exception was taken to this recommendation by some at Rancho Seco because of a reference to "shelf life" vice "active life".

On May 11, 1987 a follow up letter was received from Bingham, stating that concern was not only for "shelf life" but for a number of issues that Bingham thought would be taken care of since they (Bingham) did not expect opposition to term "shelf life". These concerns include:

- Seal faces and springs should be inspected and refurbished if required.
- Because of length of time the seals have been in pumps ("C" RCP-7 years and "A" RCP 5 or 6 years), they could not provide a basis for not changing the seals.
- Take action on other 2 RCPs based on findings on the "A" and "C" pumps.

A letter from Babcock and Wilcox to Rancho Seco on June 23, 1987 supported the vendor recommendation.

ATTACHMENT 6.3

REQUEST FOR INFORMATION (RI)

RI NO: 228 SYSTEM CODE: RCS ISSUE DATE: 9-17-87

SUBJECT: LOOSE PARTS MONITORING SYSTEM (LPMS)

DEPARTMENT: SYSTEMS ENGINEERING COORDINATOR/EXT: J. ITTNER/4701

TEAM LEADER/EXT: K. PRINCE/3851

POTENTIAL CONCERN/QUESTION:

The Loose Parts Monitoring Program (LPMP) does not appear to adequately address NRC Rdg Guide 1.133 or take advantage of lessons learned from industry operating experience.

NRC Reg. Guide 1.133, Rev. 1, was issued in May 1981 defining the NRC position on the "Loose-Part detection program for the Primary System of Light-Water-Cooled Reactors". Among other proposed requirements, the NRC stated that it was intended that the licensee document their loose parts detection program and make appropriate provisions for equipment and program revisions. A search of Rancho Seco correspondence and talks with Licensing and Maintenance did not determine that Rancho Seco responded to the Reg. Guide request. Also, Rancho Seco is changing their monthly loose parts Surveillance Procedure SP.209.06 to a variable frequency routine test RT-RCS-008.

A look at the Rancho Seco specification indicated that the sensors for the LPMs should be "radiation hardened". The vendor supplied specification sheet did not address the capabilities of the sensor and allows possibility that the synthetic crystal may deteriorate with use. The radiation hardened sensor issue is also addressed in Reg. Guide 1.133.

The Rancho Seco System Status Report (SSR) addresses four potential sources (RCS SSR 36, 71, 72 and 77) for loose parts in the RCS. Also referenced is a problem at Oconee Nuclear Station, Unit 3, where RCPs identical to the Rancho Seco RCPs released at least 200 pounds of parts into the RCS. Other sources only document that parts can and will become loose in the RCS. History of other nuclear stations document instances where lack of a good loose parts monitoring program allowed excessive damage and where use of a good program prevented excessive damage because of early detection.

ATTACHMENT 6.3

RI NO: 228 (Continued)

In summary, it appears that:

- Reg. Guide 1.133 calls for an upgraded system.
- Reg. Guide requested a response.
- Rancho Seco has not provided a response.
- Rancho Seco has down-graded the test program.
- Reg. Guide 1.133 and Rancho Seco Specification specify radiation hardened sensors.
- Vendor supplied Sensor Specification Sheet does not address radiation tolerance of sensors.

REQUEST FOR INFORMATION (RI)

RI NO: 230 SYSTEM CODE: RCS ISSUE DATE: 9-17-87

SUBJECT: PRESSURIZER RELIEF PIPING

DEPARTMENT: NUCLEAR ENGINEERING COORDINATOR/EXT: T. TELFORD/3849

TEAM LEADER/EXT: K. PRINCE/3851

POTENTIAL CONCERN/QUESTION:

Pipe supports for pressurizer piping may not be adequate since discharge subcooled liquid was not considered in the stress calculations.

The design calculation (Z-RCS-M1965 Rev. 1) for the pressurizer relief piping loads does not include the loads for discharge of subcooled liquid. These loads may be significantly higher than those for the discharge of saturated steam. This case should be included in the design basis because subcooled liquid discharge may occur during HPI cooling mode.

The calculation omitted the subcooled liquid discharge case based on a letter from Stone & Webster (P. Ervin to J. Bisset June 17, 1986) which states that, "subcooled liquid discharge is much less severe than the discharge of saturated steam' which is not true (see attached technical justification). The discharge of subcooled liquid is not properly addressed.

This RI is closed. Additional calculations that considered subcooled liquid were completed but not found in file. Calculations were in Administration Building for microfilming.

ATTACHMENT 6.3

REQUEST FOR INFORMATION (RI)

RI NO: 232 SYSTEM CODE: RCS ISSUE DATE: 9-18-87

SUBJECT: RCP OIL SYSTEM

DEPARTMENT: NUCLEAR ENGINEERING COORDINATOR/EXT: T. TELFORD/3849

TEAM LEADER/EXT: K. PRINCE/3851

POTENTIAL CONCERN/QUESTION:

The D.C. emergency oil lift pump may be unable to perform its intended design function. Presently the design is that in the event of loss of power to the common oil control scheme (BKR 72-E02) the start capability of all AC and DC oil lift pumps will be disabled. This could jeopardize the RCS system and cause equipment damage. D.C. oil lift pump by design is emergency backup to the AC pump in the event the AC system is inoperable and cannot supply the required oil pressure.

ELementary diagram (E-203 sh 13 Rev. 9) shows that there is a common start scheme for AC and DC emergency oil lift pump. The AC pump will start manually through a start pushbutton at the control room or automatically through RCP speed switch signals and arm the start circuit of the D.C. emergency oil lift pump. The DC pump will start permissive to low oil lift pump discharge pressure (PSL-21005) through time delay set at 5 secs.

- This RI is closed. Engineering response satisfied the concern.

ATTACHMENT 6.3

REQUEST FOR INFORMATION (RI)

RI NO: 237 SYSTEM CODE: RCS ISSUE DATE: 9-24-87

SUBJECT: WIRING AND CONDUIT DEFICIENCIES

DEPARTMENT: MAINTENANCE COORDINATOR/EXT: T. DARKE/4817

TEAM LEADER/EXT: K. PRINCE/3851

POTENTIAL CONCERN/QUESTION:

Actions taken to rectify some identified items of concern with respect to wiring and conduit problems have not provided adequate solutions.

The teams assessment of the System Status Report indicated that corrective actions taken to address some concerns appear inadequate. Investigation of this problem led to finding other similar discrepancies. Equipment deficiencies and problems are identified in System Status Reports, NCRs, and Work Request which are not resolved upon close out of the document.

Glass tape insulation has loose ends and appears to be coming loose from the wire leading to the Pressurizer Relief Valve Acoustic Monitor Sensor. This sensor is located on the tail pipe of the relief valve for the pressurizer. (XE21521)

This wire insulation is the subject of the "SSR" Problem #75. The resolution was to repair damaged insulation by wrapping it with glass tape. It appears that the repair is inadequate because the glass tape is not holding and is frayed.

There are identical acoustic monitors mounted on the Main Steam Safety Valves which are the subject of RI #10. The concerns on RI-10 include the inadequate installation of the sensor cables.

There are several wires, without conduit, routed across steps and grating near the top of the Pressurizer. Some of the wires were found to have damaged insulation and shielding. These wires were for the Reactor Building Vibration Sensors which have not been used since initial Plant Startup. The manner in which these wires are routed across the steps poses a safety hazard to personnel.

Also, two wires were found leading to a temperature element (TE-21179) where the conduit ends six inches from the temperature element connection. This temperature element does not have a connection box for securing the conduit and protecting the connectors.

ATTACHMENT 6.3

RI No: 237 (Continued)

A work request and a NCR (S-6633) were written to repair frayed insulation. The repair was to wrap these wires with glass tape. The glass tape is loose and the conduit has slipped such that there is approximately 6 inches of wire outside of the conduit.

It appears that the original Work Request did not properly identify the problem of the missing connection box and loose conduit. This resulted in an NCR which described the frayed wires, but did not describe the missing connection box. The NCR was dispositioned and closed out. The wires were wrapped with glass tape per the subsequent Work Request to close the NCR. The problem of the missing connection box still exists.

REQUEST FOR INFORMATION (RI)

RI NO: 245 SYSTEM CODE: RCS ISSUE DATE: 9-22-87

SUBJECT: REACTOR COOLANT PUMP STARTING CIRCUIT

DEPARTMENT: NUCLEAR ENGINEERING COORDINATOR/EXT: T. TELFORD/3849

TEAM LEADER/EXT: K. PRINCE/3851

POTENTIAL CONCERN/QUESTION:

Reactor Coolant Pump (RCP) P-201A-M intermittent short to ground which caused the pump to start with no operator action is rectified but the team found no indication that Engineering has determined the fix to be adequate.

The control scheme for RCP breakers allowed inadvertent start or stop operation of the pump without operator action due to ground fault in the related control circuit. Additional concern is the potential of other breakers closing/tripping without operator's action since the RCP breaker control is typical for most of the 125VDC breaker control in all voltage levels, i.e., 480VAC, 4160VAC, and 6900VAC, including QA Class 1 circuit breakers feeding loads required for the safe shutdown of the plant.

The problem was reported and identified through Work Request No. 64948 for Maintenance corrective action. Per the work request, an intermittent ground was traced to lower bearing oil level switch LSL-21145. An interview with Supervising Electrical Technician determined that the RCP P-210A-M started without operator's action at least three (3) additional times, twice when the breaker was placed in test position. The result of Electrical Technician shop investigation indicated that ground at level switch LSL-21145 impressed approximately 65VDC across the breaker closing circuit enough to operate the breaker inadvertently without operator's action.

The problem was rectified by isolating the ground fault but there is no evidence of engineering study to analyze the breaker control scheme that will determine any modifications/actions required to prevent recurrence.

- This RI is closed. Engineering approval of actions taken are satisfactory.

REQUEST FOR INFORMATION (RI)

RI NO: 246 SYSTEM CODE: RCS ISSUE DATE: 9-22-87

SUBJECT: CIRCUIT BREAKER OPERATION SYSTEM (RCS)

DEPARTMENT: TRAINING/OPERATIONS COORDINATOR/EXT: F. THOMPSON/4115

R. MACIAS/4589

TEAM LEADER/EXT: K. PRINCE/3851

POTENTIAL CONCERN/QUESTION:

Emergency Procedure calls for operator action that could be dangerous unless the operator is properly trained.

- Review of Emergency Procedure E.07 (Inadequate Core Cooling) indicates that if a RCP circuit breaker does not close remotely, the operator should check the spring loaded mechanism, and if it is charged, manually close the circuit breaker by pulling the lever at the bottom of the breaker.

- Checking the indicator as to whether or not the spring is charged is not sufficient to determine if the breaker can be closed manually. A loose part in the circuit breaker mechanism could cause a breaker not to close remotely and if closed manually, the loose part could fall and cause a short circuit which could cause the breaker to blow and cause harm to the operator.

- A training procedure for the operators on circuit breakers operation and how and what should be tested was not found.

REQUEST FOR INFORMATION (RI)

RI NO: 250 SYSTEM CODE: RCS ISSUE DATE: 9-24-87

SUBJECT: REACTOR VESSEL ANCHOR BOLTS

DEPARTMENT: NUCLEAR ENGINEERING COORDINATOR/EXT: T. TELFORD/3849

TEAM LEADER/EXT: K. PRINCE/3851

POTENTIAL CONCERN/QUESTION:

Material used in the fabrication of the Reactor Vessel (RV) skirt fasteners has not been determined and may be susceptible to SCC.

RCS-SSR Problem #13 addressed the potential of stress corrosion cracking (SCC) in RV anchor bolts as a generic issue. A Preliminary Safety Concern (PSC-9-81) was analyzed by B&W addressing the generic concern which involved the specific fastener material SA520. B&W did not provide Rancho Seco's RV anchor bolts and did not evaluate the Rancho Seco bolts.

Action List Item Number 20.0007 (RCS-SSR Problem #13 equivalent) dispatched as acceptable the Rancho Seco RV anchor bolt problem by stating Rancho Seco did not have SA540 material. Source of this information was Dwg C-368 Rev. 7, which shows Rancho Seco has A193-B7 material. The issue of potential SCC on A193-B7 material was not addressed. Further, a search of files attempted to find the Receiving Inspection Data Report (RIDR) for these studs, and none was found. In addition, the torque value was specified by B&W without knowing what material was used at Rancho Seco. Torque values are important in determining probability of SCC.

Without identification of the RV anchor bolt material and knowledge of pre-load torque, the bolts are suspect for SCC.

- This RI is closed. Material search indicates that bolt material is A193-B7 although receiving paperwork has not been located.

ATTACHMENT 6.3

REQUEST FOR INFORMATION (RI)

RI NO: 251 SYSTEM CODE: GENERIC ISSUE DATE: 9-23-87

SUBJECT: TRAINING OF QC INSPECTORS-TRAINING RECORDS

DEPARTMENT: QUALITY ASSURANCE COORDINATOR/EXT: J. SULLIVAN/4585

TEAM LEADER/EXT: K. PRINCE/3851

POTENTIAL CONCERN/QUESTION:

It appears that the procedures and training program for QC Inspectors is inadequate or not followed.

Training of Quality Control (QC) inspectors in not administered and documented as required by QAP-18, Rev. 16. Refresher training of inspectors to revised procedures appears inadequate:

Training Records on file at the QC office were reviewed for four level II QC Electrical Inspectors. There was no evidence by assigned reading or training attendance records to demonstrate that the inspectors received training to QAP-17, "Nonconforming Material Control" or QAP 18, "Quality Assurance Records".

The method of providing refresher training to QC inspectors does not appear to be performed in an adequate and timely manner. For example, an Assigned Reading routing slip was sent to all inspectors for refresher training to RSAP-0803, Rev. 1. By direction of management, the reading was to be completed by 7/28/87. The slip was routed 7/16/87 but assigned reading was not completed until 9/3/87.

QC training records are not forwarded by the QA Department Training Coordinator to Nuclear Training (Main Frame) within 30 days of completion of the training. This is required by TDAP 7100. Training records on file at the QC office indicate that QC Inspectors received training in requirements of QAP-16, Rev. 1, during March and April, 1987. These records have not been received by the Nuclear Training Department.

Apparently, QC clerical personnel who have responsibility for record turnover to NTD are not familiar with TDAP 7100 requirements. Also, QA has recently appointed a Training Coordinator responsible to forward records to training, but is not doing so. The Training Coordinator is not able to devote time to forwarding records because of his additional responsibilities. Also, the Training Coordinators responsibilities are not defined in any of the QA Procedures.

ATTACHMENT 6.3

REQUEST FOR INFORMATION (RI)

RI NO: 257 SYSTEM CODE: GENERIC ISSUE DATE: 9-23-87

SUBJECT: INADEQUATE WORK REQUEST HISTORY

DEPARTMENT: MAINTENANCE COORDINATOR/EXT: J. DARKE/4817

TEAM LEADER/EXT: K. PRINCE/3851

POTENTIAL CONCERN/QUESTION:

The team's assessment of work request history indicated that the "On Line" work request history file does not contain important information about previous recurring problems with plant equipment.

The NUCLEIS on line history file is a new system that will track the reoccurring problems, but this system has been in service for about 3 months and there is no plan to update the information on previous work requests. A back log of more than 20,000 completed work requests contains pertinent information that could indicate the root cause of recurring problems.

In one case noted by the EASRTP audit of the Decay Heat System, there was a class 1 check valve (DHS-003) which had a recurring gasket leak. There were only 4 work requests referenced in the history file, yet 16 documents were found on microfilm that referenced gasket leaks on this valve. One NCR indicated that this valve has a manufacturer's defect in the depth of the gasket groove. However, this information is not available in the work request history file for the Nucleus system.

Administrative Procedure AP.42, Rev. 5 does not mention the Nucleus system. It also states in Section 3.3 that the entire equipment history for a particular piece of equipment is available by obtaining work request information which occurred within the quarter from the on line system, and information older than that from the microfiche.

There are cases where work was performed on equipment and the work request information was entered in the history file using a number other than the equipment ID. This was done because the equipment was not listed in the MEL.

- This RI is closed. Required data is available, but from five (5) sources. Action #207 is written to provide estimate of costs to incorporate MIMS Closed any Work Request History (PM/CM) into NUCLEIS.

ATTACHMENT 6.3

REQUEST FOR INFORMATION (RI)

RI NO: 259 REV 1 SYSTEM CODE: RCS ISSUE DATE: 9-24-87

SUBJECT: RELAY CALCULATION AND SETTING

DEPARTMENT: NUCLEAR ENGINEERING COORDINATOR/EXT: T. TELFORD/3849

TEAM LEADER/EXT: K. PRINCE/3851

POTENTIAL CONCERN/QUESTION:

There is no evidence of reactor coolant pump motor overcurrent protection calculations. Also relay coordination curves cannot be found.

Drawing E-1011-SH85 and 86 indicates a tabulation of relay settings for the 6.9 KV BVS including the reactor coolant pump motors. However, there is no reference to any motor overcurrent protection calculations as a basis for the relay settings.

The description of drawings E-1011 Sh 85 and 86 under "Equipt. No." and "Service" are incorrect. Correct readings are as follows:

Sheet 85

Bkr. No.	Equipt. No.	Service
6A01	RCP BUS 6A	Normal Feed
6A04	RCP BUS 6A	Alternate Feed

Sheet 86

Bkr. No.	Equipt. No.	Service
6B01	RCP BUS 6B	Normal Feed
6B04	RCP BUS 6B	Alternate Feed

It is suggested that PAG consider this concern a "valid covered" concern and incorporate it with the resolution of the generic issue identified in RI-175.

ATTACHMENT 6.3

REQUEST FOR INFORMATION (RI)

RI NO: 272 SYSTEM CODE: RCS ISSUE DATE: 9-28-87

SUBJECT: RCS HIGH POINT VENT VALVE FUSES

DEPARTMENT: OPERATIONS/MAINTENANCE COORDINATOR/EXT: R. MACIAS/4589
J. DARKE/4181

TEAM LEADER/EXT: K. PRINCE/3851

POTENTIAL CONCERN/QUESTION:

Adequate controls are not in place to assure that the RCS High Point Vent Valve Fuses will be readily available if needed in the event of an emergency.

The fuses are removed to meet Appendix R requirements and by procedure, are stored in the cabinet. However, there is no holder for the fuses and they are loose on the bottom of the cabinet. There is no identification to indicate their purpose and no routine checks to verify availability when needed. In a recent check, the fuses could not be found in cabinet H3RPB1. The fuses are required by Emergency Procedure EOP E.07 (Inadequate Core Cooling).

It is suggested that PAG consider this concern a "valid covered" concern and incorporate it with the resolution of the generic issue identified in RI-035.

ATTACHMENT 6.3

REQUEST FOR INFORMATION (RI)

RI NO: 273 SYSTEM CODE: GENERIC ISSUE DATE: 9-28-87

SUBJECT: TRAINING-QUALITY CONTROL ELECTRICAL INSPECTORS

DEPARTMENT: QUALITY ASSURANCE COORDINATOR/EXT: N. SAMSON/3965

TEAM LEADER/EXT: K. PRINCE/3851

POTENTIAL CONCERN/QUESTION:

A review of Four Level II Quality/Control Electrical Inspectors Training Records revealed that they have not received training to all of the Quality Assurance procedure policy sections, Quality Assurance procedures and QAIPs as required by QAIP 18 Rev. 1 "Inspector Training and Qualification Requirements".

Training records on file at QC and at the L and E Building indicate that not all inspectors have been provided training to QAP Policy Section XV, Rev. 0 (Nonconforming Material Control), QAIP 15, Rev. 0 (Electrical System Upgrade Program), QAP 18, Rev. 3 (Quality Assurance Records), or any of the previous revisions. Review of records indicates that two of the inspectors did not receive training to the requirements of QAP 6 Rev. 0 through Rev. 4 (QC Inspections).

This RI is closed. Already covered in RI-251.

REQUEST FOR INFORMATION (RI)

RI NO: 274 SYSTEM CODE: RCS ISSUE DATE: 9-28-87

SUBJECT: RCP-D OVERCURRENT RELAY SET POINT

DEPARTMENT: MAINTENANCE COORDINATOR/EXT: J. DARKE/4817

TEAM LEADER/EXT: K. PRINCE/3851

POTENTIAL CONCERN/QUESTION:

The plant relay setting sheet, which records setting of relays for the Reactor Coolant Pump D dated 6/15/87, does not conform to the load design data sheet E-1011, Sh 86, Rev. 10.

The instantaneous trip setting, per data sheet E-1011, is required to be set at 40 amps. However, it is recorded as set at 42 amps. All four RCP motors have the same Westinghouse type COM-11 overcurrent relays. Three of the motors have their instantaneous device set at 40 amps.

It is suggested that PAG consider this concern a "valid covered" concern and incorporate it with the resolution of the generic issue identified in RI-171.

ATTACHMENT 6.3

REQUEST FOR INFORMATION (RI)

RI NO: 277 SYSTEM CODE: RCS ISSUE DATE: 9-28-87

SUBJECT: MAINTENANCE EQUIPMENT TEST (FAILURE ANALYSIS AND DOCUMENTATION)

DEPARTMENT: MAINTENANCE COORDINATOR/EXT: J. DARKE/4817

TEAM LEADER/EXT: K. PRINCE/3851

POTENTIAL CONCERN/QUESTION:

Reactor Coolant Pump (RCP) Motors failed polarization index test but there is no indication that additional testing was performed.

Work Request No.s 130815, 130816, 130817, and 130818 were issued to perform megger test for Reactor Coolant Pump motors P-210A-M, P-210B-M, P-210C-M, and P-210D-M respectively. Test failed to meet the acceptance criteria for polarization index of not less than 2 in accordance with EM-125, Rev. 5. Maintenance Engineer review noted that the test is invalid because motor surge capacitors were left connected during the test. However, there is no indication in the work request package to show that a follow up or corrective action will be performed to rectify the problem. Work Requests were closed in the interim.

- This RI is closed. Maintenance Engineer recognized that surge capacitors were left connected and made allowances which brought test into acceptance specifications.

ATTACHMENT 6.3