APPENDIX

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Inspection Report 50-285/92-21 Operating License: DPR-40

Docket: 50-285

Licensee: Omaha Public Power District
444 South 16th Street Mall
Omaha, Nebraska 68102-2247

Facility Name: Fort Calhoun Station

Inspection At: Blair, Nebraska, and Huntsville, Alabama

Inspection Conducted: August 24 through September 3, 1992

Inspectors: P. Goldberg, Reactor Inspector, Plant Systems Section, Division of Reactor Safety

D. Hunter, Senior Reactor Inspector, Operational Programs Section, Division of Reactor Safety

P. Wagner, Team Leader, Operational Programs Section, Division of Reactor Safety

Approved: Thomas F. Stetka 9/11/92
Thomas F. Stetka, Chief, Operational Programs Section, Division of Reactor Safety Date

Inspection Summary

Areas Inspected:

A special inspection was conducted from August 24 through September 3, 1992, of the events related to the premature opening of Pressurizer Safety Valve RC-142 on August 22, 1992. The inspection included the evaluation of the initiating event, the plant and personnel response to the event and the actions implemented to preclude recurrence of the event. The inspectors also evaluated the actions implemented to verify proper operation of the affected systems and components, and the post-event valve testing activities conducted onsite and at the Wyle Laboratory.

Results:

The inspectors found the actions of the reactor operators in the immediate response to the event to have been good. The inspectors determined that all systems and components operated as expected throughout the event with the exception of Pressurizer Safety Valve RC-142 opening prematurely.

The actions implemented by the licensee in response to the event included modifications to the electrical power system for the turbine generator's electrohydraulic control system and lowering the high pressurizer pressure reactor trip setpoint and the pressurizer power operated relief valves' setpoint to minimize challenges to the pressurizer safety valves. The licensee also conducted extensive pressurizer safety valve testing both onsite and at the testing laboratory to establish valve operating characteristics.

The inspectors determined that the modifications to the electrical power system for the electrohydraulic control system were implemented in a timely manner and would improve the reliability of the facility.

The licensee's pressurizer safety valve testing program was considered to be noteworthy. The licensee was considered to be proactive due to the in-situ testing which resulted in the determination of normal plant operating temperatures of the valves. The test program at the Wyle Laboratory was comprehensive in scope. The inspectors found the licensee's engineering analysis, EA-92-067, to be comprehensive in scope and well engineered.

The inspectors found the licensee's analyses satisfactory in that it provided an appropriate basis for the revision to the high pressurizer pressure reactor trip and pressurizer power operated relief valve setpoints. The inspectors also found the prompt implementation of the modification to readjust those setpoints to be commendable.

No violations or deviations were identified during the inspection.

ATTACHMENTS:

- Attachment A - Persons Contacted and Exit Meeting
- Attachment B - EHC System Power One-Line Sketch
- Attachment C - Safety Valve Thermocouple Locations
- Attachment D - Licensee Presentation Handout

1 INTRODUCTION (93702)

The licensee reported that a reactor trip had occurred at the Fort Calhoun Station on August 22, 1992. The trip had been initiated because of problems with the turbine generator's electrohydraulic control system and a pressurizer safety valve. Because similar initiating events had occurred on July 3, 1992, the NRC responded with this special inspection effort. This event differed significantly from the earlier event, however, because there was no uncontrolled loss of reactor coolant into the containment building as had occurred during the July 3 event.

2 EVENT DESCRIPTION

The event descriptions provided below were derived from the operational records and from interviews with various licensee personnel. The first section provides a general overview of the event. The second provides a detailed sequence of events.

2.1 General Description of the Event

On August 22, 1992, the licensee experienced a failed power supply for two pressure transmitters in the control circuitry for the main turbine. When power was lost, the circuitry caused the main turbine control valves to ramp in the closed direction because the loss of power to the circuitry indicated a lower steam flow demand.

When the turbine control valves closed, the heat removal from the reactor coolant system was reduced which resulted in a pressure increase. The pressure increased to just less than 2400 psia when Pressurizer Safety Valve RC-142 prematurely opened and caused a rapid decrease in the reactor coolant system pressure. Since the setpoints for the high pressure reactor trip and the power operated relief valves was 2400 psia, these setpoints were not reached because the opening of RC-142 mitigated the pressure increase. The decreased pressure coupled with a high reactor power signal (the plant had been operating at near full power prior to the event) was sensed by the reactor protective circuitry which initiated reactor and turbine trips. The operators implemented the requirements stated in the emergency operating procedures and the plant was placed in a safe shutdown condition.

All of the plant systems and components functioned as designed in response to the failed power supply in the electrohydraulic control system with the exception of Pressurizer Safety Valve RC-142 which opened at a pressure of about 2397 psia instead of its setpoint of 2500 ± 25 psia. The pressurizer safety valve reclosed within a short period of time. The small amount of reactor coolant that was relieved through the pressurizer safety valve (approximately 500 gallons) was contained in the pressurizer quench tank and there was no uncontrolled release of coolant to the containment building.

2.2 Detailed Sequence of Events

The following listing provides a detailed sequence of events. The event sequence was reconstructed by review of documentation and by interviews with operations and other personnel.

August 22, 1992

1:52:27 a.m.	The plant experienced a partial loss of turbine load. The load decrease from about [illegible] MWe to 380 Mwe was caused by a power supply failure in the electrohydraulic control system for the main turbine. The loss of load caused the reactor coolant system temperature and pressure to increase rapidly.
1:52:33 a.m.	The pressurizer spray valve opened between 2175 and 2225 psia and all four of the pressurizer high pressure pre-trip alarms were annunciated at about 2350 psia.
1:52:54 a.m.	One of the four pressurizer high pressure trip signals (Channel "C") initiated at about 2400 psia as the reactor coolant system temperature and pressure continued to increase.
1:52:55 a.m.	One of the two pressurizer safety valves (RC-142) opened at about 2397 psia and the reactor coolant system pressure immediately started to decrease. The safety valve lift was confirmed by the operators by a high tail pipe temperature; a high flow alarm; and an increase in the indicated pressurizer quench tank parameters (pressure, level, and temperature).
1:53:04 a.m.	A reactor trip was automatically initiated by Channels A and B from thermal margin/low pressure signals at about 2000 psia, decreasing. The reactor trip initiated a turbine trip. The operators initiated the actions required by EOP-00, "Standard Post Trip Actions" to respond to the plant transient and diagnose the event.
1:53:05 a.m.	The pressure in the steam generators reached a maximum of about 1004 psia and at least one of two main steam safety valves (set at 1000 psia) lifted momentarily.
1:53:06 a.m.	The reactor coolant system hot leg temperature reached a maximum of about 603° F and the

average temperature reached a maximum of about 573° F.

1:53:26 a.m. The pressurizer quench tank pressure peaked at about 12.5 psig. This indicated that Valve RC-142 had reseated. The valve was apparently open for about 31 seconds. The pressurizer quench tank level increased to about 82 percent indicating that about 500 gallons of reactor coolant had been discharged from the pressurizer.

1:53:56 a.m. The reactor coolant system pressure decreased to a minimum of about 1728 psia.

2:02 a.m. The operators transitioned to EOP-01, "Reactor Trip Recovery" from EOP-00. All safety functions were noted to be satisfactory including:

- Pressurizer Level - 42.9 percent
- Pressurizer Pressure - 1975 psia
- Reactor Vessel Level - 100 percent
- Core Differential Temperature - 2.0° F
- Reactor Coolant Temperature - 532° F
- Steam Generator Level - 76.3 percent
- Steam Generator Pressure - 908 psia
- Primary Containment Pressure - 1.09 psig
- Primary Containment Temperature - 100.3° F

2:04 a.m. The normal charging pump suction was aligned to the refueling water storage tank to provide borated water makeup to the reactor coolant system.

2:08 a.m. The group "N" control rods were fully inserted into the reactor core.

2:10 a.m. The block valves (HCV-150 and 151) for the pressurizer power operated relief valves were closed.

2:22 a.m.	The emergency diesel generators were shutdown following the normal automatic start as a result of the reactor trip.
2:35 a.m.	Normal power backfeed from the 345 Kv switchyard was established.
2:40 a.m.	The pressurizer quench tank level (82 percent to 76 percent) and pressure (12.5 to 5.6 psig) were returned to normal.

3 OPERATOR RESPONSE

The inspectors reviewed documentation and conducted interviews with personnel involved with the response to, and recovery from, the August 22, 1992, event.

The partial loss of secondary load was considered to be an analyzed plant transient. With the exception of the early lifting of Pressurizer Safety Valve RC-142, the plant and operator responses were as expected. The reactor trip-turbine trip occurred about 37 seconds after the secondary load reduction and was initiated from a thermal margin/low pressure signal as a direct result of the opening of RC-142 and the resultant reactor coolant system pressure decrease. The pressurizer safety valve opened 9 seconds prior to the reactor trip and reseated about 22 seconds after the reactor trip.

When the reactor tripped, the operators initiated and followed EOP-00, "Standard Plant Trip Actions." The plant conditions were as predicted for a standard reactor trip with the opening and closing of a pressurizer safety valve. Since RC-142 closed with no other abnormal conditions, and the plant conditions were generally stabilized, the operations staff transitioned to EOP-01, "Reactor Trip Recovery."

The operators subsequently closed the block valves (HCV-150 and 151) for the pressurizer power operated relief valves because of a decrease in reactor coolant system pressure. The block valves were closed to preclude potential leakage through a power operated relief valve. In this situation, however, the pressure decrease was the result of feedwater being supplied to the steam generators and the recovery of reactor coolant system letdown flow, and not the result of a leaking relief valve. Document reviews and personnel interviews revealed that the decrease in the reactor coolant system pressure was limited. Further, other instrumentation and alarms were available to aid the operators in identifying a leaking power operated relief valve in order to substantiate the need to isolate the relief valves. The indications available to the operators included valve position indications, and relief valve tail pipe temperatures and flows. The block valves were maintained closed until 2:55 a.m.

The inspectors noted that the closing of the block valves would create a situation that would prevent the implementation of the once-through-cooling option of EOP-20, "Functional Recovery Procedure, HR-4, "Reactor Coolant

System and Core Heat Removal for Once-Through-Cooling." The licensee stated that they had also identified this issue during the routine post-trip review. They further stated that appropriate actions would be taken to emphasize to plant personnel the dual role of the block valves in providing both an isolation function and a potential core cooling flow path function.

Overall, the inspectors found the licensee's actions in response to the event to be good. The inspectors determined that all systems and components operated as expected throughout the event with the exception of having RC-142 open prematurely.

4 PLANT SYSTEMS AND COMPONENTS

This section of the report discusses the equipment failure in the electrohydraulic control system that initiated the event and the performance of Pressurizer Safety Valve RC-142. The licensee's actions in response to those problems are also discussed.

4.1 Electrohydraulic Control System

The electrohydraulic control system is a nonsafety-related system that supplies control signals to the main turbine stop (trip) valves and control valves. The four control valves are modulated by hydraulic pressure against spring tension to control the steam flow to the turbine and thereby control the turbine's load carrying capability. The hydraulic system pressure is governed by electronic control circuitry, that senses various parameters including steam line pressure, turbine first-stage pressure, turbine speed, and turbine intermediate-stage pressure, to monitor the status of the main turbine.

The electrohydraulic control system includes devices which convert the power supplied from the plant's electrical distribution system through distribution panel AI-50 to four separate voltage levels to operate various components. Most of the electrohydraulic control system components were connected to the +30, +24, or -22 Vdc internal power distribution busses. These three power distribution busses were supplied from duplicated power conversion devices; one receiving power from the licensee's distribution system and the other receiving power from a permanent magnet generator attached to the shaft of the main generator. There were, however, four pressure transmitters which had two separate power conversion devices that were only connected to the licensee's distribution system. One of the conversion devices provided power for Pressure Transmitters PT-939, "Throttle Pressure Sensor," and PT-944, "Intermediate Pressure Transducer," and the other conversion device provided power for Pressure Transmitters PT-943, "Throttle Compensation Pressure Sensor," and PT-945, "First-Stage Turbine Pressure Sensor." A simplified sketch of this power distribution configuration is provided as Attachment B.

On July 3, 1992, the licensee experienced a loss of the plant's distribution system power to the electrohydraulic control system which led to a loss of reactor coolant event. Following that event, the licensee implemented modifications to provide a more reliable source of power from the plant's distribution system to the electrohydraulic control system. The licensee also implemented modifications that were designed to initiate a turbine and subsequent reactor trip on a similar loss of power condition. The modifications included the initiation of a turbine trip signal when Control Valve CV-1 closed and when a power imbalance of 40 percent between the turbine and electrical demands was detected.

On August 22, 1992, the failure of the internal power conversion device for Pressure Transmitters PT-943 and -945 caused their output signal to drop to 0 Vdc. The loss of power to only two of the four pressure transmitters caused the control valves to ramp in the closed direction but not close completely enough to initiate a turbine trip signal. The resultant reactor systems response was similar to, but not as severe as, the July 3 event.

As a result of the August 22 event and to further improve the reliability of the electrohydraulic control system and thereby limit challenges to the reactor protective system, the licensee initiated Engineering Change Notice No. 92-308 to evaluate additional modifications to the electrohydraulic control system. The licensee determined that the system would be enhanced by connecting the four pressure transmitters discussed above to the distribution systems internal to the electrohydraulic control system rather than the separate power conversion devices (Acopian Convertors) that had been used. The power for Pressure Transmitters PT-939 and PT-944 was connected to the +30 Vdc power distribution system and the power for PT-943 and PT-945 was connected to the -22 Vdc distribution system (see Attachment B). These modifications provided an automatic backup power supply to the transmitters to improve their reliability.

The inspectors reviewed the modifications and noted that the potential for a single pressure transmitter's failure resulting in a plant transient would still exist because the transmitters were not redundant. The licensee stated that the advantages of installing redundant pressure transmitters was being considered as part of the normal plant improvement evaluation being conducted in response to the event.

The inspectors determined that the modifications to the electrical power system for the electrohydraulic control system were implemented in a timely manner and were improvements to the reliability of the facility.

4.2 Pressurizer Safety Valves

As discussed above, on August 22, 1992, a loss of load problem resulted in a pressure increase in the reactor coolant system. The pressure increased to approximately 2397 psia when Pressurizer Safety Valve RC-142 opened prematurely. The valve had been set under laboratory conditions to open at 2500 ± 25 psia. The safety valve relieved the system pressure and then reclosed. The relieved reactor coolant was directed to the pressurizer quench tank.

Pressurizer Safety Valves RC-141 and -142 are 3-inch, nozzle-type safety valves, Size 3K6, Style HB-86-BP, Type E, manufactured by the Crosby Valve and Gage Company. Pressurizer Safety Valve RC-141 had a setpoint of 2545 ± 25 psia. The pressurizer safety valves were designed to limit reactor coolant system pressure to 110 percent of design pressure (2750 psia) following a loss-of-load condition on the main turbine, without a simultaneous reactor trip, while operating at 100 percent power. To accomplish this objective, Pressurizer Safety Valves RC-141 and RC-142 are provided to relieve pressure at a rate to ensure that design pressure of the reactor coolant system would not be exceeded and then shut, with a blowdown of approximately 20 percent (i.e., shut when pressure was reduced to approximately 2000 psia).

A more detailed discussion of the operation and history of the pressurizer safety valves is presented in NRC Inspection Report 50-285/92-18, which documents the findings of the Augmented Inspection Team that evaluated the July 3, 1992, event.

4.2.1 In-Situ Testing

In response to the August 22 event where Pressurizer Safety Valve RC-142 opened at approximately 4 percent below its set pressure of 2500 psia, the licensee performed in-situ testing of the Pressurizer Safety Valves, RC-141 and RC-142. Trevitest equipment, which is owned and operated by the licensee's contractor, Furmanite, was used for the in-situ testing. During the testing, the plant was at hot shutdown, the pressurizer pressure was maintained between 1900 and 2000 psia, and water was in the loop seals for the initial valve lift during each series of tests. The Trevitest equipment was attached to the spindle of the valve and exerted an upward hydraulic force on the spindle which compressed the spring and opened the valve. The applied force was measured by a load cell and its corresponding readout was converted to valve set pressure by the use of calculations. The valves were tested in accordance with Setpoint/Procedure Number SE-ST-RC-3001, Revision 0, "Pressurizer Safety Valves Verification of The Lift Point Using Furmanite's Trevitest Equipment", dated August 24, 1992.

The in-situ testing was performed by Fermanite personnel and witnessed by the licensee, Crosby (valve vendor), and NRC personnel. The purpose of the testing was to establish the normal operating conditions of the valves and to determine setpoint changes caused by temperature changes within the valve. Resistance temperature detectors were attached to the valves at the loop seal, the body inlet neck, the lower bonnet, and the upper bonnet and their corresponding temperatures were recorded for each valve opening during the testing. The valve temperatures determined from the in-situ tests were used during the testing at Wyle Laboratory.

The test results for RC-141 showed a set pressure change of approximately -2 percent and +3 percent from the valve set pressure of 2545 psig. The set pressure change for RC-142 varied from approximately -6 percent to +2 percent of the valve set pressure of 2500 psia. The licensee concluded with regard to the lower setpoint for RC-142, that setting a steam safety valve setpoint while on a water loop seal can be misleading. During the July 3 and August 22 events, Valve RC-142 demonstrated a 4 percent or less downward shift of the setpoint when it lifted under the actual plant operating conditions (i.e., with a loop seal established). After the in-situ testing was completed, the plant was cooled down to cold shutdown and Pressurizer Safety Valves RC-141 and RC-142 were removed and shipped to Wyle Laboratory for additional testing.

The licensee was found to be proactive in performing in-situ testing of the pressurizer safety valves so that normal plant operating temperature conditions of the valves could be determined and applied to the testing performed at the Wyle Laboratory.

4.2.2 Laboratory Testing

An inspector observed the receipt, inspection, and testing of Pressurizer Safety Valves RC-141 and -142 at the Wyle Laboratory in Huntsville, Alabama. Receipt, inspection, and testing were also observed by the licensee, Crosby and Wyle personnel.

Valve RC-141 was not disassembled during the receipt inspection. The receipt inspection included valve cap removal and adjusting ring and nozzle ring checks to ensure there was no binding. No abnormalities were noted during the inspection of the valve. Following the receipt inspection, RC-141 was installed on the test stand and preparations were made for testing.

The receipt inspection of RC-142 included a partial disassembly of the valve which was performed to determine if any of the valve internal components had been damaged during the event. The valve cap was removed, the nozzle ring and adjusting ring were checked for binding, and a jacking device was used to compress the spring and lift the valve internals off of the nozzle seat so that the valve could be disassembled without disturbing the valve set pressure. During disassembly, the only damage noted were a few nicks and scratches on the nozzle seat. The valve seats were lapped with a minimal material removal (<0.001 inches), and the valve was reassembled for testing.

Set pressure tests were performed to first determine the set pressures of the valves under simulated normal operating conditions of the plant and then the set pressures of the valves were adjusted and the valves tested under steam conditions in accordance with Wyle Test Procedure Number 41011, Revision A, as modified by OPPD engineering and dated August 29, 1992. To simulate the normal plant operating conditions with the valve mounted on a loop seal, an attempt was made to thermally stabilize the valve using low pressure steam at 235°F at the inlet of the valve in order to obtain temperatures of 176° ± 10°F at the body inlet neck, 148° ± 10 F at the lower bonnet, and 126° ± 10 F at the upper bonnet of the valve. Thermocouples were installed on the valve at these locations and the valve was insulated using the Fort Calhoun valve insulation materials. The stabilization temperatures were based on temperature measurements obtained from in-situ test data. In addition, thermocouples were installed on the valve at the upper body, the upper and lower portions of the outlet flange neck, the nozzle and the adjusting ring for information purposes (see Attachment C for thermocouple locations).

The attempt to set pressure test the valves under simulated normal operating conditions was performed to establish a reliable set point first at cold conditions and then under stabilized hot conditions with the Fort Calhoun insulation installed. A set pressure difference was expected between the two cases due to temperature differences throughout the valve resulting in different rates of thermal growth of valve components.

The test results under the low temperature simulated loop seal conditions demonstrated that the opening pressures of the valves were not repeatable due to limitations of the test facility. The desired temperatures could not be maintained at simulated loop seal conditions. The valves were set pressure tested a few times while they were being heated up for the testing under the stabilized hot conditions. The test data indicated that initially the set pressure increased as the nozzle temperature increased and then decreased as the body temperatures heated up. When the valves reached their hot stabilized (steam) condition, with approximately 650°F steam at the valve inlet and a body inlet neck temperature in excess of 450°F, tests showed the set pressures to be low. The set pressures of RC-141 and RC-142 were increased by turning the adjusting bolt one flat clockwise and the valves were again set pressure tested. Each valve was tested three times at the hot stabilized conditions and the set pressures were repeatable within ±1 percent of the specified set pressure. The final three lift pressures for RC-141 were 2535, 2549, and 2531 psia, which were within the tolerance for the required set pressure of 2545 psia ± 1 percent. The lift pressures for RC-142 were 2499, 2495, and 2495 psia, again within the tolerance for the required set pressure of 2500 psia ± 1 percent.

Following the steam tests and a cooldown period for the valves, a nitrogen opening pressure test was performed on each valve. The opening pressure for RC-141 was 1 percent high at 2572 psia and the opening pressure for RC-142 was 4 percent low at 2396 psia. No adjustments were made to the valve setpoints. The valves were again installed on the steam test facility and seat leak tested at 90 percent of their set pressure prior to shipping back to Fort Calhoun.

The licensee's Pressurizer Safety Valve testing program at the Wyle Laboratory was considered to be noteworthy and comprehensive since the program included set pressure testing under simulated normal operation conditions of the plant at low temperatures and testing on steam as required by the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code. Both the high and low temperature tests had been conducted to bound the opening pressure range of the valves.

4.2.3 Set Pressure Study

The inspectors reviewed the licensee's Engineering Analysis No. EA-92-067, "Pressurizer Safety Valves RC-141 and RC-142 Setpoint Evaluation With Loop-Seal", Revision 0, dated September 1, 1992. This engineering analysis provided a summary of the August 22 event, a discussion of the in-situ testing, and a detailed discussion of the set pressure testing at the Wyle Laboratory. Based on the results of the tests, the licensee concluded that variations in set pressure occur due to different temperature profiles within the valve body, bonnet and internals. In addition, the licensee noted that using the Fort Calhoun valve insulation while testing the valves at Wyle yielded a set pressure approximately 3 percent lower than the set pressures previously established at Wyle, prior to resetting the valves to their required set pressures. The engineering analysis concluded that RC-142 might open approximately 1 percent below the set pressure established at Wyle using the Fort Calhoun insulation since it had opened at 4 percent below the set pressure previously established. Additionally, the engineering analysis determined that the valve could open up to 5 percent higher than the set pressure if a second lift occurred before the valve temperatures had time to stabilize.

The licensee's engineering analysis was viewed as being comprehensive and well engineered.

5. CORRECTIVE ACTIONS

The licensee implemented a number of corrective actions to minimize challenges to the pressurizer safety valves. The changes to the electrical power supply circuitry for the electrohydraulic control system, that were discussed in paragraph 4.1, were designed to decrease the probability of a loss of load event being caused by power supply problems. The licensee also implemented changes to reactor protective system setpoints in order to limit challenges to the pressurizer safety valves.

The licensee decided to reduce the high pressurizer pressure reactor trip actuation setpoint in order to reduce the probability of opening the pressurizer safety valves. Since the same actuation signal was also provided for opening the pressurizer's power operated relief valves, these valves would open, as designed, prior to the safety valves and further reduce the probability of opening a safety valve.

The inspectors reviewed the licensee's evaluation dated September 1, 1992, for reducing the high pressurizer pressure reactor trip and power operated relief valve setpoint from 2400 psia to 2350 psia. The licensee determined that the setpoint change would not involve an unreviewed safety question nor require a change to the plant's technical specifications. Therefore, the licensee concluded that the change could be performed in accordance with the provisions of 10 CFR Part 50.59. The inspectors agreed that, since the technical specifications only required that the setpoint be less than 2400 psia, the change could be made without changing the technical specifications. In addition, the inspectors noted that the high pressure reactor trip was designed to limit the peak pressure that the reactor coolant system would experience during a pressure transient. By lowering the setpoint, the trip would occur sooner in a transient and the resultant pressure peak would be lower.

The inspectors also reviewed the licensee's Engineering Analysis No. EA-FC-92-066, which was completed on August 29, 1992. The analysis calculated the peak reactor coolant system pressure that would result from a worst case loss of load event (which results in the worst pressure transient) assuming various safety valve drift allowances. The analysis indicated that the peak reactor coolant system pressure would not exceed the Safety Limit of 2750 psia with up to a 6 percent drift allowance and the earlier high pressure trip and relief valve setpoint of 2400 psia. The licensee also calculated the peak reactor coolant system pressure with a 6 percent drift allowance at the reduced trip setpoint of 2350 psia and determined that the margin to the Safety Limit of 2750 psia would be increased when the modification was implemented.

In addition, the inspectors reviewed the licensee's proposed corrections and clarifications to the Updated Safety Analysis Report. The licensee had developed a draft change that reflected the reduced trip setpoint and added a discussion on pressurizer safety valve setpoint drift. The licensee also proposed to add a tabulation of the calculated peak reactor coolant system pressure for various values of safety valve setpoint drift. The inspectors determined that these changes and additions would enhance the information provided in the Updated Safety Analysis Report.

The inspectors found that the licensee's analyses provided a strong basis for the revision to the high pressurizer pressure reactor trip and pressurizer power operated relief valve setpoints. The inspectors also found the prompt implementation of the modification to readjust those setpoints to be commendable.

6. LICENSEE PRESENTATION

Immediately prior to the public exit meeting that was conducted on September 3, 1992, the licensee provided a presentation to describe the event that occurred on August 22, 1992, and the actions that were taken in response to that event. A copy of the licensee's presentation handout is attached (Attachment D).

ATTACHMENT A

1 PERSONS CONTACTED

The inspectors contacted the following persons during this inspection, in addition to other personnel.

Omaha Public Power District

C. Carlson, Shift Supervisor
*J. Chase, Assistant Plant Manager
*S. Gambhir, Division Manager, Production Engineering
J. Gasper, Manager, Training
R. Jaworski, Manager, Station Engineering
*W. Jones, Senior Vice President
J. Knight, Lead Special Services Engineer
*R. Lewis, Principle Engineer, Design Engineering
D. Lippy, Licensing Engineer
W. Orr, Manager, Quality Assurance and Quality Control
T. Patterson, Manager, Fort Calhoun Station
*F. Peterson, President
*R. Phelps, Manager, Design Engineering
A. Richards, Assistant Manager, Fort Calhoun Station
*R. Short, Manager, Nuclear Licensing and Industry Affairs
*C. Simmons, Station Licensing Engineer
J. Skiles, Manager, Mechanical Engineering
J. Tills, Assistant Manager, Fort Calhoun Station

Crosby Valve and Gage Company

P. Dalpe, Field Services Representative
S. Morse, Service Representative

Fermanite Corporation

M. Ballard, Field Services Representative

Wyle Laboratories

P. Turrentine, Engineering Supervisor, Steam Test Services

NRC Region IV

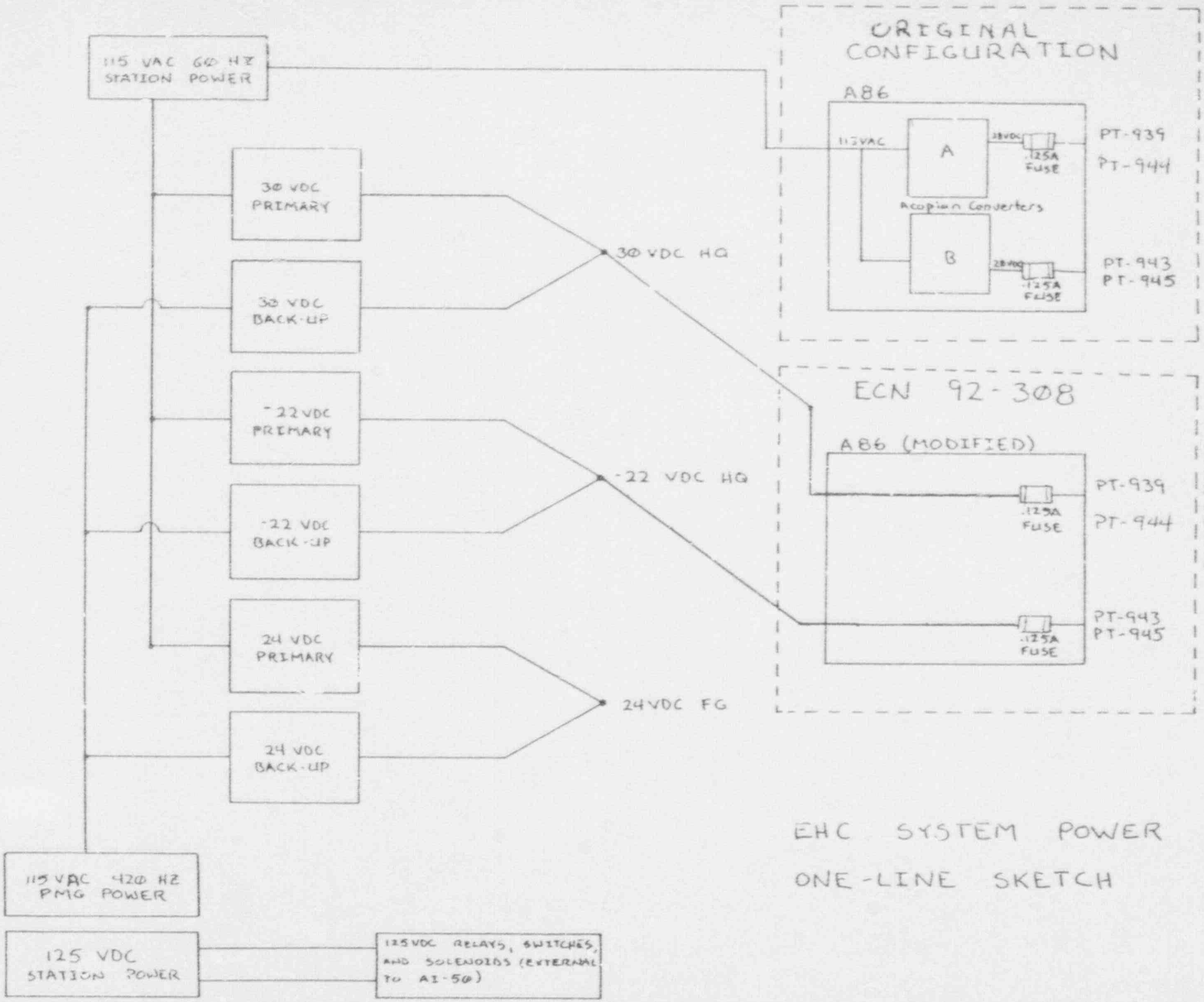
*S. Collins, Director, Division of Reactor Safety
*J. Gilliland, Public Affairs Officer
*P. Harrell, Chief, Reactor Projects Section C
*P. Wagner, Team Leader

* Denotes personnel that attended the public exit meeting conducted on September 3, 1992.

2 EXIT MEETING

A public exit meeting was conducted on September 3, 1992. During this meeting, the lead inspector reviewed the scope and findings of the inspection. The licensee did not identify as proprietary any of the materials provided to, or reviewed by, the inspectors during this inspection.

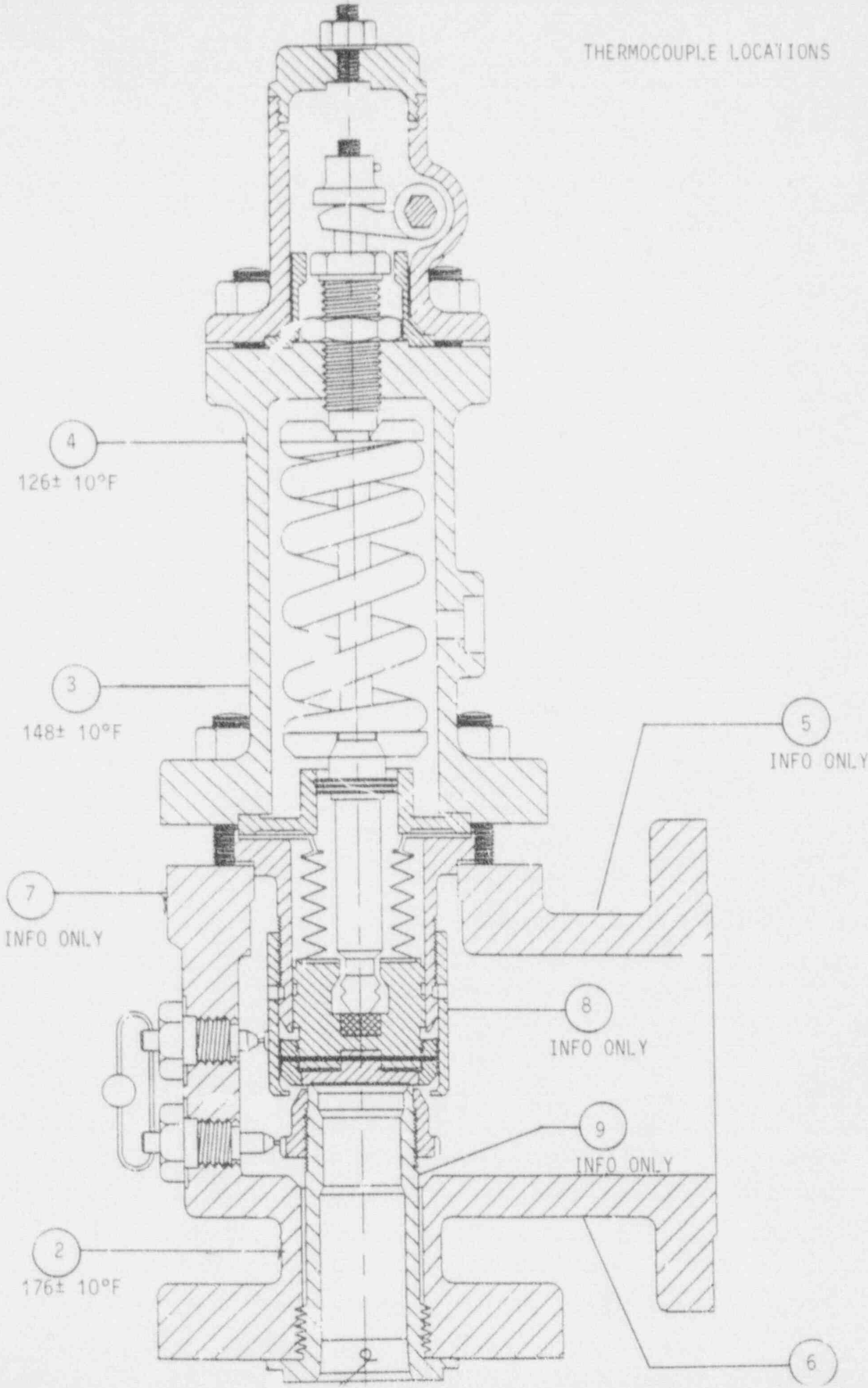
ATTACHMENT B



EHC SYSTEM POWER
ONE-LINE SKETCH

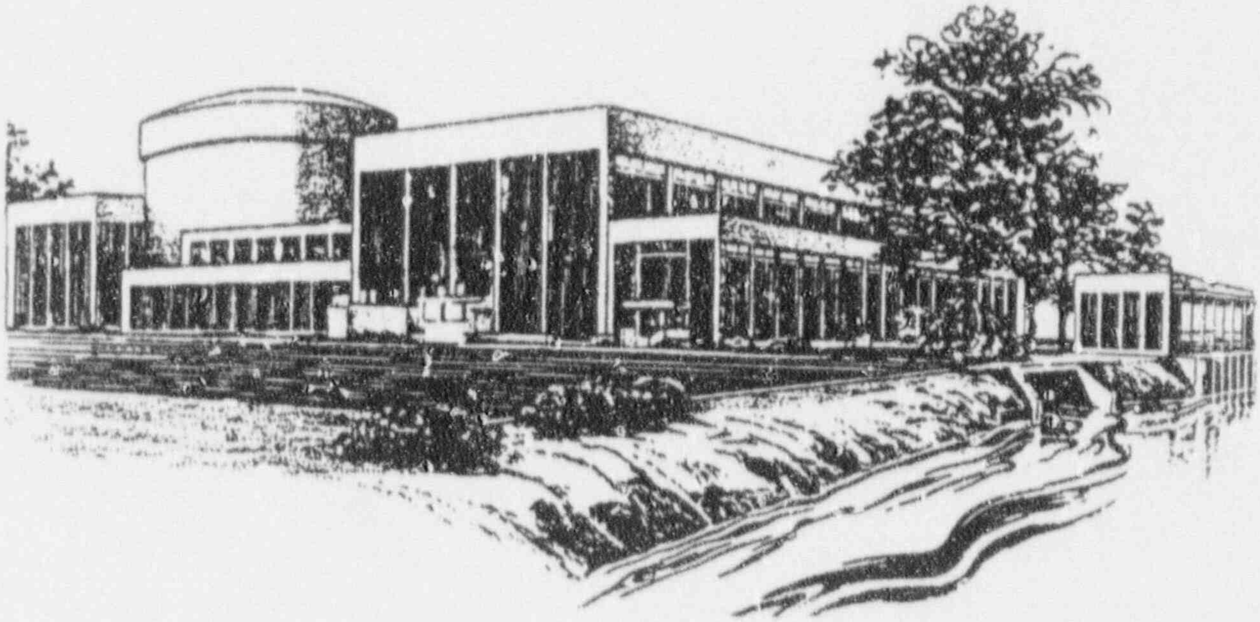
ATTACHMENT C

THERMOCOUPLE LOCATIONS



ATTACHMENT D

OMAHA PUBLIC POWER DISTRICT



FORT CALHOUN STATION

OPPD/NRC MEETING

September 3, 1992

AGENDA

OPENING REMARKS/EVENT REVIEW	W. C. JONES
ROOT CAUSE ANALYSIS/CORRECTIVE ACTIONS	S. K. GAMBHIR
- TRANSMITTER POWER SUPPLY	
- PRESSURIZER SAFETY VALVES	
SAFETY EVALUATION	R. L. PHELPS
CLOSING REMARKS	W. C. JONES

EVENT REVIEW

TIME

1:52:27 AM	Transmitter Power Supply Fails Causing Turbine Control Valves to Ramp from 40% to 22% Open.
1:52:55 AM	RC-142 Lifts at 2397.6 psia, followed by TM/LP Reactor Trip.
1:54 AM	RC-142 Reseated. Pressure Begins to Recover from Minimum Value of 1715 psia.
2:02 AM	EOP-00, "Standard Post Trip Actions", Exited With All Safety Functions Met.
2:04 AM	Charging Pumps Suction Realigned to Refueling Water Storage Tank.
2:35 AM	Backfeeding Buses 1A1/1A2 From 345KV.
3:19 AM	Unrodded Shutdown Margin Confirmed. Secured Boration.

ROOT CAUSE ANALYSIS/CORRECTIVE ACTIONS

TRANSMITTER POWER SUPPLY

FUNCTIONAL REQUIREMENTS

- Power Supply for PT-945 (First Stage Pressure Feedback) and PT-943 (Throttle Pressure Compensator)

CAUSE OF FAILURE

- Exact Cause Being Evaluated

CORRECTIVE ACTIONS

- Redesigned Control Circuit
- Eliminated Power Supply
- Initiated Reliability Review for Turbine Control System

PRESSURIZER SAFETY VALVES

FUNCTIONAL REQUIREMENTS

- Protect Reactor Coolant System (RCS) Against Overpressurization
- Maintain RCS Integrity

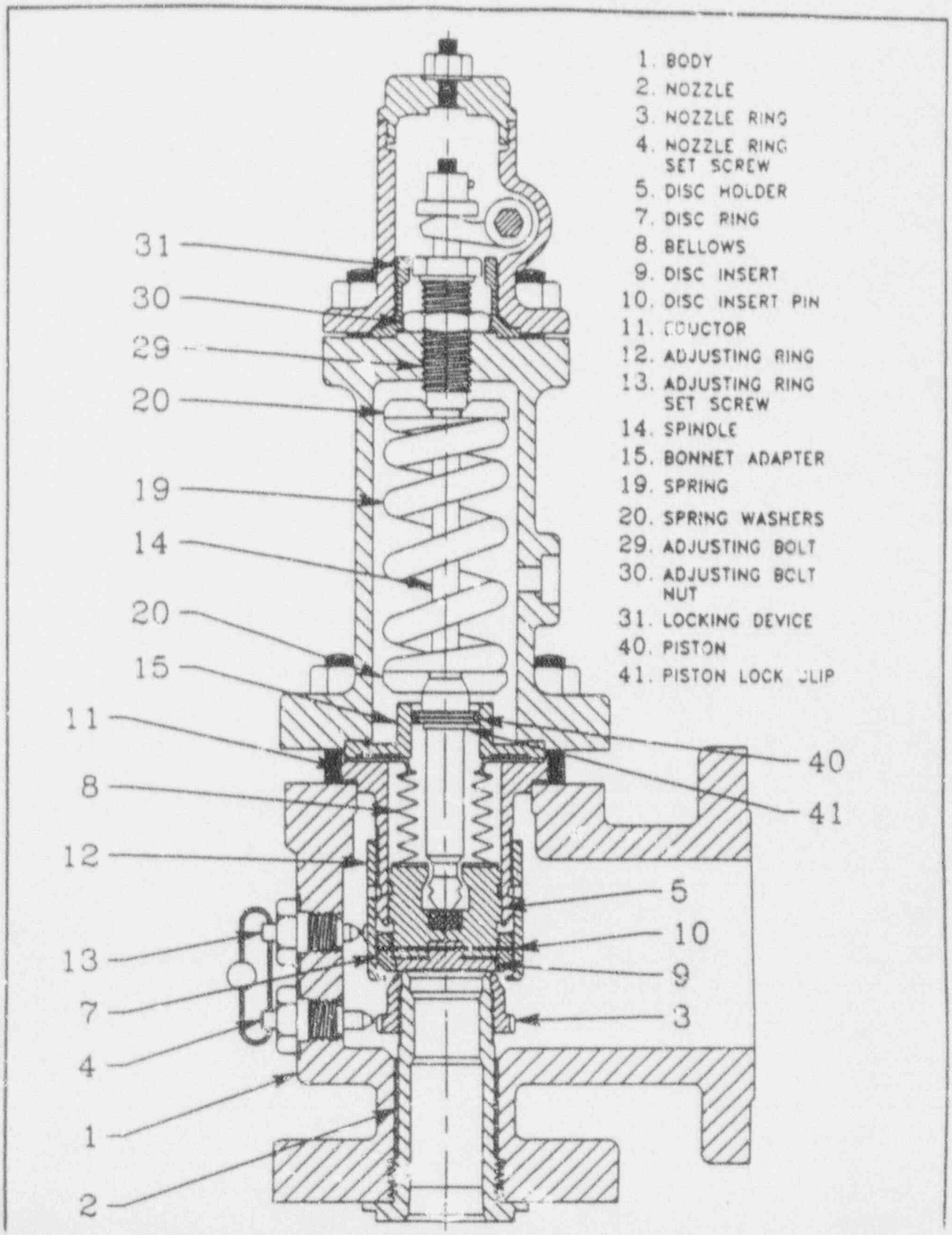
CAUSE OF PREMATURE LIFT (RC-142)

- Shift in Setpoint Because of Temperature Effects

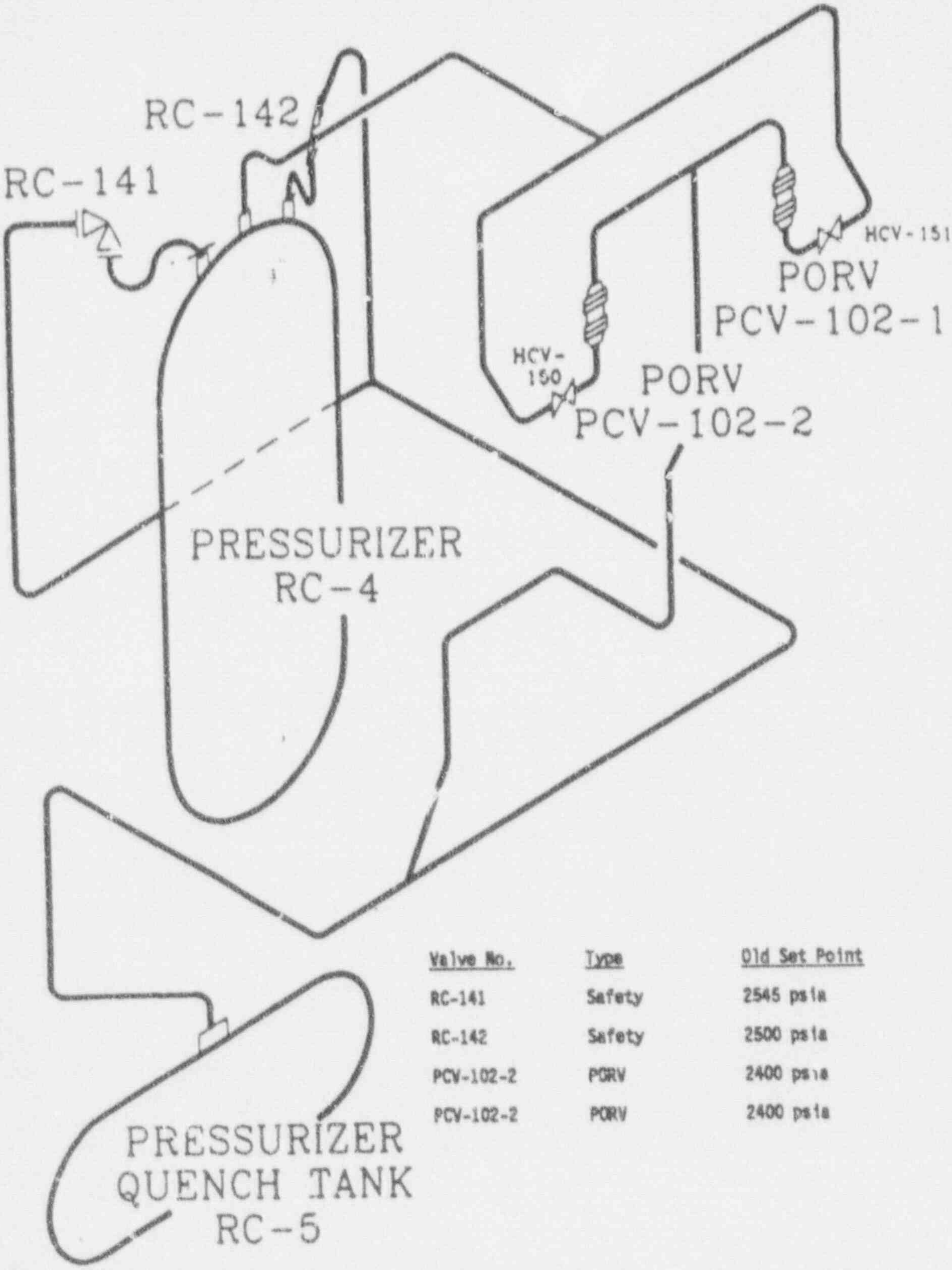
CORRECTIVE ACTIONS (RC-141/142)

- On-Site Testing
- Wyle Laboratory Testing
- Modelled Temperature Effects
- Valves Calibrated per Technical Specifications
- Lowered PORV and High Pressure Trip Setpoints
- Completed Safety Evaluation for Expected Performance
- Procedure Changes/Operator Training

PRESSURIZER SAFETY VALVE (RC-141/142)



PRESSURIZER SAFETY & RELIEF VALVE PIPING DIAGRAM



Valve No.	Type	Old Set Point	New Set Point
RC-141	Safety	2545 psia	2545 psia
RC-142	Safety	2500 psia	2500 psia
PCV-102-2	PORV	2400 psia	2350 psia
PCV-102-2	PORV	2400 psia	2350 psia

SAFETY EVALUATION

PURPOSE

- DOCUMENT THE BASES FOR SAFE AND RELIABLE OPERATION OF THE PRESSURIZER SAFETY VALVES (PSVs) IN THEIR EXISTING CONFIGURATION.

METHOD

- ANALYZED LIMITING EVENT TO DEFINE ALLOWABLE MARGIN IN PSV SETPOINT UNCERTAINTY.
- EVALUATED TEST DATA TO VERIFY PSV PERFORMANCE IS WITHIN ANALYSIS MARGIN.
- SPECIFIED PSV SETPOINT CALIBRATION TEST PARAMETERS.
- LOWERED HIGH PRESSURE REACTOR TRIP AND PORV SETPOINTS.

RESULTS

- SAFE AND RELIABLE OPERATION IS ASSURED.
- POTENTIAL FOR CHALLENGES TO THE PSVs HAS BEEN REDUCED.
- NO NEW CHALLENGES TO THE PORVs.
- CALIBRATED THE PSVs TO MINIMIZE THE POTENTIAL FOR AN EARLY LIFT.