ENCLOSURE 4

U.S. NUCLEAR REGULATORY COMMISSION
REGION I INSPECTION REPORT

Report No.	50-272/89-27
License No.	DPR-70
Licensee:	Public Service Electric and Gas Co. P. O. Box 236 Hancocks Bridge, NJ.
Facility Name:	Salem Unit No. 1
Inspection At:	Hancocks Bridge, NJ.
Inspection Conducted:	November 29, 1989 thru December 1, 1989

Inspectors:

N. F. Dudley, Project Engineer, DRP — 1/3/90 Date

J. M. Trapp, Sr. Reactor Eng., DRS, EB — 1/3/90 Date

J. Yerokun, Reactor Eng., DRS, EB — 1/3/90 Date

Approved by:

P. D. Swetland, Chief
Reactor Projects Section 2A — 1/3/90 Date

Inspection Summary: Reactive Unannounced Inspection on November 29, 1989 through December 1, 1989.

Areas Inspected: Review of events surrounding the November 9, 1989 entry into Technical Specification 3.0.3 during the performance of the Turbine Volumetric Flow Test.

Results: The inspector found that some actions taken by the licensee during the Turbine Volumetric Flow Test did not conform with NRC Regulations. This inspection resulted in two violations. Three additional items were not resolved prior to the exit meeting held on December 1, 1989 and will be tracked as Unresolved Items.

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PDR ADOCK 05000272
Q PDC

Details

1.0 Persons Contacted

1.1 PSE&G

* L. Curran, Operating Engineer
* D. Dodson, Principal Engineer, Licensing
* W. Grau, Licensing Engineer
* B. Gorman, Manager External Affairs
M. Gwirtz, Senior Nuclear Shift Supervisor(SNSS)
* R. Heaton, System Engineer
* E. Krufka, Engineer
* S. LaBruna, Vice President Nuclear Operations
D. Martrano, Engineer
* M. Metcalf, Project Manager
* L. Miller, General Manager Salem Operations
H. Onorato, Licensing Engineer
* K. Pike, Reactor and Plant Perf. Engineer
* V. Polizzi, Operations Manager
* B. Preston, Manager, Licensing and Regulation
* R. Reichel, Engineer
* W. Schulk, Manager Station QA
J. Serwan, SNSS

1.2 U.S. NRC Personnel

* N. Dudley, Project Engineer
* K. Gibson, Senior Resident Inspector
* S. Pindale, Resident Inspector
* P. Swetland, Chief, RPS-2A
* J. Trapp, Senior Reactor Engineer
* J. Yerokun, Reactor Engineer

* denotes present at exit meeting held on December 1, 1989

2.0 Introduction

The purpose of this inspection was to review the events surrounding the entrance into Technical Specification Limiting Condition for Operation (LCO) 3.0.3 during the conduct of the Turbine Volumetric Flow Test. Specifically the scope of the inspection included the following:

- ° Establish a sequence of events.
- ° Review the adequacy of the test procedure/conduct.
- ° Review the root cause analysis for P10 jumper re-energizing trips.
- ° Conduct interviews with key personnel.
- ° Review actions taken to shutdown plant following entry into Technical Specification LCO 3.0.3.
- ° Review corrective actions.
- ° Assess performance of plant staff associated with the test.

The inspectors findings with regard to these issues are contained in this inspection report.

3.0 Summary of Events

The sequence of events surrounding the performance of the Turbine Volumetric Flow Test, is provided in Attachment A, in chronological order. A summary of the events is provided below.

On November 9, 1989 a Turbine Volumetric Flow Test, REM T-1, Rev. 0, was being conducted to collect data needed as part of the Rerating Feasibility Program. The objective of the test was to determine the minimum steam pressure and the corresponding average reactor coolant loop temperature (Tavg) for full power operation with the main turbine control valves in the full open position. The test procedure required Tavg to be reduced in 2°F increments by adding boron to the Reactor Coolant System. At each 2°F step, a calormetric calculation was planned with the Power Range Nuclear Instrumentation (NI's) being adjusted as necessary. Calibration of the NI's was required because a decrease in reactor inlet temperature (Tcold), corresponding to the decrease in Tavg, will cause the downcomer water to shield more neutrons; thus causing the NI indicated power to be less than actual reactor thermal power.

At 569°F the calormetric calculation was performed, NI's were adjusted, and the second boration to decrease Tavg an additional 2°F was completed. At 567°F a second calormetric calculation was performed and three NI channels N41-N43 were satisfactorily adjusted. While attempting to adjust N44 the fine gain adjustment reached its lower limit stop before the indicated power could not be raised to equal the actual power. The difference between the indicated and actual power was .8% in a non conservative direction. A difference of .8% satisfied the ± 1% acceptance criteria provided in the calormetric calculation procedure, however a second criteria that the average of the four channels be greater than or equal to the calormetric calculation could not be met. Therefore the operators chose conservatively to declare N44 inoperable per Technical Specification. At this point the turbine governor valves were full open and data collection for the test was complete.

Nuclear Instrumentation Channel N44 was declared inoperable at 5:40 a.m. per Technical Specification 3.3.1, Reactor Trip System Instrumentation. The Limiting Condition for Operation (LCO) action statement for an inoperable NI requires the inoperable channel be placed in trip within one hour. Shift operating personnel were confident that a course gain adjustment could be performed within one hour by I&C technicians supporting this test. If the course gain could be adjusted, calibration of N44 could be performed, and the LCO could then be exited. Operations requested that I&C perform the course gain adjustment, but did not convey adequately the urgency required due to the one hour action statement. I&C encountered a number of delays in making the coarse gain adjustment, and the decision was made by the operators at 6:30 a.m. to expedite the initial step of the course gain adjustment procedure which would place the channel in a trip condition. One hour and one minute after declaring N44 inoperable the channel bistables were tripped by I&C technicians. At this point a control room operator recognized that the I&C procedure did not include installation of a jumper to energize the P10 relay. The installation of the P10 jumper was included in the operations procedure for tripping an NI channel.

The operators made a decision to consider the NI not tripped until the P10 jumper was installed. Therefore the action statement for Technical Specification 3.3.1.1, which requires the inoperable channel to be tripped within one hour, was not met and Technical Specification 3.0.3 was entered. Technical Specification 3.0.3 requires in part that within one hour action shall be initiated to place the unit in a MODE in which the specification does not apply, and be in hot standby within the next six hours.

The operators believed that the successful installation of the P10 jumper would place N44 in a tripped condition per Technical Specification 3.3.1.1, and Technical Specification 3.0.3 could then be exited. At 7:36 a.m. the P10 jumper was installed by I&C. However, the installation of the P10 jumper, caused an unanticipated repowering of the previously deenergized bistables for the high flux rate trip, high flux trip (high setpoint), and the high flux trip (low setpoint). This was not understood and placed the channel in a condition not allowed by Technical Specifications. At 7:40 a.m. the one hour requirement of Technical Specification 3.0.3, to initiate action to place the unit in a MODE in which Technical Specification 3.1.1 did not apply had been exceeded. At 7:50 a.m. the Senior Nuclear Shift Supervisor (SNSS) ordered I&C to remove the P10 jumper. At the same time the SNSS, with the concurrence of the Operating Engineer, initiated the steps of the Turbine Volumetric Flow Test which restored the plant to normal conditions. These actions included diluting the Reactor Coolant

System (RCS) to raise Tavg back to it's program value. By raising Tavg, a calormetric calculation could be performed, allowing recalibration of N44. Recalibration of N44 would allow exiting Technical Specification LCOs 3.0.3 and 3.3.1.1. The SNSS thought that using the approved turbine test procedure restoration steps to restore Tavg and recalibrate the NIs was prudent and would return the N44 channel to service sooner than a power reduction which would extend the calorimetric stabilization period.

Tavg was raised to 569°F, and recalibration of the NI's was completed at 9:40 a.m. Technical Specification 3.0.3 was exited three hours after entering the specification. Technical Specification 3.3.1.1 was also exited and the plant was returned to it's normal 100% power condition.

4.0 Test Conduct

The inspectors reviewed the test procedure, "Turbine Volumetric Flow Test," REM T-1, Rev. 0. This procedure was approved by the Station's Operations Review Committee (SORC) and the General Manager.

The test objective was to determine the minimum steam pressure and the equivalent Tavg for full power operation with the turbine governor valves in the full open or near full open position. This information would be used as part of the Rerating Feasibility Program.

The 10CFR 50.59 Safety Evaluation performed for this test was reviewed by the inspectors. The evaluation acknowledged that the test was not described in the FSAR. The evaluation also imposed certain limits on the test in order to remain within analyzed conditions. The test was to be terminated when either all the turbine control valves are full open or Tavg had been reduced by 14°F. Test duration was limited to 16 hours or less. The impact of test performance on the active 16 reactor trips and five ESF actuation signals was analyzed in the Safety Evaluation. It was determined that the test would not reduce the margin of safety provided by these trips. The inspectors concluded the Safety Evaluation was adequate.

The "Precautions and Limitations" section of the test contained necessary information for a safe test performance except it did not list explicitly all the limitations contained in the 10CFR 50.59 Safety Evaluation. For example, the imposed 14°F temperature reduction limit was not listed. The inspectors acknowledged that this limit was mentioned elsewhere in the procedure.

The pre-test briefings did not include personnel from the I&C department. While personnel in the I&C department were on site specifically for the test, they were not included in the briefings. This was inadequate because the involvement of this department in the test was anticipated and mentioned in the procedure. Step 6.3 of the procedure indicated that briefings should be conducted with operators and test support personnel.

The test procedure referenced I&C procedures IC-14.4-021,022,023 or 024 to be used for instrument adjustment, if required. These referenced procedures were not the most appropriate procedures available to the I&C department for the applicable instrument adjustment. Because I&C personnel were not properly involved in pre-test activities, this oversight was not discovered or corrected prior to initiating the test. The resultant discussions to clarify the appropriate procedures delayed tripping the N44 channel.

Other sections of the procedure were found to be adequate. The restoration section of the test procedure properly restored the plant to its pre-test condition. The Turbine Volumetric Flow Test and plant restoration were conducted in accordance with the test procedure.

5.0 P10 Jumper Installation

The P10 Permissive allows manual blocking of the Source Range Detector Voltage, Intermediate Range Detector High Flux Trip, and the Power Range High Flux Trip-Low Setpoint during power escalation when power is greater than 10%. The logic for this permissive requires 2/4 Power Range Detectors to indicate a power level greater than 10% power to allow manual bypass of these trips by the operator. During power decreases, the trips will automatically reinstate when 3/4 of the Power Range Channels indicate less than 10% power. TS 3.3 requires this reinstatement function to be operable in operating Mode 1.

The Power Range Detector drawer powers bistables which send signals to the Solid State Protection System (SSPS) indicating the status of the power range channel. Bistables are provided for the reactor trips and for the permissives, one such bistable in the power range drawer is the P10 bistable. This bistable changes state at 10% power indicating to the SSPS when 2/4 channels are greater than 10% power. The reactor trips mentioned above may then be manually blocked. The output of the P10 bistable is energized or closed when reactor power is below 10% and open or deenergized when power is above 10%. When a power range channel is removed from service, Technical Specifications requires all the reactor trip bistables be placed in their deenergized (tripped) position. This is performed by removing the control power from the bistables. This sends a trip signal to the SSPS and reduces the reactor trip logic from a 2/4 to a 1/3. Removal of the control power works satisfactorily for the reactor trip bistables, but would not be adequate for the P10 permissive.

P10 bistables must energize below 10% power to reinstate the Intermediate Range, and Low Power Range Trip, and Source Range Voltage. When control power is removed, the P10 bistable deenergizes and cannot re-energize. Therefore the logic for reinstating the trips goes from a 3/4 less than 10%, to a 3/3 less than 10%. If a single failure is assumed, it can be postulated that the low level trips bypassed by P10, would not automatically reinstate. Consequently, a jumper must be installed to energize the P10 signal from the tripped channel to the SSPS, thus changing the logic to reinstate the trips to a 2/3. Detailed information regarding the P10 permissive when tripping a power range channel was provided in NRC Information Notice 86-105, and Westinghouse Technical Bulletin 86-06.

During the performance of the Turbine Volumetric Flow Test, the decision was made to remove Nuclear Instrumentation Channel 44 (N44) from service, due to the inability to calibrate this channel. The I&C technician removed N44 from service using I&C procedure 1IC-16.4.024, "Power Range Channel 1N44 Detector Current Adjustment." This procedure was used in lieu of the normal operations procedure for removing NI's from service IV-10.3.1, "Removing, Returning to Service and Loss of Protective System Channel." Upon completion of tripping the channel by I&C personnel, a control room operator recognized that the P10 jumper required by operations procedure IV-10.3.1, was not required or installed by the I&C technicians. Following identification by the operators that a P10 jumper was required, I&C was requested to acquire the required materials and install the P10 jumper in accordance with operations procedure IV-10.3.1.

The P10 jumper consisted of supplying external 115 VAC power to the output side of the P10 bistable and thus to the SSPS. When the P10 jumper was installed per procedure IV-10.3.1, the N44 trips which had previously been deenergized using the I&C procedure became re-energized. N44 was not in a tripped condition per Technical Specifications. The intent of the P10 jumper was not to re-energize these trips and the reason the trips became re-energized was not understood by the I&C technician or the operators on shift. Twelve minutes following re-energizing the reactor trips, the SNSS ordered the jumper removed and the trips previously deenergized by removing control power, were once again deenergized.

After the event, the licensee's initial corrective action for the failure of the P10 jumper to function properly was to revise the procedure IV-10.3.1 to lift electrical leads between the P10 bistable and the SSPS, and install the jumper directly in the SSPS. This action eliminated the possibility of the jumper affecting the other N44 bistables, but did not determine whether the unexpected NI channel performance resulted from some unknown system defect which could affect system operability.

The licensee's subsequent root cause analysis for the failure of the P10 jumper to function properly determined the following. The top and bottom detector cables had been removed from the N44 instrumentation drawer as part of the I&C procedure for removing a channel from service (1IC-16.4.024). This step was not included in the Operations procedure for removing a channel from service, which was ultimately used for installation of the P10 jumper because the P10 jumper was not referenced in the I&C procedure. By removing the N44 detector cables the channel indicated power fell OFF-SCALE low. Indicated power less than 10% caused the P10 bistable to close. Closing of the P10 bistable allowed an inductive circuit to feed back power supplied by the P10 jumper, through the P10 bistable to re-energize control power to the entire NI channel. Re-energizing the control power allowed the other reactor trip bistables associated with this channel to become re-energized even though the control power fuses were removed. The P10 jumper was not installed in accordance with guidance provided in the Westinghouse bulletin. This issue remains unresolved pending further NRC review of the licensee program for review and implementation of industry operational experience and vendor recommendations (UNR 272/89-27-01).

6.0 Assessment/Findings

The licensee entered Technical Specification 3.0.3 at 6:40 a.m. on November 9, 1989. After one hour, Technical Specification 3.0.3 required actions to be initiated to place the plant in a mode where Technical Specification 3.3.1.1 did not apply. 10CFR50.72 required that a notification be made to the NRC within one hour when initiation of any nuclear plant shutdown is required by the plant's Technical Specifications. This notification was not made by the licensee until after the NRC inspection team arrived onsite, when the licensee recognized a shut down should have been initiated. The licensee committed at the exit meeting to review the Emergency Classification Guidelines to assure that notifications will be made in the future when shutdowns are required by Technical Specifications.

Technical Specification 3.0.3 was entered at 6:40 a.m. and exited three hours later at 9:40 a.m. Technical Specification 3.0.3 required within one hour actions be initiated to place the unit in a MODE in which the specification does not apply. These actions to be performed within one hour, as described in the bases for Standard Technical Specifications, include time for the operator to prepare for and coordinate the reduction in electrical generation to ensure the stability and availability of the electrical grid. At no time during the three hour period were such preparations made to make a load reduction.

In a similar event on November 17, 1989, licensee management decided to enter Technical Specification 3.0.3 while processing an Emergency Operating Procedure revision to compensate for an identified Emergency Core Cooling System Design Deficiency. The licensee decided not to initiate actions for a reactor shutdown during the hour and forty-five minute period that Technical Specification 3.0.3 would be in effect. A report was made to NRC within one hour of the recognition of the design deficiency by the station. Details of the event are discussed in NRC Special Inspection Report No. 50-272/89-25; 50-311/89-23. These two events are considered an apparent violation of Technical Specification 3.0.3. (Violation 50-272/89-27-02).

Station operators are provided guidelines in the form of Operations Directives (ODs) which document station management's position and interpretation of selected Technical Specifications. OD-12, Revision 10 (2/10/86) provides such a position on Technical Specification 3.0.3, which states that "to show intent of compliance with this requirement, load should be reduced immediately at a rate determined by the senior shift supervisor, however, if it is likely that compliance with the Action Statement can be achieved within one hour, load does not have to be reduced. The licensee has revised OD-12 to require load to be reduced one hour after entering Technical Specification 3.0.3.

The inspectors observed that the present Technical Specification Interpretations provided in OD-12 do not require SORC review. The Technical Specification Interpretations are presently approved by the Operations Manager. The licensee committed in the NRC exit meeting to have all Technical Specification Interpretations SORC approved (Unresolved Item 272/89-27-03).

The licensed operators showed poor judgement in allowing the LCO action statement requirement of tripping the channel within one hour to be exceeded prior to taking action. Licensed operators are trained and capable of removing inoperable channels from service. Communications on the urgency of removing the channel from service was not adequately stressed to the I&C personnel by the operators. Using separate sections of I&C and Operations procedures for removing the NI channel from service was unacceptable. By removing the detector cables using the I&C procedure, which is not part of the operations procedure, the re-energization of the N44 trips was permitted to occur. Following the root cause determination, it was further revealed that when reactor power was decreased below 10%, the re-enerization of the trip bistables would have occurred even if the detector cables were installed. Therefore both I&C and Operations procedures for placing an NI channel in trip per Technical Specifications were inadequate.

The root cause analysis for the reenergization of the trip bistables upon installation of the P10 jumper was not determined until after the NRC inspection team arrived. This was twenty days after the event. The lack of timely root cause analysis on the part of the licensee is a violation of NRC regulations (Violation 50-272/89-27-04).

Operator communications and actions taken during this event were reviewed. The normal on-shift operating crew had been supplemented by the unit Operating Engineer and two I&C personnel dedicated to support test performance.

Communication among all party's involved in this test were viewed as being weak. I&C technicians were not made cognizant of the one hour Technical Specification time limit until forty minutes into the LCO. I&C personnel were also not made aware of the required support requirements, by the Test Director, prior to starting the test. Communications by shift operators to plant management when Technical Specification 3.0.3 was entered was also weak. The Operations Manager was not made cognizant of the details of entering Technical Specification 3.0.3 in a timely manner. These weaknesses had been appropriately identified during licensee review of the occurrence.

The Calormetric Calculation Procedure, "Rx ENG. MAN. PART 2," provides guiduance on when NI power must be adjusted following a calormetric calculation. This procedure requires NI adjustment if the thermal power is plus or minus 1% of the indicated NI power. A second requirement for adjustment is that the average of the four NI channels must be equal to or greater than the reactor thermal power. By not specifically requiring NI adjustment when an NI channel indicates greater than reactor thermal power, a maximum of 1% nonconservative difference may exist between actual and indicated power. Subsequent channel drift could exceed the 1% difference between the High Reactor Power Trip Setpoint and the allowable value required by the Technical Specifications. The significance of this event was minimal since no channel exceeded the 1% allowable value. During the inspection, NRC also questioned which channel must be considered inoperable when the criterion not met is the average of four channels. The licensee stated at the exit meeting that an evaluation would be made as to the adequacy of the acceptance criteria and its implementation with regard to Technical Specification operability (Unresolved Item 272/89-27-05).

7.0 Exit Meeting

A summary of the inspection findings was discussed with the licensee at the conclusion of the inspection on December 1, 1989. Additional discussions with the licensee were held on December 11, 1989, and during a telephone discussion on January 8, 1990.

Attachment A
Sequence Of Events
Turbine Volumetric Flow Test
November 9,1989

Time	Event
0130	Operating shift verifies portions of the Turbine Volumetric Flow Test prerequisites.
0300	First boration made to reduce Tavg 2°F from normal 100% Tavg of 571°F.
0330	Tavg @569°F first Calormetric Calculation being performed.
0350	First adjustment to NI's made.
0400	Second boration made to reduce Tavg an additional 2°F.
0415	Turbine Governor Valves full open.
0450	Tavg 567°F, second Calormetric Calculation being performed.
0520	Second adjustment of NI's being made. N41-N43 successfully adjusted N44 fine gain bottoms out.
0530	I&C "paged" to perform course gain adjustment on N44. I&C did not hear page at this time.
0540	I&C contacted to adjust course gain.
	SNSS declares N44 inoperable, starts one hour LCO clock to trip or restore N44, concurred by Test Director and NSS.
0620	I&C contacted to expedite N44 adjustment. Operations personnel make I&C supervisor aware of one hour Technical Specification requirement.
0630	I&C told by operations to place N44 in trip to comply with Technical Specification LCO Action Statement 3.3.1.1.
0640	One hour Technical Specification Action Statement 3.3.1.1 exceeded Tech. Spec. LCO 3.0.3 entered.

0641 N44 bistables tripped in accordance with I&C procedure.

NSS notes that P10 jumper not installed per Operations procedure IV 10.3.1.

I&C Technician leaves control room to acquire material required for P10 jumper.

Channel still considered not tripped LCO 3.0.3 continues.

0736 I&C Technician installs P10 jumper in accordance with Operations Procedure IV 10.3.1.

Bistables for HIGH FLUX RATE TRIP, HIGH FLUX TRIP (high/low setpoint) reenergize.

0750 SNSS verifies proper placement of P10 jumper.

SNSS orders removal of jumper.

Dilution commenced to return Tavg to program.

0840 Tavg 569°F, Calormetric Calculation being performed.

0940 NI's adjusted

N44 calibrated and returned to service.

Technical Specification LCO 3.0.3 and 3.3.1.1 cleared.

ATTACHMENT 1

PSE&G

NRC ENFORCEMENT CONFERENCE

RESIDUAL HEAT REMOVAL

COLD LEG INJECTION

ISOLATION VALVES

SJ49

DECEMBER 11, 1989

PUBLIC SERVICE ELECTRIC AND GAS

NRC ENFORCEMENT CONFERENCE

SJ49 VALVES

AGENDA

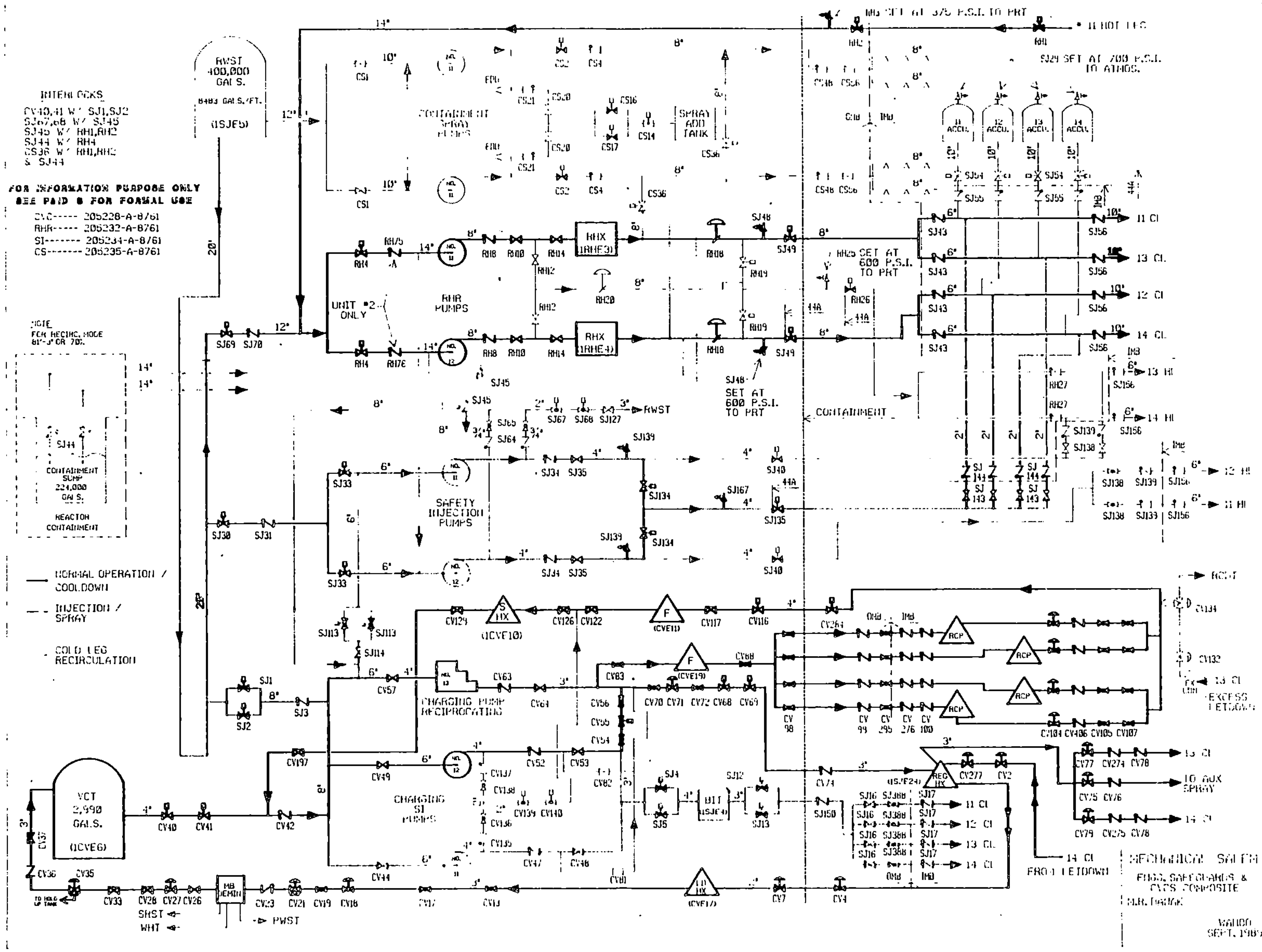
*	INTRODUCTION	T. M. CRIMMINS
	- PSE&G UNDERSTANDING OF NRC CONCERN	
*	SJ49 MODIFICATION HISTORY	J. BAILEY
	- SYSTEM DESIGN	
	- LICENSING BASIS	
	- ORIGINAL CONTROL CIRCUIT DESIGN	
	- MODIFIED CONTROL CIRCUIT DESIGN	
	- USQ DISCOVERY	
	- SAFETY SIGNIFICANCE	
*	SHORT TERM ACTIONS	L. K. MILLER
	- IMMEDIATE COMPENSATORY ACTIONS	
	- SORC REVIEW AND EFFECTIVENESS	
*	LONGER TERM ACTIONS	J. RONAFALVY
	- INVESTIGATION RESULTS	
	- CORRECTIVE ACTIONS	
	- INDEPENDENT REVIEW	
*	PSE&G ASSESSMENT OF POTENTIAL VIOLATION	B. A. PRESTON
	- APPLICATION OF GENERAL ENFORCEMENT POLICY	
*	SUMMARY	T. M. CRIMMINS
*	TECHNICAL SPECIFICATION 3.0.3 POLICY	S. LABRUNA

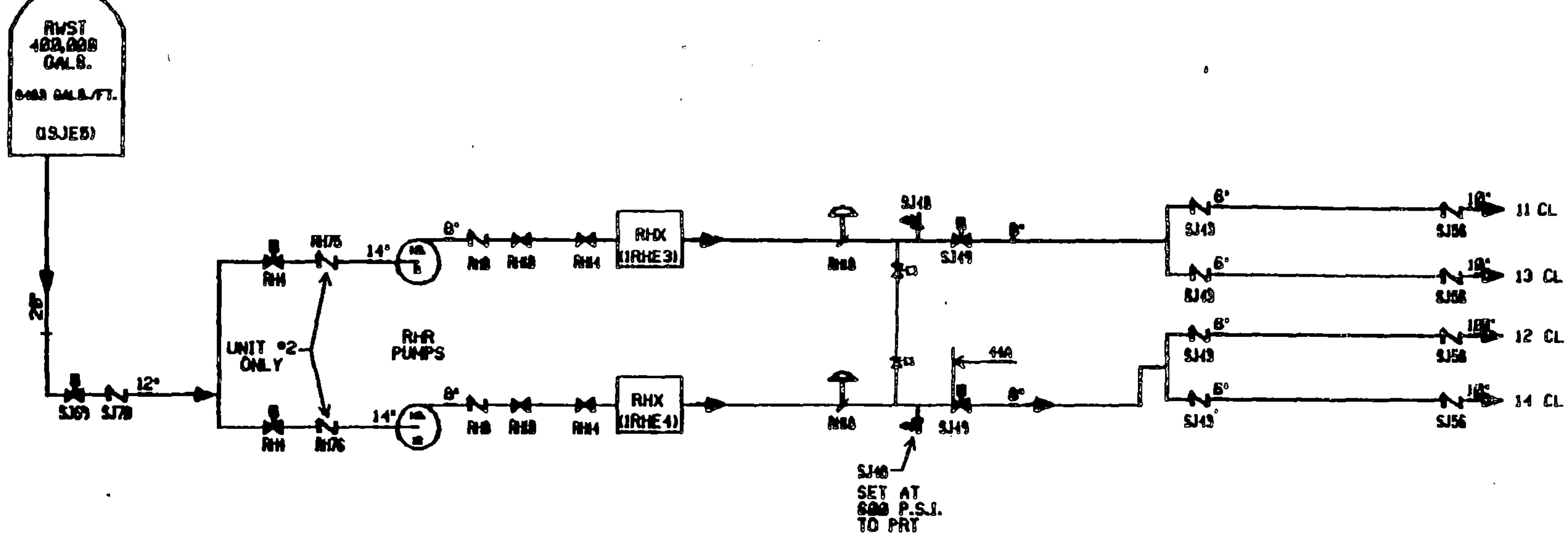
NRC FINDINGS

* APPARENT VIOLATIONS

  - FAILURE TO IDENTIFY THAT DCR 1EC-2295 FOR UNIT 1 AND 2EC-2295 FOR UNIT 2 CONTAINED A USQ BY INTRODUCING A POTENTIAL SINGLE FAILURE WHICH COULD HAVE JEOPARDIZED THE ABILITY OF THE ECCS SYSTEMS TO PERFORM THEIR SAFETY FUNCTION DURING A LOCA

  - FAILURE OF SORC TO IDENTIFY THAT PROPOSED CHANGES TO EOPs ON NOVEMBER 17 CONTAINED A USQ; SINGLE FAILURE VULNERABILITY NOT COMPLETELY RESOLVED.


INTERLOCKS
CV40,41 W/ SJ1,SJ2
SJ67,68 W/ SJ45
SJ45 W/ RH1,RH2
SJ44 W/ RH4
CS36 W/ RH1,RH2
& SJ44
FOR INFORMATION PURPOSE ONLY
SEE P&ID & FOR FORMAL USE
CVC----- 205228-A-8761
RHR----- 205232-A-8761
SI------- 205234-A-8761
CS------- 205235-A-8761
RWST
400,000
GALS.
(1SJE5)
11 HOT LEG
SJ29 SET AT 700 P.S.I. TO ATMOS.
CONTAINMENT SPRAY PUMPS
SPRAY ADD TANK
RHX (1RHE3)
RHX (1RHE4)
RHR PUMPS
UNIT #2 ONLY
RH25 SET AT 600 P.S.I. TO PRT
SET AT 600 P.S.I. TO PRT
CONTAINMENT
RWST
SAFETY INJECTION PUMPS
CONTAINMENT SUMP
224,000 GALS.
REACTOR CONTAINMENT
NORMAL OPERATION / COOLDOWN
INJECTION / SPRAY
COLD LEG RECIRCULATION
11 CL
13 CL
12 CL
14 CL
13 HL
14 HL
12 HL
11 HL
RCP
CHARGING PUMP RECIPROCATING
CHARGING SI PUMPS
VCT
2,990
GALS.
(1CVE6)
(1CVE10)
(1CVE11)
(1CVE19)
BIT
REG HX
EXCESS LETDOWN
13 CL
10 AUX SPRAY
14 CL
14 CL
FROM LETDOWN
SHST
WHT
PWST
MB DEMIN
MECHANICAL SKETCH
ECCS, SAFEGUARDS & CVCS COMPOSITE
WAHOO
SEPT. 1989



RHR - INJECTION MODE

LICENSING BASIS

* FSAR QUESTIONS

  - SEVERAL VALVES IDENTIFIED WHOSE MISPOSITIONING COULD AFFECT THE ABILITY TO MITIGATE AN ACCIDENT

* FSAR RESPONSE

  - VALVES INITIALLY COMMITTED TO HAVE POWER REMOVED

  - VALVES REQUIRED TO OPERATE IN THE SHORT TERM WOULD HAVE CONTROL POWER LOCKOUT SWITCHES

* BTP ICSB-18 / IEEE-279

  - DETAILED REQUIREMENTS FOR SINGLE FAILURE

  - SHORT CIRCUITS TO BE INCLUDED AS SINGLE FAILURE

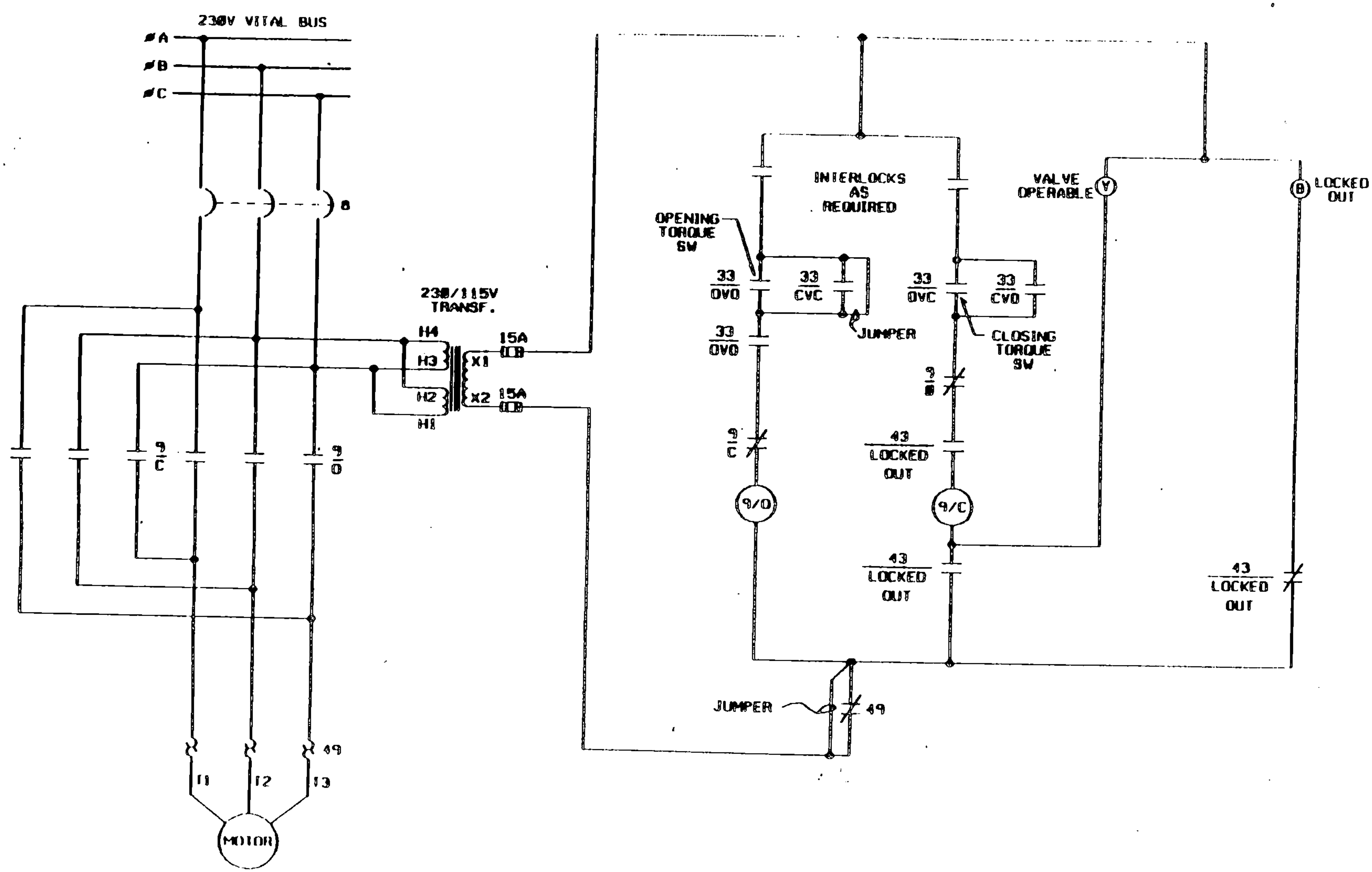
* CONCLUSION

  - CONTROL POWER LOCKOUT SWITCHES MET REQUIREMENTS

    o REQUIRED TWO OPERATOR ACTIONS TO OPERATE VALVE

    o SINGLE ELECTRICAL FAULT WOULD NOT REPOSITION VALVE

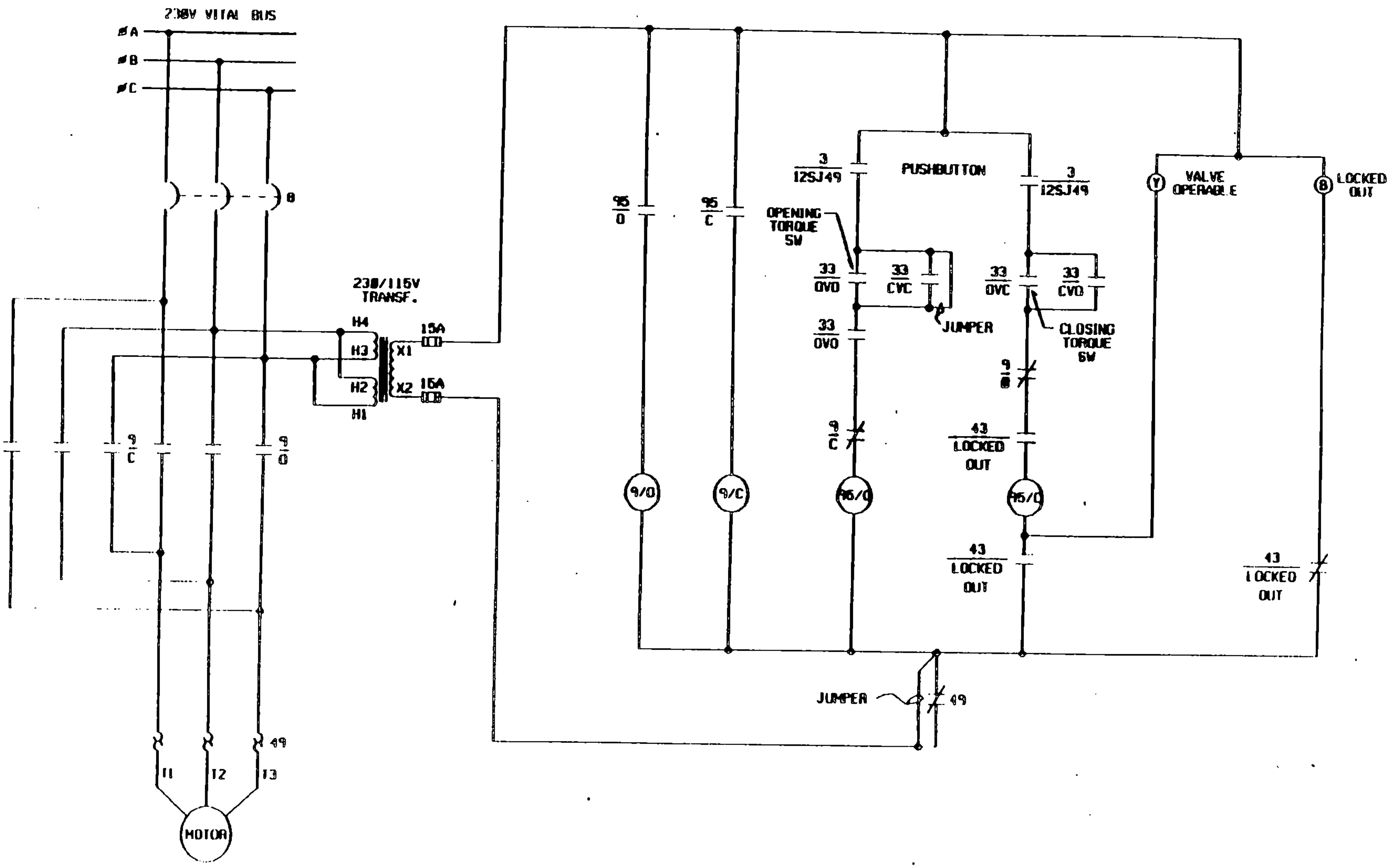
LICENSED DESIGN



TYPICAL MOV 230ØV POWER LOCKOUT

UPDATED FSAR FIGURE 7.3-3

CURRENT INSTALLATION



SJ49 MOV 230V POWER LOCKOUT

USQ DISCOVERY

* SEPTEMBER 29, 1989

  - POTENTIAL SINGLE FAILURE CONCERN IDENTIFIED

  - SENT TO PRA GROUP TO PRIORITIZE

  - CONTINUED RESEARCH INTO LICENSING BASIS AND SINGLE FAILURE CRITERIA

* NOVEMBER 13, 1989

  - DEF RETURNED TO EAG WITH NEAR TERM PRIORITY USING QUALITATIVE PRA REVIEW

* NOVEMBER 14, 1989

  - EAG SUPERVISOR CONCURS WITH PRIORITY

  - NOTIFICATION OF MANAGEMENT AND STATION AS PER PROCEDURE

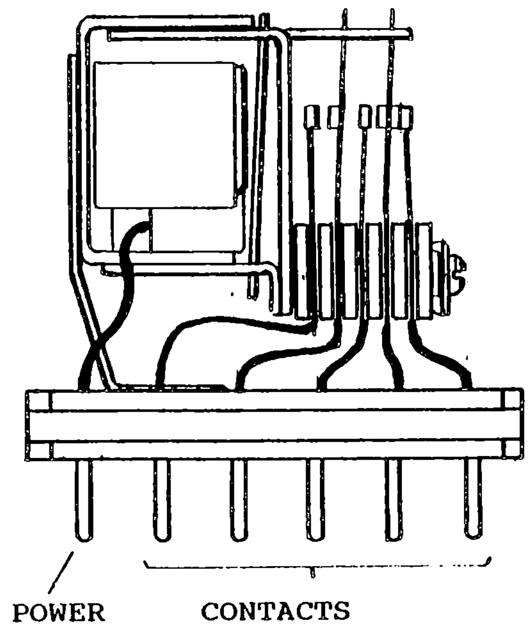
  - MANAGEMENT MEETING WITH STATION

  - QUESTIONS OVER VALIDITY OF SINGLE FAILURE

  - REQUEST FOR FURTHER RESEARCH WITH WESTINGHOUSE

HEAVY-DUTY PLUG-IN RELAY FAMILY

219 Frame 12-Pin



USQ DISCOVERY (CONT.)

* NOVEMBER 17, 1989

  - ALL RESEARCH COMPLETE
  - NO CHANGE IN ORIGINAL CONCLUSIONS
  - STATION NOTIFIED
  - INCIDENT REPORT PREPARED
  - TSAS ENTERED
  - NOTIFICATION MADE TO NRC
  - COMPENSATORY ACTION IMPLEMENTED
  - ENGINEERING EVALUATION REQUESTED BY SORC
  - INITIATED OSR INDEPENDENT INVESTIGATION OF INCIDENT

* NOVEMBER 20, 1989

  - NRC QUESTIONS ON COMPENSATORY ACTIONS

* NOVEMBER 21, 1989

  - EOP REVISIONS REVIEWED AND APPROVED BY SORC

* NOVEMBER 22, 1989

  - ENGINEERING EVALUATION PRESENTED TO SORC AND APPROVED

LOCA ANALYSIS (EXISTING)

LARGE LOCA

* LIMITING BREAK IS LOCA WITH CD = 0.4

* LIMITING SINGLE FAILURE

  - ONE RHR PUMP
  - FAILURE OF SI PUMPS ON AFFECTED TRAIN ALSO ASSUMED

* ANALYSIS ASSUMES FLOW TO ALL COLD LEGS (I.E. 3 INTACT LOOPS) CALCULATED FLOW FROM ONE SI TRAIN IS 3374 GPM AT 25 PSIA RCS PRESSURE

* CALCULATED PCT IS 2091 F (BASH ANALYSIS)

LARGE LOCA
SAFETY EVALUATION

* LIMITING BREAK IS LOCA WITH CD = 0.4

* LIMITING SINGLE FAILURE = CLOSURE OF ONE SJ49

* ALL PUMPS INCLUDING BOTH RHR PUMPS ARE RUNNING

* EVALUATION ASSUMES RHR FLOW TO ONE COLD LEG

* CALCULATED TOTAL FLOW TO ONE LOOP 2864 GPM AT 25 PSIA RCS PRESSURE (BY WESTINGHOUSE)

* CALCULATED PCT PENALTY DUE TO SI DEGRADATION IS 29 F. (BY WESTINGHOUSE)

* NO ADDITIONAL PCT PENALTY FROM ASYMETRIC FLOW DELIVERY (BY WESTINGHOUSE)

* CALCULATED NEW PCT: 2091 + 29 = 2120 F.

* NEW PCT REMAINS LOWER THAN 2200 F

* MINIMAL SAFETY SIGNIFICANCE

PROBABILITY ANALYSIS

* PROBABILITY OF OCCURRENCE

  - COMBINED PROBABILITY OF LARGE LOCA AND RANDOM SINGLE FAILURE OF CONTROL CIRCUIT (2.5×10^{-11})

* DETECTABILITY OF CONTACT FAILURE

  - INDEPENDENT VALVE POSITION INDICATIONS AND ALARMS

  - OVERHEAD ALARM ON EITHER VALVE NOT FULLY OPEN

IMMEDIATE ACTIONS

* INCIDENT REPORT GENERATED
* TECH SPEC ACTION STATEMENT 3.0.3 WAS ENTERED
* NOTIFICATION TO NRC FOLLOWED PROMPTLY
* CONVENED A SORC MEETING

COMPENSATORY ACTIONS

(TAKEN AT 11/17/89 SORC MEETING)

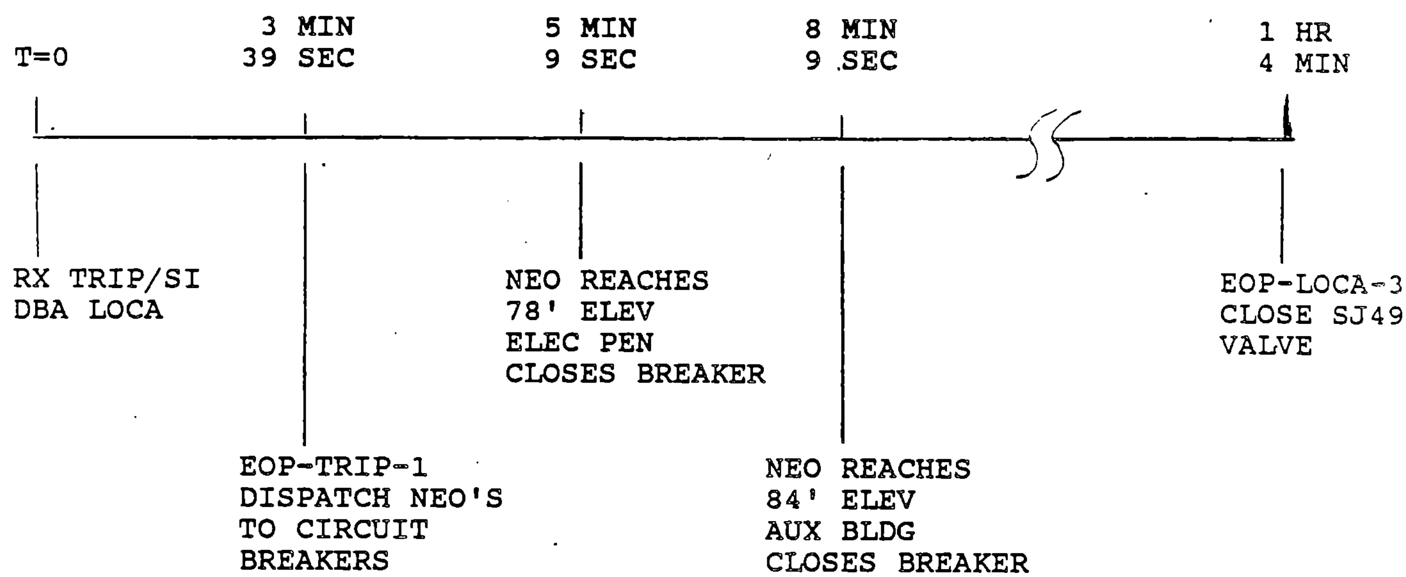
* TAGGED SJ49 MOTOR BREAKERS IN OPEN POSITION TO ELIMINATE SINGLE FAILURE CONCERN.

* REVISED OPERATOR LOGS TO REQUIRE VERIFICATION OF SJ49 OPEN POSITION, EACH SHIFT.

* CONDUCTED BRIEFINGS WITH SHIFT PERSONNEL.

* REVISED EOP'S TO ENSURE SJ49'S ARE POWERED-UP FOR SWITCHOVER TO RECIRCULATION.

TIMELINE FOR SJ49 ENERGIZATION



ROOT CAUSE AND
CORRECTIVE ACTIONS

ROOT CAUSE

* FAILURE TO IMPLEMENT REQUIREMENTS OF AP-32 AND PERFORM 50.59 EVALUATION.

CORRECTIVE ACTION

* COMPLETE REVIEW OF INCIDENT WITH ALL SORC MEMBERS
* COMPLETE RE-EVALUATION OF SORC REVIEW PROCESS FOR POTENTIAL ENHANCEMENTS
* INCIDENT WILL BE REVIEWED WITH ALL STATION PERSONNEL REQUIRED TO APPLY AP-32.
* COMPLETE REVIEW AND REVISION OF NAP-32
* TRAIN AND QUALIFY STATION QUALIFIED REVIEWERS TO NEW NAP-32

SORC REVIEW

NRC CONCERN

* FAILURE OF SORC TO IDENTIFY USQ IN PROPOSED CHANGES TO EOP'S.

SUMMARY OF ISSUE

* SORC UNDERSTOOD SINGLE FAILURE POTENTIAL WAS ELIMINATED WITH SJ49 BREAKERS TAGGED OPEN.
  - ADDITIONAL ENGINEERING REVIEW CONCLUDED THAT CIRCUIT COULD NOT BE PRE-CONDITIONED WITH A CREDIBLE SINGLE FAILURE.
* INITIAL EOP CHANGE - RESTORE BREAKER POWER ASAP.
  - REVIEWED IN DEPTH BY SORC
  - SORC RECOGNIZED MINIMAL SINGLE FAILURE POTENTIAL. QUALITATIVE PRA EVALUATION - MINIMAL RISK
  - EARLY RESTORATION OF BREAKER POWER JUDGED TO BE PRUDENT FROM A HUMAN FACTORS VIEW POINT.
    - o SWITCHOVER TO RECIRCULATION LESS COMPLICATED.
    - o ELIMINATES POTENTIAL FOR MISCOMMUNICATION TO NEO DURING CRITICAL EVOLUTION IN EOP'S.
    - o SIMILAR LOGIC APPLIED TO ACCOMPLISH OTHER EOP ACTIONS.
    - o UNNECESSARILY TIES UP NEO DURING EARLY PHASES OF ACCIDENT.
* SORC TOOK INCOMPLETE COMPENSATORY ACTION.
* FINAL EOP CHANGE - STANDBY NEO TO CLOSE BREAKER FOR RECIRCULATION SWITCHOVER.

STATION OPERATIONS REVIEW COMMITTEE (SORC) EFFECTIVENESS

* SORC REVIEW ACTIVITY

  - REVIEW OF LER'S, DCP'S, TECH SPEC CHANGES, PROCEDURES, VIOLATION RESPONSES, ETC.

  - > 100 OPEN ITEMS IDENTIFIED FOR RESOLUTION OVER LAST 4 YEARS.

  - 50 OF 638 (8%) ITEMS REVIEWED IN 1989 TO DATE REJECTED.

* OSR PERFORMS INDEPENDENT REVIEW OF ALL 50.59 SAFETY EVALUATIONS.

  - TO DATE, NO SIGNIFICANT SAFETY ISSUE IDENTIFIED AFTER SORC REVIEW.

* QA AUDIT PERFORMED EVERY 2 YEARS WHICH COVERS ALL SORC ACTIVITIES.

  - LATEST AUDIT (8/88) IDENTIFIED NO NEGATIVE FINDINGS

    o CONCLUDED "SORC" FUNCTION IS EFFECTIVE.

* THIRD PARTY REVIEW PERFORMED BY IMPELL (EARLY 1989)

  - SORC REVIEW PROCESS FOUND TO BE ADEQUATE.

* NRC INSPECTION (3/89) CONCLUDED THAT SORC REVIEW OF DCP'S WAS ADEQUATE.

* AMERICAN NUCLEAR INSURERS (ANI) AUDIT (10/89) OF SORC ACTIVITIES - DETERMINED TO BE EFFECTIVE.

* SRG MEMBERSHIP ON SORC PROVIDES FOR AN INDEPENDENT VIEWPOINT.

<u>SORC REVIEW PROCESS IS EFFECTIVE.</u>

INVESTIGATION OF EVENTS AND CAUSAL FACTORS
THAT LED UP TO THE EVENT, REVIEWED:

* APPLICABLE PROCEDURES USED IN 1987

* PLANT DESIGN AND DESIGN CRITERIA

* PEOPLE ISSUES:
  - ENVIRONMENT
  - TRAINING/QUALIFICATION

* EXISTING PSE&G AND INDUSTRY EXPERIENCE ON SIMILAR ISSUES. NO INFORMATION WAS FOUND ON SIMILAR CIRCUIT DESIGN ISSUES.

* OTHER CHANGES MADE ON POWER LOCKOUT CIRCUITS AND CHANGES MADE DURING THIS TIMEFRAME. NO OTHER CONCERNS IDENTIFIED.

CONCLUSIONS

THE ERROR IN DESIGN WAS THE RESULT OF AN INADEQUATE REVIEW OF DESIGN BASE DOCUMENTATION. THE REVIEW FAILED TO IDENTIFY THE PECULIARITY OF THE CIRCUIT DESIGN REQUIREMENT FOR MITIGATING SINGLE FAILURE CRITERION IN THE INJECTION MODE OF THIS SYSTEM. WE BELIEVE THIS TO BE AN ISOLATED EVENT.

MAJOR CONTRIBUTING FACTORS

* ENVIRONMENT

  - PLANT SHUTDOWN
  - REORGANIZATION

* THE UNIQUENESS OF THIS CIRCUIT'S CHARACTERISTIC FOR MITIGATING SINGLE FAILURE WAS NOT COMPLETELY UNDERSTOOD BY THE ENGINEER WHO WORKED ON THE DESIGN CHANGE

* PROCEDURE IN 1987 DID NOT REQUIRE DOCUMENTATION OF THE DETAILS OF AN FSAR REVIEW

* SUBSEQUENT REVIEWERS DID NOT HAVE EXPLICIT KNOWLEDGE OF THE UNIQUENESS OF THE CIRCUIT

CORRECTIVE ACTIONS

* THE FOLLOWING ACTIONS HAVE BEEN, OR WILL BE ACCOMPLISHED TO PREVENT RECURRENCE

  - EXECUTE MODIFICATIONS TO SJ49 CIRCUITS TO REESTABLISH ORIGINAL DESIGN REQUIREMENTS. TO BE ACCOMPLISHED DURING THE NEXT REFUELING OUTAGES.

  - PROCEDURES/DOCUMENTS/PROCESSES:

    o 1987 PROCEDURE AND CURRENT DESIGN CHANGE PROCEDURE COMPLY WITH APPENDIX B REQUIREMENTS

      - CURRENT PROCEDURE PROVIDES FOR BETTER ORGANIZATION AND INSTRUCTIONS FOR DEVELOPMENT OF THE DESIGN CHANGE AND FOR BETTER DOCUMENTATION OF THE CONCLUSIONS

    o PROCEDURE FOR DOING 50.59 EVALUATION WAS REVIEWED. NSAC 125 AND OUR INDEPENDENT AUDIT ENHANCEMENTS HAD BEEN INCORPORATED. DOCUMENTATION OF SECTIONS OF FSAR AND OTHER DOCUMENTS REVIEWED IS NOW A REQUIREMENT. NO FURTHER ACTION REQUIRED.

    o MODIFIED DEF RESOLUTION PROCESS TO PREVENT POTENTIAL TIME LAPSES.

    o FSAR SECTIONS DEALING WITH THIS ISSUE WILL BE REVIEWED WITH THE INTENT TO CLARIFY REQUIREMENTS.

CORRECTIVE ACTIONS (CONT.)

- PERSONNEL

  o TRAINING

    - ON 50.59 PROCEDURE AND PROCESS
    - DCP PROCEDURE AND PROCESS
    - ENGINEERING TRAINING PROGRAM

- CONTINUE OUR DEVELOPMENT OF ECCS, LOCA AND EOP ANALYSIS EXPERTISE IN ENGINEERING

- INITIATED AN INDEPENDENT INVESTIGATION BY OUR OFF-SITE SAFETY REVIEW GROUP ON THE DESIGN ERROR. FURTHER ACTIONS TO CORRECT OR STRENGTHEN OUR PROGRAM MAY RESULT

- NUCLEAR DEPARTMENT IMPROVEMENT PROGRAM INITIATED

- PARTICIPATING IN NUMARC SUBCOMMITTEE ON DISCREPANCY RESOLUTION PROCESS

- DISSEMINATE LESSONS LEARNED

PSE&G ASSESSMENT OF POTENTIAL VIOLATION
APPLICATION OF GENERAL ENFORCEMENT POLICY
(10CFR PART 2, APPENDIX C)

* APPLICATION OF MITIGATING FACTORS:

- IDENTIFICATION AND REPORTING
  - o DCP DEFICIENCY WAS SELF IDENTIFIED BY PSE&G
  - o PROMPTLY REPORTED THE VIOLATION TO NRC
- CORRECTIVE ACTION TO PREVENT RECURRENCE
  - o IMMEDIATE COMPENSATORY ACTIONS TAKEN IN PLANT
  - o IMMEDIATE INVESTIGATION UNDERTAKEN ON DCP
- PAST PERFORMANCE
  - o PERFORMANCE IN APPLICATION OF 50.59 IN DCP PROCESS HAS BEEN VERY GOOD. ISOLATED CONCERN
- PRIOR NOTICE OF SIMILAR EVENTS
  - o NO SPECIFIC NRC, INDUSTRY, OR OTHER INDICATION
- MULTIPLE OCCURRENCES
  - o ISOLATED DEFICIENCY
- SAFETY SIGNIFICANCE
  - o DETERMINISTIC EVALUATION - PCT < 2200 F
  - o PROBABILITY OF OCCURRENCE 2.5×10^{-11}

PSE&G ASSESSMENT OF POTENTIAL VIOLATION
APPLICATION OF GENERAL ENFORCEMENT POLICY
(10CFR PART 2, APPENDIX C)

* APPLICATION OF NRC DISCRETION

  - PSE&G AGGRESSIVE IN IDENTIFYING, REPORTING, AND CORRECTING VIOLATIONS

    - o NOT REASONABLY PREVENTABLE BASED ON PRIOR NRC, INDUSTRY, OR PSE&G EXPERIENCE OR NOTICE
    - o NOT WILLFUL
    - o DOES NOT REPRESENT A BREAKDOWN IN MANAGEMENT CONTROLS

BASED ON MITIGATING FACTORS AND APPLICATION OF NRC DISCRETION, PSE&G BELIEVES ESCALATED ENFORCEMENT SHOULD NOT BE APPLIED TO THE SJ49 ISSUE.

SUMMARY

* A THOROUGH EVALUATION OF THE RHR COLD LEG INJECTION ISOLATION VALVE DEFICIENCY HAS BEEN PERFORMED

* PRELIMINARY ROOT CAUSES HAVE BEEN DETERMINED

* CORRECTIVE ACTIONS HAVE BEEN TAKEN AND ARE CONTINUING

* DEFICIENCY HAS MINIMAL SAFETY IMPACT

* STUDY PERFORMED TO ENSURE THAT DESIGN CHANGES HAVE NOT VIOLATED SINGLE FAILURE CRITERIA FOR OTHER SIMILAR CONTROL POWER LOCK OUT CIRCUITS

* COMPENSATORY ACTIONS HAVE BEEN TAKEN AND PLANT IS CURRENTLY IN COMPLIANCE

* DCP DEFICIENCY WAS SELF-IDENTIFIED THROUGH A PROGRAMMATIC ASSESSMENT AND IS A POSITIVE INDICATION OF PSE&G's INTENTION TO IDENTIFY/CORRECT PROBLEMS

T/S 3.0.3 POLICY

CURRENT POLICY

* UPON ENTRY INTO T/S 3.0.3 START PREPARING FOR SHUTDOWN

* INITIATE POWER REDUCTION NO LATER THAN ONE HOUR AFTER ENTERING T/S 3.0.3.

* MAKE 10 CFR 50.72 ONE-HOUR REPORT WITHIN 60 MINUTES OF INITIATING A POWER REDUCTION.

PLANT MANAGEMENT HAS CLEARLY ARTICULATED POLICY TO ALL LICENSED OPERATORS