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APR 4 1980

MEMORANDUM FOR: Robert L. Tedesco, Chairman
B&W Reactor Transient Response Task Force

FROM: Carl Michelson, Director
Office for Analysis and Evaluation of Operational Data

SUBJECT: ADDITIONAL OPERATIONAL DATA FOR CONSIDERATION BY THE TASK FORCE

During our review of the recent Crystal River event, we came across Abnormal Report No. 74-3 (Enclosure 1) filed by the Sacramento Municipal Utility District on September 20, 1974. We believe this report contains information which should be of interest to your task force.

We note that in Section 4.2 of your draft report[1] you identify that:

> The tables of events in Appendix B represent the best information available at the time this report was prepared.

and that:

> The relatively severe transients have been identified, but the total population of transients remains unknown.

We believe that the Rancho Seco event might add some additional insight because of the following:

1. The event apparently included a high pressure transient (greater than 2400 psi) and the safety valve(s) on the primary were challenged.
2. The failure of the "X" power supply apparently affected several valve controllers producing conditions allowing the overpressurization. Additionally, as the IE RO:TAB analysis (Enclosure 2) notes, the controllers do not fail open or closed.

[1] Draft NUREG-0667, Transient Response of Babcock & Wilcox-Designed Reactors, dated April 2, 1980.

THIS DOCUMENT CONTAINS POOR QUALITY PAGES

If you have any questions regarding this matter, please contact Jim Creswell on extension 29560.

/s/

Carl Michelson, Director
Office for Analysis and Evaluation
of Operational Data

Enclosures:
1. Abnormal Report No. 74-3
2. IE RO:TAB Analysis

cc w/enclosures:
C. Berlinger
R. Bernero
D. Eisenhut
R. Hartfield
E. Jordan
C. Johnson

Distribution:
Central File
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DATE	4/ /80	4/ /80				

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Enclosure 1

SMUD

SACRAMENTO MUNICIPAL UTILITY DISTRICT ☐ 6201 S Street, Box 15830, Sacramento, California 95813; (916) 452-[illegible]

September 20, 1974

Director
Directorate of Licensing
U. S. Atomic Energy Commission
Washington, D. C. 20545

AEC Docket No. 50-312
Rancho Seco Nuclear Generating
Station, Unit No. 1
Abnormal Occurrence No. 74-3

Dear Sir:

Enclosed is a copy of Abnormal Report No. 74-3 relative to inverter S1D.

Sincerely yours,

J. J. Mattimoe

J. J. Mattimoe
Assistant General Manager
and Chief Engineer

Enclosure

cc: R. H. Engelken, Region V

50-312

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ABNORMAL OCCURRENCE REPORT

DOCKET NO. 50-312-74-3

Reporting Date: September 18, 1974

Occurrence Date: September 10, 1974 Time: 0558

Facility: Rancho Seco Nuclear Generating Station
Unit No. 1
Clay Station, California

Identification of Occurrence:

Reactor Coolant System overpressure during heatup

Condition Prior to Occurrence:

The plant was at refueling shutdown completing final testing prior to initial criticality.

Description of Occurrence:

On the morning of September 10, 1974, the reactor coolant system temperature was being increased within the limits of the heatup curve established in the Technical Specifications. The system was stabilized at 355°F and 1700 psig to insert cocked rods and remove all the shutdown bypass keys from the reactor protection instrumentation as required by the operating procedure. After removing this shutdown bypass inhibits, the reactor system was in the process of increasing pressure by energizing the pressurizer heaters. The pressure was increased to 1950 psig which is above the low pressure trip of 1900 psig and all the reactor protection channels were reset. Concurrent with this heatup, a routine inspection of plant equipment was being conducted. As an operator was walking through the "D" inverter room, he heard a "snapping" sound coming from the internals of the S1D inverter cabinet. The operator informed the Shift Supervisor of the situation and he also inspected the inverter. The plant was not being effected by the inverter problem and since an electrical technician was expected to report to work within an hour, the Supervisor decided to have the technician work on the system.

When the technician arrived at 0500, he opened the cabinet and observed an inductor heat sink ground screw lug loose and it was sparking to the grounding bar. To prevent electrical transients and further degradation of the inductor lug, he advised that the S1D inverter be de-energized and he would start repair of the unit immediately. At 0537 with the electrical technician ready, S1D was secured.

The reactor operated smoothly until 0549 when the high reactor coolant pressure alarm sounded. This indicates at least 2255 psig reactor coolant pressure and the operator inspected his indicators. The wide range pressure recorder indicator was reading 1950 psig; the narrow range pressure recorder indicator ("B" loop pressure) was at 1950 psig, and the narrow range pressure indicator ("A" loop pressure) was at 2400 psig and rising sharply. The operator decided that the latter reading was correct and he stopped the high pressure makeup pump. The pressure quickly decreased and at 2100 psig the operator energized the pressurizer heaters to assure that the "A" loop pressure was responding correctly and that there was still a steam bubble in the pressurizer. Shortly after the heaters were energized, the pressure started to increase and he was confident that the "A" loop pressure indicator was responding to true reactor coolant conditions and that a steam bubble was still in the pressurizer. The reactor coolant pressure was decreased to 1900 psig by natural decay and then reduced to 1600 psig by request of the Plant Superintendent to centralize the loci point within the limits of the heatup curve until a complete analysis of the situation could be made.

Corrective Action Taken or Which Should be Taken:

An analysis of the pressure transient by the on-site Mechanical Engineer determined that the reactor coolant system did not sustain any adverse effect because:

1. The reactor coolant system temperature was above 100°F.
2. The pressurization occurred under isothermal conditions.
3. The system design pressure of 2500 psig was not exceeded.
4. These pressure/temperature conditions are permitted by the Technical Specifications for leak testing per Section 3.1.2.2(a).

In addition to the internal analysis, an external review of the Pressure-Time transient was conducted at Babcock & Wilcox Company in Lynchburg, Virginia. They concluded that the pressure spike had no harmful consequences on the reactor coolant system.

Designation of Apparent Cause: Design

Analysis of Occurrence:

One of the systems that the S1D inverter supplies is the reactor non-nuclear instrumentation "X" power. This bus feeds numerous signal converters, transmitters, indicators, controllers, recorders and selector stations. The specific modules on this bus which relate directly to the pressurization of the reactor coolant system are:

1. Seal Injection Flow Transmitter and Valve Controller. When the S1D inverter was secured the seal injection valve went to 50% open which is the neutral (zero signal) control position.

2. Makeup Flow Controller. When the SID inverter was secured, the makeup flow control valve went to 50% open which is the neutral (zero signal) control position.

3. Letdown Flow Transmitter. When the SID inverter was secured, the letdown flow valve went to 50% open which is the neutral (zero signal) control position.

4. Reactor Coolant Pressure Narrow Range Recorder B Loop. When the SID inverter was secured the recorder drive stopped and the indicator stopped at the pressure indicated.

5. Reactor Coolant Pressure Wide Range Recorder B Loop. When the SID inverter was secured the recorder drive stopped and the indicator stopped at the pressure indicated.

6. Pressure Level Recorder. When the SID inverter was secured, the recorder drive stopped at the level indication then current.

The response of Items 1 and 2 above amounted to a greater quantity of water injection into the reactor coolant system than was being removed by Item 3. The increased inventory accumulated in the pressurizer with a resultant reactor coolant increase.

The pressure indicator/recorders (Items 4 and 5) were indicating a non-transient condition during the pressure increase. The pressurizer level recorder (Item 6) was also indicating a non-transient condition during the level and pressure increase. Only the alarm function and the indicator/recorder on "A" loop were operational and indicating the transient condition. The operator had had difficulty in making the loop "A" recorder properly ink its recorder trace prior to the occurrence. Although the recorder did properly indicate pressure throughout the event, the short period over which the pressure transient occurred made the visual display less apparent, particularly when the two other pressure recorders were indicating an apparent stable condition. The operator stopped the pressure increase at 2400 psig which is 350 psig above the allowable pressure at 355°F during heatup. The pressurization occurred over a period of approximately 22 minutes and the depressurization to 1900 psig was over a period of one hour and 30 minutes. The temperature during transient did not change from the initial 355°F. As described under "Corrective Action Taken," a thorough analysis of the transient revealed no adverse effects to the reactor coolant system.

Equipment I.D.:

The SID inverter is manufactured by Static Products of Garland, Texas, Model No. SP-DC120-611-250-60. The inductor which had a loose screw on the heat sink ground was in tray 2-5, Section A-9, which is part of the 3φ switching bridge and is Part No. 044-013 A012-1.

Action Required to Prevent a Reoccurrence:

1. Operating personnel have been issued a more detailed explanation of what indication will be lost upon de-energizing "X" power.

2. A design change has been initiated to provide dual power supplies to "X" power by means of rapid automatic switching.

3. Until material and installation of Item 2 above can be accomplished, a manual switch has been installed to provide for manual switching of "X" AC power to an alternate power supply (lighting) when required. The "X" DC power has been split into two separate auctioneered power supplies either of which can carry the entire "X" load.

Failure Data: None

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SACRAMENTO MUNICIPAL UTILITY DISTRICT 1708 59th Street, Box 15830, Sacramento, California 95813; (916) 452-3211

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September 10, 1974

Mr. E. L. Engelken
Director of Regulatory Operations
Region V
AEC Regulatory Operations Office
1990 No. California Boulevard
Walnut Creek Plaza, Suite 202
Walnut Creek, California 94596

RECEIVED SEP 11 1974 REGION V DIR. OF REG. OPERATIONS

Re: AEC Docket No. 50-312
Rancho Seco Nuclear Generating
Station, Unit No. 1
Abnormal Occurrence No. 74-3

Dear Mr. Engelken:

This is to report Abnormal Occurrence No. 74-3.

During a partial loss of instrumentation with the Reactor Coolant System approximately 335°, system pressure reached 2400 pounds before action was taken to reduce it. This pressure temperature relationship exceeded the allowable pressure temperature relation by approximately 500 pounds. Pressure was subsequently reduced to an allowable value of 1900 psig.

Sincerely yours,

R. J. Rodriguez

R. J. Rodriguez
Plant Superintendent
Rancho Seco Unit 1
Clay Station, California 95638

RJR:sal

PROPOSED IE BULLETIN: "REACTOR COOLANT SYSTEM OVERPRESSURIZATION DUE TO LOSS OF NON-NUCLEAR INSTRUMENTATION AT THE RANCHO SECO REACTOR FACILITY." - DOCKET NO. 50-312

In response to your review request,[1] we offer the following comments on the subject proposed bulletin.

Our review of the licensee's report on the balance of plant (BOP) instrumentation malfunction and our discussions with knowledgeable people from Babcock & Wilcox (B&W), the licensee, and other B&W type nuclear facilities, lead us to conclude that the absence of an alternate source of electrical power for the BOP instrumentation systems at Rancho Seco was the primary cause for the overpressurization of the reactor coolant system which is the subject of the above bulletin. We have been informed by B&W that the above design (i.e., no back-up power source) of the BOP instrumentation is unique to the Rancho Seco Station. In view of this information, we recommend that no IE bulletin on this matter be issued to other B&W nuclear facilities. Additionally, since other nuclear facility designs (i.e., Westinghouse, Combustion Engineering, General Electric, and the like) do not utilize Bailey instruments in the affected systems, we do not expect the problem to occur at these facilities.

The bases for our conclusions and recommendations are discussed below.

In brief, the licensee reports[2] that the Bailey instruments which are utilized in the non-safety related [illegible] systems [illegible] either in the "as is" or "mid-range" position following a loss of system excitation voltage. It is significant [illegible] that all Bailey instruments used [illegible]. (i.e., [illegible] or "full [illegible] position.)

As a result of the series of events reported by the licensee[2], revisions to the electrical power system are planned. These include: (1) installation of an alternate source of electrical power, and (2) installation of a transfer switch to provide an automatic transfer of the BOP instrumentation load to the alternate power source if the preferred source of electrical power is lost.

1/ Action Item Control Form [illegible]

2/ Abnormal Occurrence Report 50-312/74-3 - Reactor Coolant System Overpressurization During [illegible], [illegible] September [illegible], 1974, Rancho Seco

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We recommend that the responsible inspector for Rancho Seco verify that the above modifications have been installed. With regard to other B&W facilities, we suggest that the appropriate regional inspectors confirm that these licensees have been apprised of the problem through B&W site problem report. (SPR No. 220)

The B&W reactor facilities in question include:

Arkansas Unit 1
Crystal River Unit 1
Davis Besse
Three Mile Island Units 1 and 2
Oconee Units 1, 2, and 3

Should you have any questions concerning the above findings, please contact Vince Thomas on extension 7421.

Original signed by
V. Thomas for

Karl V. Seyfrit, Chief
Technical Assistance Branch, RO

cc: B. H. Grier

CGO:724/F50000HI

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VDThomas:sga	KVSeyfrit
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