EVALUATION OF LICENSEE'S COMPLIANCE

WITH THE NRC ORDER DATED MAY 7, 1979

SACRAMENTO MUNICIPAL UTILITY DISTRICT

RANCHO SECO NUCLEAR GENERATING STATION

DOCKET NO. 50-312

578232

June 27, 1979

7907300435

INTRODUCTION

By Order dated May 7, 1979, (the Order) the Sacramento Municipal Utility District (SMUD or licensee) was directed by the NRC to take certain actions with respect to Rancho Seco Nuclear Generating Station. Prior to this Order and as a result of a preliminary review of the Three Mile Island Unit No. 2 (TMI-2) accident, the NRC staff initially identified several human errors that contributed significantly to the severity of the event. All holders of operating licenses were subsequently instructed to take a number of immediate actions to avoid repetition of these errors, in accordance with bulletins issued by the Commission's Office of Inspection and Enforcement (IE). Subsequently, an additional bulletin was issued by IE which instructed holders of operating licenses for B&W designed reactors to take further actions, including immediate changes to decrease the reactor high pressure trip point and increase the pressurizer power-operated relief valve (PORV) setting.*

The NRC staff identified certain other safety concerns that warranted additional short-term design and procedural changes at operating facilities having B&W designed reactors. Those were identified as items (a) through (e) on page 1-7 of the "Office of Nuclear Reactor Regulation Status Report to the Commission" dated April 25, 1979. After a series of discussions between the NRC staff and the licensee concerning possible design modifications and changes in operating procedures, the licensee agreed, in a letter dated April 27, 1979, to perform promptly certain actions. The Commission found that operation of the plant

*[IE Bulletins Nos. 79-05 (April 1, 1979), 79-05A (April 5, 1979), and 79-05B (April 21, 1979) apply to all B&W facilities.]

should not be resumed until actions described in paragraphs (a) through (e) of paragraph (1) of Section IV of the Order were satisfactorily completed.

Our evaluation of the licensee's compliance with items (a) through (e) of paragraph (1) of Section IV of the Order is given below. In performing this evaluation we have utilized additional information provided by the licensee on May 14, 22, 24, 29, 30, and June 6, 1979, and numerous discussions with the licensee's staff. Confirmation of design and procedure changes was made by members of the NRC staff at the Rancho Seco site. An audit of the Rancho Seco reactor operators was also performed by the NRC staff to assure that the design and procedure changes were understood and were being correctly implemented by the operators.

EVALUATION

Item a

It was ordered that the licensee take the following action:

> "Upgrade the timeliness and reliability of delivery from the Auxiliary Feedwater System by carrying out actions as identified in Enclosure 1 of the licensee's letter of April 27, 1979."

The Rancho Seco auxiliary feedwater (AFW) design has one turbine/motor tandem drive pump (P-318) that is automatically actuated and controlled independent

578234

of offsite power, and one motor-driven AFW pump (P-319) that is automatically started, but must be manually transferred to a vital AC bus if offsite power is lost. The turbine/motor driven pump will be manually started, according to procedure, from a vital AC bus if the turbine drive fails. By reference above to Enclosure (1) of the licensee's letter of April 27, 1979, it was ordered that the licensee:

"1. Review procedures, revise as necessary and conduct training to ensure timely and proper starting of motor driven auxiliary feedwater (AFW) pump(s) from vital AC buses upon loss of offsite power."

The licensee has developed Section 7.5 of Operating Procedure A.51 ("Auxiliary Feedwater System") to provide specific direction for the operator on the steps required to load motor driven pump P-319 on nuclear service bus 4A and to secure the steam to the turbine on the dual-drive pump P-318, in the event of inoperability of the steam drive, and load the motor drive on nuclear service bus 4B. Bypass keys are required to complete the connection of the auxiliary feedwater pump motors to the diesel powered buses (nuclear service buses 4A and 4B); these keys are available in the office adjacent to the control room. Emergency Procedure D.1 ("Load Rejection") directs the operator to use Operating Procedure A.51 if main feed pump operation cannot be maintained. The NRC staff verified that the operators are knowledgeable in the procedure for loading the AFW pumps on the vital AC buses. The NRC staff concludes that the licensee has adequate procedures and the operators are trained to start the AFW system from diesel powered buses upon loss of offsite power or load rejection and therefore, is in compliance with this part of the Order.

578235

It was also ordered that:

"2. To assure that AFW will be aligned in a timely manner to inject on all AFW demand events when in the surveillance test mode, procedures will be implemented and training conducted to provide an operator at the necessary valves in phone communications with the control room during the surveillance mode to carry out the valve alignment changes upon AFW demand events."

Surveillance Procedures SP 210.01A and SP 210.01B are used for the quarterly surveillance and inservice testing of auxiliary feed pumps P-318 and P-319, respectively. These procedures have been revised to include the following statement; "Station an operator at FWS-055, auxiliary feedwater system full flow recirculation valve in continuous communication with the control room until FWS-055 is secured closed at the completion of this test." In addition to the above procedure revisions, the licensee has added FWS-492 (bypass valve for FWS-055) to the "Locked Valve List" (SP 214.03). The licensee has also incorporated independent verification of valve lineups following surveillance testing and/or maintenance of the AFW system.

The NRC staff has reviewed SP 210.01A and SP 210.01B to verify that the procedures contain specific directions to return each valve that was operated during the conduct of the surveillance test to its proper position. The local operator has to close a valve (FWS-Ø55) when so instructed by the control room operator or if he loses communication with the control room. The NRC staff has verified that the operators are familiar with this test procedure. We conclude that the licensee has adequate procedures to assure that AFW will be aligned in a timely manner to inject on all AFW demand events when in the surveillance test mode and therefore, is in compliance with this part of the Order.

It was ordered that:

> "3. Procedures will be developed and implemented and training conducted to provide for control of steam generator level by use of safety grade AFW bypass valves in the event that ICS steam generator level control fails."

The licensee has developed Emergency Procedure D.14 ("Loss of Steam Generator Feed") that describes the symptoms that would result from a loss of main feedwater control that may have been caused by an integrated control system (ICS) failure. The procedure has been reviewed by the NRC staff. The operator is directed to restore feedwater to the steam generators by one of three methods. The preferred method is described in Section 7.7 of Operating Procedure A.51 ("Auxiliary Feedwater System"). Section 7.7 directs the operator to: close the ICS controlled AFW control valves; start the AFW pumps; and maintain the steam generator levels, specified in the procedure, by manually operating the motor driven AFW bypass valves from the control room. In this mode the pumps and valves will operate independent of the ICS. The operator is provided with AFW flow rate and steam generator level indications in the control room for each steam generator.

578237

Since the AFW bypass valves will fully open on a safety features actuation signal (SFAS)*, the operator is provided with instructions on how to take manual control of the valves after a SFAS. NRC staff has conducted an audit of the operator training and verified that the operators have been trained to carry out those procedures.

The NRC staff concludes that the licensee has developed adequate procedures and operator training to control AFW flow to the steam generators to specified values independent of the ICS, should a failure of the ICS occur, and therefore, is in compliance with this part of the Order.

It was also ordered:

"4. Verification that Technical Specification requirements of AFW capacity are in accordance with the accident analysis will be conducted. Pump capacity with mini flow in service will also be verified."

The licensee has conducted the verification that Technical Specification requirements of AFW capacity are in accordance with the accident analysis for the Rancho Seco Nuclear Station. The Technical Specification states, as a

*[The safety features actuation system (SFAS) monitors variables to detect loss of reactor coolant system boundary integrity. Upon detection of "out-of-limit" conditions of these variables, it initiates emergency core cooling (ECC) which consists of high pressure injection (HPI) and low pressure injection (LPI), Reactor Building cooling and isolation, and Reactor Building spray systems. Additionally, it starts diesel generators GEA and GEB, which are in standby redundance with the nuclear service buses 4A and 4B.]

limiting condition for operation, capability to supply feedwater at a process flow rate corresponding to a decay heat level of 4.5 percent of full reactor power from at least one of the following means:

(a) a condensate pump and a main feed pump, or

(b) a condensate pump, or

(c) an auxiliary feedwater pump.

A letter from Babcock & Wilcox to the licensee, dated May 16, 1979, states that it has performed an analysis of the required AFW flow rate for the Rancho Seco Plant which shows that a decay heat level of 4.5 percent of rated power, plus the heat input from the RCPs, will require a total flow rate to either or both steam generators of approximately 760 gpm.

Each of the two AFW pumps are sized to deliver 780 gpm to steam generators with 60 gpm mini flow in service. This pump capacity exceeds the minimum required AFW flow rate in the Rancho Seco safety analysis and Technical Specifications. AFW pump capacity, with mini flow in service, has been verified by performing the quarterly "AFW System Surveillance Test" and the "Auxiliary Feedwater Flow Indicator Functional Test" (STP 612). The results of these tests demonstrated that each of the two AFW pumps has the capability to deliver a minimum of 780 gpm into the steam generators, with mini flow in service. The licensee will reconfirm the minimum AFW flow rate to the steam generators in a test immediately following startup.

578239

Based on our review of the AFW flow rate test results, performed to date, we conclude that the licensee is in compliance with this part of the Order.

It was also ordered that:

"5. Modifications will be made to provide verification in the control room of AFW flow to each steam generator."

To verify that AFW is being pumped to the steam generators, the licensee has installed Clampitron Flowmeters on both of the AFW injection flow paths, downstream of the AFW control valves, so that the actual flow rate to each steam generator will be measured. The Clampitron Flowmeters consists of transducers, attached to the AFW piping, connected to a flow display computer. On command from the flow display computer, the transducer transmits an ultrasonic beam through the water inside the pipe and the velocity of the beam, as affected by AFW flow, is analyzed by the flow display computer, which calculates the AFW flow rate in gpm. The AFW flow rate is displayed in the control room. A calibration test (STP-612) was conducted by the licensee to functionally test the performance of the flowmeters. Performance of this test demonstrated that the indicated flow rate agreed with the calculated flow rate within the ±20% acceptance criteria specified in the procedure.

Based on our review of this design modification and test results, we conclude that the licensee is in compliance with this part of the Order.

578240

It was also ordered that the licensee:

"6. Review and revise, as necessary, the procedures and training for providing alternate sources of water to the suction of the AFW pumps."

Control room alarms are available to alert the operator to perform the manual transfer of the AFW supply source from the condensate storage tank (CST) to the plant reservoir. The CST is designed to seismic Category I criteria. The licensee has reviewed and revised his Emergency Procedures D.10 ("Loss of Reactor Coolant Flow/RCP Trip"), D.14 ("Loss of Steam Generator Feed"), and Operating Procedure A.51 ("Auxiliary Feedwater System") to provide guidance for the operator to obtain an alternate source of water for the suction of the AFW pumps. The revised procedures require the operator to break condenser vacuum when the level reaches a level alarm point of approximately 29 feet and to shift the AFW pump suction to the plant reservoir when the CST level is down to a second alarm point of approximately 3 feet from the bottom of the tank. The capacity of the CST is large enough to provide cooling for about 24 hours before this transfer is required. The shifting to an alternate source of AFW pump suction is accomplished by manually operating four isolation valves at a local valve station. The operator has about 40 minutes to effect the transfer. The NRC staff has reviewed the revised Emergency Procedures D.10 and D.14 and Operating Procedure A.51 and concludes that these procedures provide sufficient guidance to the operator for a timely shifting to an alternate water source for the AFW pumps, before the CST is emptied.

578241

The NRC staff has verified that the control room operators are properly trained to carry out these procedures. We conclude that the licensee has complied with the requirements of this part of the Order.

It was also ordered that:

"7. Design review and modification, as necessary, will be conducted to provide control room annunciation for all auto start conditions of the AFW system."

The licensee has provided indication for all auto start conditions of the AFW system on an annunciation panel inside the control room. The conditions which will actuate the annunciator are:

(a) loss of all reactor coolant pumps, or

(b) low discharge pressure (850 psig) on both main feedwater pumps, or

(c) manual start of the motor driven AFW pump.

A safety features actuation signal, which will also automatically start AFW, had already been annunciated in the control room before the current modifications. Based on our review of this design modification, we conclude that the licensee is in compliance with this part of the Order.

It was ordered that:

578242

"8. Procedures will be developed and implemented and training conducted to provide guidance for timely operator verification of any automatic initiation of AFW."

The conditions that will automatically initiate auxiliary feedwater are adequately described in Operating Procedure A.51, ("Auxiliary Feedwater System"). The operators are directed, as an immediate action, to verify that the AFW flow has automatically started on loss of both main feedwater pumps in Emergency Procedure D.14 ("Loss of Steam Generator Feed") and on loss of all reactor coolant pumps in Emergency Procedure D.10 ("Loss of Reactor Coolant Flow/RCP Trip"). Both procedures require the following immediate actions by the operator: verify that the auxiliary feedwater pumps have automatically started; that there is flow to the steam generators; and that the proper steam generator levels are being maintained. The NRC staff has performed an audit and verified that the operators are trained in these procedures.

Based on review of these procedures, we conclude that the licensee has provided guidance for timely operator verification of any automatic initiation of AFW and therefore, is in compliance with this part of the Order.

It was also ordered:

"9. Verification will be made that the air operated level control valves (a) Fail to the 50% open position upon loss of electrical power to

578243

the electrical to pressure converter, and (b) Fail to the 100% open position upon loss of service air. The AFW bypass valves are safety grade."

The licensee has completed its verification test for the failure mode of the air operated level control valves. The test results show that both air operated level control valves fail to the 100% open position on loss of air pressure at the valve operators. On tests for loss of control signal to the electric to pressure converters, one level control valve failed to the 50% open position and the other one failed to the 60% open position, which are acceptable. The AFW bypass valves are safety grade, motor-operated valves which are operated independently from the ICS as discussed in Part 3 above. Based on our review of the test results on the air operated level control valves and the safety grade design of the bypass valves, we conclude that the licensee is in compliance with this part of the Order.

Based upon our evaluation, we conclude that the licensee has upgraded the timeliness and reliability of delivery from the AFW system by carrying out the actions identified in Enclosure 1 of the licensee's letter of April 27, 1979, and therefore, is in compliance with Item (a) of the Order.

Item (b)

It was ordered that the licensee:

578244

"Develop and implement operating procedures for initiating and controlling auxiliary feedwater independent of Integrated Control System (ICS) control."

We have reviewed the revised procedures for the AFW system to assure that there is sufficient guidance for the operator to actuate the system if the automatic initiation failed, and to control steam generator levels at the required values. The review of the procedures focused on verifying that the operator is directed to observe the proper instruments and that the operator is directed to maintain specific values of parameters by manual control, such as steam generator levels. The review also determined that the operator should confirm the validity of the instrument readings of certain key parameters, such as steam generator levels. The necessary modifications to the procedures to satisfy these requirements were presented to the licensee, and the NRC staff has verified that the modifications have been incorporated in the procedures. (See further discussion of these procedures in part 3 of Item (a.)

The licensee will conduct a startup test at low power (<15%) to demonstrate the capability to provide and control flow to the steam generators using the AFW bypass valves.

During the visit to the site, the NRC staff walked through the AFW procedures with the operators to evaluate whether the procedures were functionally adequate. In addition, the NRC staff audited a sample of Rancho Seco operators to determine if they were familiar with the revised procedures and could implement them correctly. Based on the NRC staff audit, we conclude that the revised procedures and operator training are satisfactory and therefore, the licensee is in compliance with Item (b) of the Order.

578245

Item (c)

The original Rancho Seco design did not have any direct reactor trips that would be initiated by a malfunction in the secondary system. To obtain an anticipatory reactor trip (rather than delaying the trip until a primary system parameter exceeded its trip setting) the licensee committed to install a hard-wired, control-grade, reactor trip on loss of all main feedwater and/or turbine trip. The Order requires that the licensee:

"Implement a hard-wired control-grade reactor trip that would be actuated on loss of main feedwater and/or turbine trip."

The licensee has added control-grade circuitry to Rancho Seco, which is designed to provide an automatic reactor trip when either the main turbine trips or all main feedwater is lost. The purpose of the anticipatory trip is to minimize the potential for opening of the power-operated relief valve (PORV) and/or the safety valves on the pressurizer. The licensee has indicated that this new circuitry meets this objective by providing a reactor trip during the incipient stage of the related transients (turbine trip and/or loss of main feedwater).

The main turbine trip is sensed by an existing, normally deenergized relay in the main turbine/generator protection system. The relay is energized by the protective trips of the turbine and/or generator. Power is supplied by an onsite battery source.

The loss of all main feedwater is sensed by two newly installed pressure switches (one in each of the two main feedwater pump discharge lines). The pressure switches actuate (close) on low pressure in the header. Power is supplied by the same onsite battery source. In order to prevent an inadvertent reactor trip during startup or shutdown, the loss of all main feedwater trip input is cut-out of the circuitry by a keylock switch. The key for this switch is maintained in the custody of the shift supervisor and is located in the control room. When the switch is placed in the "cut-out" position, it is annunciated on the main control board. The operating procedures specify when the switch is placed in the "normal" or "cut-out" position.

Either signal (turbine trip or loss of all main feedwater) will actuate a reactor trip relay, which in turn provides an input to both of the shunt coils of the AC reactor trip breakers. Energizing both of the shunt coils causes a reactor trip.

The licensee has analyzed this additional circuitry with respect to its independence from the existing reactor trip system. They have stated that the shunt coil is part of the existing AC reactor trip breaker. Each shunt coil is powered by a separate Class IE 125 VDC supply and operates independently from the 120 VAC undervoltage trip coil which receives the safety-grade reactor trip signal.

An NRC inspector has confirmed that the check-out tests for this circuitry have been completed successfully. In addition, the licensee has committed to perform a monthly periodic test on the added circuitry in order to demonstrate its ability to open the AC reactor trip breakers via the shunt coil.

Based on our review of the implementation of the trip circuitry, with respect to its independence from the existing reactor trip circuitry, we conclude that this addition will not degrade the existing reactor protection system design.

Based on the licensee's design modifications and commitment to perform a monthly test on the new circuitry, we conclude that there is reasonable assurance that the system will perform its function.

On the basis of the evaluation above, we conclude that the licensee has complied with the requirements of Item (c) of the Order.

Item (d)

This item in the Order requires the licensee to:

> "Complete analyses for potential small breaks and develop and implement operating instructions to define operator action."

In the licensee's letter of April 27, 1979, the licensee committed to providing the analyses and operating procedures of this requirement.

Babcock and Wilcox, the reactor vendor for the Rancho Seco plant, submitted analyses entitled, "Evaluation of Transient Behavior and Small Reactor Coolant System Breaks in the 177 Fuel Assembly Plant" and supplements to these analyses (References 1 through 6). The major parameters used in this generic study

578248

bound the Rancho Seco plant. The staff evaluation of the B&W generic study has been completed and the results of the evaluation will be issued as a NUREG report in June 1979.

A principal finding of our generic review is a reconfirmation that Loss-of-Coolant Accident (LOCA) analyses of breaks at the lower end of the small break spectrum (smaller than 0.04 sq. ft.) demonstrate that a combination of heat removal by the steam generators, the high pressure injection system and operator action ensure adequate core cooling. The AFW system used to remove heat through the steam generators has been modified to enhance its reliability as discussed in item (a). The high pressure injection system is capable of providing emergency core cooling even at the safety valve pressure setpoint. The ability to remove heat via the steam generators has always been recognized to be an important consideration when analyzing very small breaks. Separate sensitivity analyses were performed assuming permanent loss of all feedwater (with operator initiation of the high pressure injection system at 20 minutes) and loss of feedwater for only the first 20 minutes of the accident for breaks of 0.01 sq. ft. Reactor core uncovery is not predicted for these events. The calculated peak cladding temperature was less than 800°F, well below the 10 CFR 50.46 requirement of 2200°F. These results are applicable to Rancho Seco considering the ability to manually start the redundant AFW pumps from the control room, assuming failure of automatic AFW actuation.

Another aspect of the study was the assessment of recent design changes on the lift frequency of pressurizer safety and relief valves. The design changes included: a change in the setpoint of the pressurizer power-operated relief valve (PORV) from 2255 psi to 2450 psi; change in the high pressure reactor trip setpoint from 2355 psi to 2300 psi; and the installation of an anticipatory reactor trip on turbine trip and/or on loss of all main feedwater. In the past, during the turbine trip or loss of feedwater transients, the PORV lifted. With the design changes the initial pressure increase of these transients do not result in lifting of this valve. However, the consequent depressurization could initiate safety injection which in turn could repressurize the system and lift the relief valve. It is expected that the operator would terminate HPI before the relief valve or safety valves lift, since the 50°F subcooling criteria would be satisfied at pressures below the PORV setpoint. Also, lifting of both the PORV and safety valves might occur in the case of control rod withdrawal or inadvertent boron dilution transients, using the normally conservative assumptions found in the Chapter 15 safety analyses. The above design changes do not effect the lift frequency of the valves for these Chapter 15 safety analyses.

Based on our review of the small break analyses presented by B&W, the staff has determined that a loss of all main feedwater with (a) an isolated PORV, but safety valves opening and closing as designed, or (b) a stuck open PORV does not result in core uncovery, provided either AFW or 2 HPI pumps is initiated within 20 minutes. Based on the acceptable consequences calculated for small break LOCAs and loss of all main feedwater events coupled with the expected reliability of the AFW and HPI systems, we conclude that the licensee has complied with the analyses portion of Item (d) of the Order.

578250

To support longer term operation of the facility, requirements will be developed for additional and more detailed analyses of loss of feedwater and other anticipated transients. More detailed analyses of small break LOCA events are also needed for this purpose. Accordingly, the licensee will be required to provide the analyses discussed in Section 8.4.1 and 8.4.2 of the recent NRC "Staff Report of the Generic Assessment of Feedwater Transients in Pressurized Water Reactors Designed by the Babcock and Wilcox Company" (NUREG 0560). Further details on these analyses and their applicability to other PWRs and BWRs will be specified by the staff in the near future. In addition, to assist the staff in developing more detailed guidance on design requirements of relief and safety valve reliability during anticipated transients, as discussed in Section 8.4.6 of the NUREG report, the licensee will be required to provide analyses of the lift frequency and mechanical reliability of the pressurizer relief and safety valves of the Rancho Seco facility.*

The B&W analyses show that some operator action, both immediate and followup, is required under certain circumstances for a small break accident. Immediate operator action is defined as those actions committed to memory by the operators which must be carried out as soon as the problem is diagnosed. Follow-up actions require operators to consult and follow the steps in written and approved procedures. These procedures must always be readily available in the control room for the operators' use. Guidelines were developed by B&W to assist the operating B&W facilities in the development of emergency procedures for the small break accident.

578?54

The "Operating Guidelines for Small Breaks" were issued by B&W on May 5, 1979 and reviewed by the NRC staff. Revisions recommended by the staff were incorporated in the guidelines. In response to these guidelines, the staff at Rancho Seco made substantial revisions to Emergency Procedure D.5 ("Loss of Reactor Coolant/ Reactor Coolant System Pressure") and Operating Procedure B.4 ("Plant Shutdown and Cooldown"). These procedures define the required operator action in response to a spectrum of break sizes for a loss-of-coolant accident in conjunction with various equipment availability and failures.

Emergency Procedure D.5 (EP D.5) is divided into three sections. The first section deals with a small leak within the capability of a makeup pump. In this case, the operators proceed with an orderly plant shutdown unless pressurizer or makeup tank levels fall below prescribed limits. If these limits are exceeded the reactor is manually tripped and high pressure injection is initiated.

The second section of EP D.5 defines the required operator action for a small break not within the capability of a makeup pump. This section provides the operator with the guidance necessary to achieve a safe hot shutdown condition for a variety of degraded conditions. If all feedwater is lost, a heat removal path is established by the high pressure injection system through the break and the pressurizer power-operated relief valve or the safety valves. Once feedwater is reestablished, the steam generators can be used as a heat sink. If the reactor coolant pumps are not available, the operator is directed to Operating Procedure B.4 (OP B.4) which defines the actions necessary to cool down the plant by natural circulation. Additional guidance is provided in OP B.4 if natural circulation is not immediately achieved.

The third section of EP D.5 defines the actions necessary in the event of a large rupture. In this case the system depressurizes to the point of low pressure injection.

For all cases in which high pressure injection is manually or automatically initiated, the operators are specifically instructed in EP D.5 to maintain maximum HPI flow unless one of the following criteria are met:

(1) The LPI system is in operation and providing cooling at a rate in excess of 1000 gpm and the situation has been stable for 20 minutes, or

(2) All hot and cold leg temperatures are at least 50 degrees below the saturation temperature for the existing RCS pressure. If the 50 degrees subcooling cannot be maintained after HPI cutoff, HPI shall be reactuated.

A requirement to determine and maintain 50°F subcooling has been incorporated in all other procedures in which HPI has been manually or automatically initiated. These procedures include, "Steam Supply System Rupture," and "Loss of Steam Generator Feedwater." Each of these procedures, in addition to the "Loss of Reactor Coolant/Reactor Coolant System Pressure" procedure, provide additional instructions to the operators in the event of faulty or misleading indications.

578253

A subsequent action statement directs the operators to check alternate instrumentation channels to confirm key parameter readings. The Rancho Seco staff has made revisions to all of their emergency procedures to include this requirement.

If feedwater is not initially available following a transient or accident, core cooling is maintained by flow from two HPI pumps and relief through the PORV, which is opened by the operator. B&W has performed studies that show that density differences between the downcomer and reactor core will cause recirculation flow between the core exit and downcomer via the vent valves. Mixing of the hot core exit water with the cold HPI water will provide sufficiently warm vessel temperatures to preclude any significant thermal shock effects to the vessel. Under these conditions with no circulation of water from the steam generators, the cold leg thermocouple (located upstream of the reactor coolant pump) does not provide a satisfactory indication of the vessel temperature. B&W has recommended using the core exit thermocouples as a measure of vessel temperature, based on B&W analyses that conservatively show that the vent valves will open at temperature differences between the core exit and downcomer of less than 150°F. They have also proposed a more appropriate pressure-temperature limit curve for the vessel that reflects allowable stresses under these faulted conditions (no feedwater).

The NRC staff has reviewed these guidelines and finds them acceptable because of the expected recirculation through the vent valves and the vessel stress limits used. The licensee has incorporated these revised guidelines in his procedures for loss of all feedwater. Subsequent restoration of AFW would depressurize the reactor coolant system to below 600 psi where pressure vessel integrity is assured for any reasonable thermal transients that might subsequently occur. We conclude that further reliability analyses are needed as part of the long-term requirements of the Order to confirm that AFW can be restored (if lost) in a reasonable period of time. B&W has agreed to provide a detailed thermal-mechanical report on the behavior of vessel materials for these extreme conditions, to be applicable generically to the Oconee class of plants, which includes Rancho Seco.

The "Loss of Reactor Coolant/Reactor Coolant System Pressure" procedure was reviewed by the NRC staff to determine its conformance with the B&W guidelines. Comments generated as a result of this review were incorporated in a further revision to the procedure. A member of the NRC staff walked through this emergency procedure in the Rancho Seco control room. The procedure was judged to provide adequate guidance to the operators to cope with a small break loss-of-coolant accident. The instrumentation necessary to diagnose the break, the indications and controls required by the action statements, and the administrative controls which prevent unacceptable limits from being exceeded are readily available to the operators. We conclude that the operators should be able to use this procedure to bring the plant to a safe shutdown condition in the event of a small break accident.

An audit of seven of 14 licensed operators and senior operators assigned to shift duty (22 total licensed personnel) was conducted by the NRC staff to determine the operators' understanding of the small break accident, including how they are required to diagnose and respond to it. The Rancho Seco staff has conducted special training sessions for the operators on the concept and

578255

use of EP D.5. The audit revealed that, except for one deficiency, the operators had sufficient knowledge of the small break phenomenon and the requirements of the procedure. This deficiency, verification of natural circulation, was brought to the attention of the plant staff. Each licensed individual received additional training in this area by the plant training organization and General Physics Corporation. They also received training on the revisions made to EP D.5 as a result of the NRC review. This additional training has been completed and verified by the NRC staff.

The audit of the operators also included questioning about the TMI-2 incident and the resulting design changes made at Rancho Seco. The discussions covered the initiating events of the incident, the response of the plant to the simultaneous loss of feedwater and small break LOCA (PORV stuck open), and the operational actions that were taken during the course of the incident. We identified a deficiency in interpreting the initial sequence of the TMI-2 incident on the part of several of the operators. Additional training has been conducted in this area by the plant staff and their consultant and verified by the NRC staff.

Otherwise, we found their level of understanding sufficient to be able to respond to a similar situation if it happened at Rancho Seco. We also concluded they have adequate knowledge of subcooling and saturated conditions and are able to recognize each in the primary coolant system by various methods. The AFW system was also discussed during the audit to determine the operators' ability to assure proper starting and operation of the system during normal conditions, as well as during adverse conditions such as loss of offsite power or loss of normal feedwater. The long term operation of the system was examined to evaluate the operators' ability to use available manual controls and water supplies. The level of understanding was found to be sufficient to assure proper short and long term AFW flow to the steam generators.

In addition to the oral audit conducted by the NRC, the licensee administered a written examination to all licensed personnel. Individuals scoring less than 90 percent on the exam will receive additional training and will not assume licensed duties until a score of at least 90 percent is attained on an equivalent, but different exam. The written exam and the grading was audited by the NRC staff and judged to be satisfactory. The staff will also review all subsequent results and records as part of the normal inspection function of the Rancho Seco requalification program. We conclude that there is adequate assurance that the operators at Rancho Seco have and will continue to receive a high level of training concerning the TMI-2 accident and the consequent impact at their station.

Based on the foregoing evaluation, we conclude that the licensee has complied with the requirements of Item (d) of the Order.

Item (e)

The Order requires that the licensee:

> "Provide for one senior licensed operator assigned to the control room who has had TMI-2 training on the B&W simulator."

578267

The licensee has confirmed that this item of the Order has been completed and has further committed that all reactor operators and senior reactor operators will have completed the TMI-2 simulator training at B&W by June 21, 1979.* This training consists of a class discussion of the TMI-2 event followed by a demonstration of the event on the simulator as it occurred and the proper actions that should be taken to control the accident. The class discussion is about four hours long and the remainder of the session is conducted on the simulator. The TMI-2 event, including operational errors, is demonstrated to each operator. The event is again initiated and the operators are given "hands-on" experience in successfully regaining control of the plant by several methods. Other transients which result in depressurization and saturation conditions are presented to the operators and they must maneuver the plant to a stable, subcooled condition.

Based on the above actions by the licensee, we conclude that the licensee is in compliance with Item (e) of the Order.

Conclusion

We conclude that the actions described above fulfill the requirements of our Order of May 7, 1979 in regard to Paragraph (1) of Section IV. The licensee having met the requirements of Paragraph (1) may restart Rancho Seco as provided by Paragraph (2). Paragraph (3) of Section IV of the Order remains in force until the long term modifications set forth in Section II of the Order are completed and approved by the NRC.

---

* This action has been completed and satisfies the long-term portion of the Order in this regard.

REFERENCES

1. Letter from J. H. Taylor (B&W) to R. J. Mattson (NRC) transmitting report entitled, "Evaluation of Transient Behavior and Small Reactor Coolant System Breaks in the 177 Fuel Assembly Plant," dated May 7, 1979.

2. Letter from J. H. Taylor (B&W) to R. J. Mattson (NRC) transmitting revised Appendix 1, "Natural Circulation in B&W Operating Plants (Revision 1)," dated May 8, 1979.

3. Letter from J. H. Taylor (B&W) to R. J. Mattson (NRC) transmitting additional information regarding Appendix 2, "Steam Generator Tube Thermal Stress Evaluation," to report identified in Item 2 above, dated May 10, 1979.

4. Letter from J. H. Taylor (B&W) to R. J. Mattson (NRC), providing an analysis for "Small Break in the Pressurizer (PORV) with no Auxiliary Feedwater and Single Failure of the ECCS," identified as Supplements 1 and 2 to Section 6.0 of report in Item 2, dated May 12, 1979.

5. Letter from J. H. Taylor (B&W) to R. J. Mattson (NRC), providing an analysis for "Small Break in the Pressurizer (PORV) with no Auxiliary Feedwater and Single Failure of the ECCS" identified as Supplements 1 and 2 to Section 6.0 of report in Item 2, dated May 12, 1979.

578269

6. Letter from J. H. Taylor (B&W) to R. J. Mattson (NRC), providing Supplement 3 to Section 6 of report in Item 2, dated May 24, 1979.

578260