EVALUATION OF LICENSEE'S COMPLIANCE

WITH THE NRC ORDER DATED MAY 16, 1979

FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT NO. 3 NUCLEAR GENERATING STATION

DOCKET NO. 50-302

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INTRODUCTION

By Order dated May 16, 1979, (the Order), the Florida Power Corporation (licensee or FPC) was directed by the NRC to take certain actions with respect to Crystal River Unit No. 3 (CR-3). 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 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 Babcock & Wilcox (B&W) designed reactors to take further actions, including immediate changes to decrease the reactor high pressure reactor 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 May 1, 1979, to perform promptly certain actions. The Commission found that operation of the plant should not be resumed until actions described in paragraphs (a) through (e) of paragraph (1) of Section IV of the Order were satisfactorily completed.

*[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.]

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 16, and June 12, 15, 22 and 29, 1979, and numerous discussions with the licensee's staff. Confirmation of design and procedural changes was made by members of the NRC staff at the Crystal River site. An audit of the Crystal River reactor operators was also performed by the NRC staff to assure that the design and procedural 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 Emergency Feedwater System by carrying out actions as identified in Enclosure 1 of the licensee's letter of May 1, 1979."

The Crystal River emergency feedwater (EFW) system design has one turbine-driven pump that is automatically actuated and controlled independent of offsite power, and one motor-driven EFW pump that is automatically started if offsite power is available, but must be manually started on a vital AC bus if offsite power is lost. By reference above to Enclosure (1) of the licensee's letter of May 1, 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 emergency

feedwater (EFW) pump from engineered safeguards bus A upon loss of offsite power."

The licensee has revised EP-101 ("Unit Blackout") to provide the operators with a procedure for loading the motor-driven EFW pump on engineered safeguards bus 3A. This procedure will be used only if the following three conditions are met: (1) loss of offsite power, (2) the turbine-driven pump is not functioning, and (3) EFW is required. The procedure directs the operators to strip (remove) the following loads from bus 3A prior to loading the motor-driven EFW pump on the bus: (1) the decay heat removal pump, (2) building spray pump, (3) closed loop cooling pump, and (4) raw water pump. The loads stripped from the bus are not required for shutdown cooling during the period that EFW is used for decay heat removal. However, a redundant decay heat removal train would be available on the other emergency bus. When EFW is no longer needed, the motor-driven EFW pump would be removed from the bus and the other loads restored to the emergency bus.

The NRC staff performed an audit at the site and verified that the operators were knowledgeable in the steps of this procedure.

The NRC staff concludes that the licensee has adequate procedures and the operators are properly trained to start the motor-driven EFW pump from the diesel powered engineered safeguards bus 3A upon a loss of offsite power, and therefore, meets the requirements of this part of the Order.

It was also ordered that:

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

Surveillance Procedure SP-349 ("Emergency Feedwater System Operability Demonstration") directs the operators to close the EFW pumps' discharge valves to perform the test and then provides direction to reopen the valves to their normal operating positions following the test. These discharge valves are motor-operated valves that are controlled from the main control room; therefore, there is no need to station an operator at the valve locations during the surveillance testing. SP-349 requires that an operator determine that the EFW system valves are properly aligned and a second operator verifies the valve positions following the test. In addition to the independent verification of valve lineups required following surveillance testing, independent verification is also required upon completion of maintenance on the system. A valve lineup check list for the EFW system is included in SP-300 ("Operating Daily Surveillance Log").

The NRC staff has conducted an audit to verify that the operators are aware of the valve lineup requirements.

The NRC staff concludes the licensee has developed adequate procedures and has properly trained operators to verify correct valve alignments in the EFW system, and, therefore, is in compliance with this part of the Order.

It was ordered that:

3. "Emergency feedwater bypass valves are normally in the open position. Procedures have been developed and implemented to require the operator to take control of these valves upon failure of the ICS steam generator level control. If the ICS level control does not fail the operator will close the bypass valves. Those valves in the EFW system not locked in position are verified to be in the proper position on a daily basis. Training will be conducted on these revised procedures prior to June 1, 1979."

Emergency feedwater flow is normally controlled by the integrated control system (ICS) to maintain the required steam generator levels by actuation of the air-operated feedwater startup valves. An alternate path, independent of the ICS, is provided through the motor-operated EFW bypass valves. The bypass valves are normally maintained in the open position; however, following EFW activation, the operators must close the bypass valves and monitor steam generator levels and EFW flow to determine if the ICS is functioning properly. If the ICS fails to maintain the proper steam generator levels, the operator is directed to control the level by throttling flow with the bypass valves.

The licensee has modified the following emergency procedures to provide this guidance to the operators: EP-101 ("Unit Blackout"), EP-103 ("Loss of RC Flow/RC Pump Trip"), EP-106 ("Loss of Reactor Coolant or Reactor Coolant Pressure"), and EP-108 ("Loss of Steam Generator Feed").

The licensee has installed an ultrasonic flowrate meter system to provide control room indication of emergency feedwater flowrate in gallons per minute (gpm) to each steam generator. Each system consists of the ultrasonic flow transducer, mounted on the EFW piping, and the associated flow display computer, mounted locally. Flowrate indicators are also located in the control room on the main control board.

In addition to the directions for operator control of EFW flow if required, the licensee has provided for a daily verification of valve lineup in the EFW system in SP-300 ("Operating Daily Surveillance Log").

The NRC staff has conducted an audit at the site and verified that the operators are trained in these procedures. The NRC staff concludes that the licensee has provided adequate procedures and operator training to control the EFW system independent of the ICS, and is thus in compliance with this part of the Order.

It was also ordered that:

4. "The EFW pumps will be verified operable in accordance with the CR #3 Technical Specifications and Surveillance Procedures."

The Technical Specifications for CR-3 require a monthly test of the turbine-driven EFW pump to demonstrate its operability. The surveillance procedure requires running both of the EFW pumps with their discharge valves closed, with flow through the recirculation line, and measuring the discharge pressure of the pumps. We have reviewed the test procedure and find it acceptable. Satisfactory results of

this monthly surveillance test is an acceptable basis for demonstrating the operability of the EFW pumps and, therefore, we conclude that the licensee is in compliance with this part of the Order.

The licensee was also ordered to:

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

Emergency Procedure EP-108 ("Loss of Steam Generator Feed") provides adequate direction to the operators for providing alternate sources of water to the suction of the EFW pumps. The primary source of water to the EFW pumps is the condensate storage tank (CST), which has a capacity of 150,000 gallons. The operator is alerted by a level alarm on the CST when the level drops to 89,000 gallons. The condenser hotwell, which has a capacity of 200,000 gallons, is the alternate source of water to the EFW pumps. The procedure directs the operators to open the motor-operated valves that will connect the hotwell to the suction of the EFW pumps and then to close the motor-operated suction valves from the CST. The procedure also contains instructions on valve operation to provide a third source of demineralized water from the water treatment system if needed.

The NRC staff, at the site, has verified that the control room operators are properly trained to carry out these procedures. The NRC staff concludes that the licensee has complied with the requirements to review and revise procedures and operator training for providing alternate sources of water for the EFW system, and, is thus in compliance with this portion of the Order.

The licensee was ordered to:

6. "Remove the interlock which prevents the turbine-driven emergency feedwater pump operation when the motor-driven emergency feedwater pump is running."

The licensee has removed the interlock. The turbine-driven EFW pump will start, if required, regardless of the motor-driven EFW pump status.

Based on the above design modification, we conclude that the licensee has complied with this portion of the Order.

It was also ordered that:

7. "In the event emergency feedwater is necessary and offsite power is available, an auto start signal will be provided to the motor-driven emergency feedwater pump."

The licensee has installed circuitry to provide automatic starting of the motor-driven EFW pump if offsite power is available. The auto start signals include either of the following:

(1) coincident loss o. both main feedwater pumps, sensed by the loss of control oil pressure; or

(2) coincident low-low steam generator level in both steam generators, detected by existing and new equipment in the ICS.

Provisions have been included to manually bypass the loss of main feedwater pumps signal to allow for startup and/or shutdown. The bypass switch is keylocked, with annunciation and administrative control. The steam generator low-low level signal is not bypassed.

In addition, the licensee has modified the turbine-driven EFW pump start circuitry to include the same set of signals. Previously, this pump automatically started only on loss of both main feedwater pumps. Based on the above design modifications, we conclude that the licensee has complied with this portion of the Order.

It was also ordered that:

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

The licensee has provided control room annunciation to alert the operators that the EFW pumps (motor-driven and/or turbine-driven) have started when required or failed to start when required.

The conditions which initiate the above alarms include the same signals as discussed in Part 7 above:

(1) loss of main feedwater; or

(2) low-low level in both steam generators.

These signals are combined with the pump status (start or fail to start) to provide the annunciation. Based on the above modifications, we conclude that the licensee has complied with this portion of the Order.

It was also ordered:

9. "Verification has been made that the air-operated level control valves (a) fail to the 50% open position upon loss of power to the electrical/pressure converter, and (b) fail to the as is position upon loss of instrument air and electrical power to the air lock. At full load these valves are in the full (100%) open positions and at low power levels (below 15%) they are partially open controlling flow. If these valves were to fail closed, feedwater flow would be controlled using the EFW bypass valves as described in Item 3 above."

The licensee has completed its verification tests of the failure mode of the air-operated level control valves. The results show that one air-operated level control valve fails to a 54% open position and the other fails to a 47% open position upon loss of electrical power to the electrical/pressure converter. These failure positions are within acceptable tolerance of the 50% open position specified in the Order. On a test for loss of instrument air, both air-operated level control valves failed as is, i.e., remained at approximately the 50% open position during the test. The EFW bypass valves are motor-operated regulating valves which are operated independently from the ICS as discussed in part 3 above. If the air-operated level control valves remain closed or ICS fails, EFW flow would be manually

controlled using the EFW bypass valves. We conclude that the licensee has satisfied this portion of the Order.

Conclusion:

Based upon our evaluation of Parts 1 through 9 above, we conclude that the licensee has upgraded the timeliness and reliability of delivery from the EFW system by carrying out the actions identified in Enclosure 1 of the licensee's letter of May 1, 1979, and is, therefore, in compliance with Item (a) of the Order.

Item (b)

The licensee was ordered to:

> "Develop and implement operating procedures for initiating and controlling emergency feedwater independent of Integrated Control System Control."

The NRC staff has reviewed the revised procedures for the EFW system to assure that there is sufficient guidance for the operators to actuate the system if automatic initiation fails and to control steam generator levels at the required values. The The NRC staff review of the procedures and the operator training focused on whether the operators were directed to observe the proper instruments and whether operators were given specific values of parameters, such as steam generator level, to maintain by operating the control valves. The review also determined that the validity of the instrument readings of certain key parameters, such as steam generator level, would be confirmed. The modifications to the procedures to satisfy these determinations were verified by the NRC staff. (See further discussion of these procedures in part 3 of Item (a)).

We will require the licensee to perform a test during power ascension (less than 15% rated power) to demonstrate the capability to provide and control EFW flow to both steam generators. The primary objective is to verify that the operators can initiate EFW and control steam generator levels independent of the ICS. A member of the NRC staff will witness the test and verify acceptability prior to authorizing the licensee to proceed to full power operation.

The NRC staff audited a sample of Crystal River operators to determine if they were familiar with the revised procedures and could implement them correctly. Based on the NRC 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.

Item (c)

The Order required that the licensee:

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

The CR-3 original design did not have a direct reactor trip from a malfunction in the secondary system (loss of main feedwater and/or turbine trip). To obtain an earlier reactor trip (rather than delaying the trip until an operator took action or until a primary system parameter exceeded its trip setpoint), the licensee committed to install a hard-wired, control-grade reactor trip on the loss of all main feedwater and/or on turbine trip (letter from B. L. Griffin (FPC) to H. Denton (NRC) dated May 1, 1979). The purpose of this anticipatory trip is

to minimize the potential for opening of the PORV and/or the safety valves on the pressurizer.

The licensee has added control-grade circuitry which is designed to provide an automatic reactor trip wher either the main turbine trips or all main feedwater is lost.

The main turbine/generator trip is sensed by an existing pressure switch in the turbine electro-hydraulic control system. On a turbine trip, the pressure switch energizes a normally deenergized relay in the ICS. A contact from this relay is arranged in a normally energized circuit containing two parallel reactor trip actuation relays. Deenergizing both of these relays provides an output to energize the 125 DC volt shunt trip coils of the two reactor trip breakers. Energizing both reactor trip breakers trips the reactor.

The loss of main feedwater is sensed by either of two signals: loss of the main feedwater pumps or low-low level in both steam generators (the same signals which start EFW). The signals are generated separately for each feedwater path. Any one of these signals will energize a relay in the ICS (one relay for each feedwater path). The contacts from these relays are arranged in the same circuity as the reactor trip actuation relays such that any coincidence of signals from the two feedwater paths will deenergize the relays, in the same manner as the turbine trip, causing a reactor trip.

Provisions have been included to automatically bypass and reinstate the loss of main feedwater pump and turbine trip signals at less than 10% power to allow

for normal startup and shutdown of equipment without tripping the reactor. Operator verification of the bypass removal is required by procedure during power ascention.

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 volt supply and operates independently from the 120 volt AC undervoltage trip coil of the same AC reactor trip breaker, which receives a safety-grade reactor protection system trip signal.

An NRC inspector has confirmed that the checkout tests for the circuitry were completed successfully. In addition, the licensee has committed to perform a monthly test on the added circuitry in order to demonstrate its ability to open the AC reactor trip circuit breakers.

Based on our review of the implementation of the trip circuitry with respect to its independence from the existing 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 intended function.

Based on the above evaluation, we conclude that the licensee is in compliance with the requirements of Item (c) of the Order.

Item (d)

This item in the Order required the licensee to:

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

By letter dated May 1, 1979, the licensee committed to providing the analyses and operating procedures of this requirement.

B&W, the reactor vendor for the Crystal River 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 bound the Crystal River 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 July 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 and the high pressure injection (HPI) system combined with operator action ensure adequate core cooling. The EFW system, used to remove heat through the steam generators, has been modified to enhance its reliability as discussed in Item (a). The HPI system is capable of providing emergency core cooling up to 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 (for breaks

smaller than 0.01 sq. ft.) were performed assuming permanent loss of all feedwater (with operator initiation of the HPI system at 20 minutes), and loss of feedwater for only the first 20 minutes of the accident. Uncovering of the reactor core was 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 Crystal River considering the ability to manually start the redundant EFW pumps and HPI pumps from the control room, assuming failure of automatic EFW actuation.

Another aspect of the study was the assessment of recent design changes on the lift frequency of the pressurizer PORV and safety valves. The design changes included: (1) a change in the setpoint of the PORV from 2255 psig to 2450 psig; (2) change in the high pressure reactor trip setpoint from 2355 psig to 2300 psig; and (3) the installation of an anticipatory reactor trip on turbine trip and/or 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 HPI which could repressurize the system and lift the PORV valve. It is expected that the operator would terminate HPI before the PORV 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 cases of control rod withdrawal or inadvertent boron dilution transients, using the normally conservative assumptions found in the Final Safety Analysis Report Chapter 15 safety analyses. The above

*(The 50°F subcooling criteria is discussed on page 20 of this evaluation.)

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 (1) an isolated PORV, but safety valves opening and closing as designed, or (2) a stuck open PORV does not result in uncovering the reactor core, provided either EFW or HPI (2 pumps) is initiated within 20 minutes. Based on the consequences calculated for small break LOCAs and loss of all main feedwater events, and taking into account expected reliability of the EFW and HPI systems, we conclude that the licensee has complied with the analyses portion of Item (d) of the Order.

To support long-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 Sections 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 PORV and safety valve reliability during anticipated transients, as discussed in Section 8.4.6 of the NUREG 0560, the licensee will be required to provide analyses of the lift frequency and mechanical reliability of the pressurizer PORV and safety valves of the Crystal River facility.

The B&W analyses show that some operator actions, both immediate and follow-up, are required under certain circumstances for a small break accident. Immediate operator actions are 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. "Operating Guidelines for Small Breaks" were issued by B&W on May 5, 1979, and reviewed by the NRC staff. These guidelines were revised on May 15, 1979, to include revisions recommended by the staff (Reference 7). In response to these guidelines, the licensee made substantial revisions to EP-106 ("Loss of Reactor Coolant/RC System Pressure"), and EP-103 ("Loss of RC Flow/RC Pump Trip"). These emergency procedures define required operator actions in response to a spectrum of break sizes for a LOCA in conjunction with various equipment availability and failures.

EP-106 ("Loss of Reactor Coolant/RC System Pressure") is divided into three sections. The first section deals with a leak or rupture that is within the capability of one makeup pump.* In this case, the operators proceed with an orderly plant shutdown, if the leak is in excess of the Technical Specification limits.

The second section of EP-106 defines required operator actions for a small break that is within the capability of the HPI system to maintain RCS pressure and

*[At CR-3 the HPI pumps are used for makeup pumps.]

pressurizer level. This assumes that the initial break was of a size sufficient to cause a depressurization with a resulting reactor trip and HPI actuation. This part of the procedure provides the operators 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 HPI system through the break and the pressurizer PORV 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 EP-103 ("Loss of Reactor Flow/RCP Trip") which defines the actions necessary to establish natural circulation. Additional guidance is provided in EP-103 if natural circulation is not immediately achieved. This includes "bumping" reactor coolant pumps or if they cannot be operated, using the PORV to control RCS pressure until either forced flow or natural circulation can be achieved. If natural circulation has been established and plant conditions are stable, the operator is directed to AP-113 ("Reactor Cooldown by Natural Circulation"). If forced circulation is established, the normal plant cooldown procedure (OP-209, "Plant Cooldown") is used in conjunction with EP-106.

The third section of EP-106 deals with a large pipe rupture in which the system depressurizes to the point of low pressure injection (LPI). The system response is not dependent upon the availablity of reactor coolant pumps or feedwater and, therefore, no other procedures need be referenced.

For all cases in which HPI is manually or automatically initiated, the operators are specifically instructed to maintain maximum HPI flow unless one of the following criteria is 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, EP-105 ("Steam Supply System Rupture"), EP-108 ("Loss of Steam Generator Feed"), EP-101 ("Unit Blackout"), and EP-103 ("Loss of RC Flow/RC Pump Trip"). Each of these procedures, in addition to EP-106 ("Loss of Reactor Coolant/RC System Pressure") procedure, provide additional instructions to the operators in the event of faulty or misleading indications. A subsequent action statement directs the operators to check alternate instrumentation channels to confirm key parameter readings. The Crystal River staff has made revisions to all of their emergency procedures to include this confirmation. The CR-3 incore thermocouples will be hard-wired to a dedicated monitoring system which is programmed to alarm at high temperature. In addition, the operators will be able to check all input readings and/or get a printout of the status of each thermocouple with this system. A process computer in the control room is also available to provide this indication.

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 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 resistance temperature detectors (RTD) may 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 based on the expected recirculation through the vent valves and the vessel stress limits used. The licensee has incorporated these revised guidelines into Emergency Procedure EP-108 ("Loss of Steam Generator Feed"). Subsequent restoration of EFW would depressurize the reactor coolant system to below 600 psig 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 EFW 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 Crystal River.

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The Crystal River Unit 3 main control board has an annunciator which alarms when the PORV solenoid is energized (to open the valve). In addition, there are 3 indicating lights which are actuated by 2 selector switches of the valve control circuitry. The green light is lit when the "AUTO-OPEN" selector switch is in the "AUTO" position. In this position, the pressure signal will provide the open and close control of this valve. The red light is lit when the same switch is in the "OPEN" position. In this position, the selector switch will control the valve (to the open position). The amber light is lit when the "NORM-LO" selector switch is in the "LO" position. In this position, the low pressure protection circuit is operable and can open the valve for this mode of operation.

EP-106 ("Loss of Reactor Coolant/RC System Pressure") 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 Crystal River control room. The procedure was judged to provide adequate guidance to the operators to cope with a small break LOCA. The instrumentation necessary to diagnose the break, the indications and controls required by the action statements, and the administrative controls which prevent acceptable 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 8 of the 28 licensed operators assigned to shift duty 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 Crystal River staff has conducted special training sessions for the operators on the concept and use of EP-106 and other emergency procedures related to the small break accident. The audit revealed several deficiencies in the knowledge of the small break phenomenon and the requirements of the procedure. Additionally, there were deficiencies in the knowledge of the details of the recent design modifications made to the Crystal River plant. These deficiencies were primarily the result of design modifications and procedure revisions not finalized at that time. As a result of the audit, each licensed individual received additional training by the plant training organization and by the General Physics Corporation (GPC). This additional training has been completed and verified by the NRC staff. A subsequent reaudit of 10 licensed individuals by the NRC revealed satisfactory results.

The audit of the operators also included questioning about the TMI-2 incident and the resulting impact on the Crystal River plant. 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 GPC and has been verified by the NRC staff.

In summary, we found their level of understanding sufficient to be able to respond to a similar situation if it happened at Crystal River. We also conclude they

have adequate knowledge of subcooling and saturated conditions and are able to recognize each in the primary coolant system by several methods. The EFW 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 EFW 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 were audited by the NRC staff and judged to be acceptable. The staff will also review all subsequent results and records as part of the normal inspection function of the Crystal River re-qualification program. We conclude that there is adequate assurance that the operators at Crystal River have and will continue to receive a high level of training concerning the TMI-2 accident and the consequent impact on their unit.

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 required that:

> "All licensed reactor operators and senior reactor operators will have completed the TMI-2 simulator training at B&W."

The licensee has confirmed that all reactor operators and senior reactor operators have completed the TMI-2 simulator training at B&W as required by the Order. This training consisted of a class discussion of the TMI-2 event and a demonstration of the event on the simulator as it occurred and how it should have been controlled. The class discussion was about one hour long and the remainder of the four hour session was conducted on the simulator. The TMI-2 event, including operational errors, was demonstrated to each operator. The event was again initiated and the operators were given "hands-on" experience in successfully regaining control of the plant by several methods. Other transients which resulted in depressurization and saturation conditions were presented to the operators in which they maneuvered the plant to a stable, subcooled condition.

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

CONCLUSION

We conclude that the actions described above fulfill the requirements of our Order of May 16, 1979, in regard to Paragraph (1) of Section IV. The licensee having met the requirements of Paragraph (1) may restart Crystal River as provided by Paragraph (2). Paragraph (3) of Section IV of the Order remains in

force until the long-term actions set forth in Section II of the Order are completed and approved by the NRC.

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 1 above, dated May 10, 1979.

4. Letter from J. H. Taylor (B&W) to T. M. Novak (NRC) providing background information on reactor coolant pump operation, dated May 10, 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 1, dated May 12, 1979.

6. Letter from J. H. Taylor (B&W) to R. J. Mattson (NRC), providing Supplement 3 to Section 6 of report in Item 1, dated May 24, 1979.

7. Letter from J. H. Taylor (B&W) to Z. R. Rosztoczy (NRC) transmitting revised "Operating Guidelines for Small Breaks," dated May 16, 1979.