UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

EVALUATION OF LICENSEE'S COMPLIANCE WITH THE NRC ORDER DATED MAY 7, 1979

DUKE POWER COMPANY

OCONEE NUCLEAR STATION, UNITS NOS. 1, 2 AND 3

DOCKETS NOS. 50-269, 270 AND 287

INTRODUCTION

By order dated May 7, 1979 (the Order), the Duke Power Company (DPC or the licensee) was ordered by the NRC to take certain actions with respect to Oconee Nuclear Station, Units 1, 2 and 3. Prior to this Order and as a result of a preliminary review of the Three Mile Island Unit No. 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 Babcock & Wilcox (B&W) designed reactors to take further actions, including immediate changes to decrease the reactor high pressure trip point and increase the pressurizer pilot-operated relief valve 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 on April 25, 1979. After a series of discussions between the NRC staff and the

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licensee concerning possible design modifications and changes in operating procedures, the licensee agreed in letters dated April 25, 26, and May 4, 1979 to perform promptly certain actions. The Commission found that operation of all units should not be resumed or continued on an indefinite basis 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 3, 8, 10, and 16, 1979 and numerous discussions with the licensee's staff. We have utilized confirmation of design and procedure changes by the NRC resident inspector at the Oconee site and an audit by the NRC staff of the training of the Oconee reactor operators to assure that the design and procedure changes are understood by the operators and that the revised procedures are being correctly implemented by the operators.

EVALUATION

Item (a)

The original Oconee design had a single emergency feedwater (EFW) pump for each unit that was actuated automatically when the main feedwater was lost on that unit. There were provisions for manually interconnecting the discharge of the EFW pumps so that they could service all three units. In letters from W. Parker (DPC) to H. Denton (NRC) dated April 25, 1979 and W. Lee (DPC) to H. Denton (NRC) dated April 26, 1979, the Duke Power Company committed to

installing an automatic starting feature of the interconnected emergency feedwater system so that all three EFW pumps will receive a start signal from any affected unit, and to performing a stability test of the system when more than one unit was using EFW. A trained operator stationed at each EFW pump to actuate it locally, if required, and a trained operator in the control room to maintain required steam generator water levels were also included in the licensee's commitments.

In particular, it was ordered that the licensee shall take the following actions with respect to Oconee 1, 2 and 3:

> "Install automatic starting of the interconnected emergency feedwater system so that all three pumps will receive a start signal from any affected unit..." "The emergency feedwater pump discharge flow will be connected to the interconnection headers such that each or all of the EFW pumps can supply water to any unit." "The discharges of these pumps have been tied together by alignment of manual valves such that each and all of the pumps can supply emergency feedwater to any Oconee Unit requiring it."

The licensee has aligned his EFW system so that all three EFW pumps feed a common header and there are separate lines from this common header which deliver EFW through a control valve to each of the two steam generators in the three units. This alignment, which has been verified by an NRC inspector at the site, was accomplished by isolating other alternate flow paths in the feedwater system with existing manual valves.

We have reviewed the piping and instrumentation drawings and have determined that no active failure to a mechanical component such as a pump or valve would preclude obtaining EFW flow in any unit. The licensee has performed flow tests with two EFW pumps operating and providing flow to the four steam generators associated with two units. The minimum flow under this condition was greater than 720 gpm per unit which is acceptable. We have reviewed the modification to determine the minimum flow going to a unit if only one steam generator was functioning. The test to determine the maximum flow to one steam generator was not performed, but the licensee stated that the results of his analysis show that a minimum of 720 gpm EFW flow will be delivered to any unit with only one steam generator functioning. The licensee also stated that the flow characteristics of the EFW system used in the analysis were confirmed in the flow tests. In one of these flow tests with only one steam generator functioning in one of the units, the flow to that steam generator with about a 60% valve opening was approximately 500 gpm. We expect higher flows would be obtained by operater action. This 500 gpm flow rate with one steam generator available is acceptable based on automatic initiation of EFW.

Based on the above evaluation, the NRC staff concludes that the licensee has complied with the requirement for an interconnected EFW system such that each or all of the EFW pumps can supply water to any unit.

The licensee has added new circuitry which is designed to automatically start all three of the existing turbine-driven EFW pumps on loss of all main feedwater to any of the three units and to open the EFW regulating valves in the affected unit(s). The loss of main feedwater supply to each unit is sensed by a normally energized auxiliary relay. The relay is deenergized by either low pressure in the main feedwater discharge header or by both main feedwater pumps tripped in the affected unit. The relay actuates each of the three EFW pumps and the solenoids which control the EFW regulating valves to the two steam generators in the affected unit. Loss of the 115 VAC power will therefore actuate the three turbine-driven EFW pumps and automatically open the EFW regulating valves to the preselected setting and deliver EFW to the steam generators. This actuation is independent of the Integrated Control System (ICS).

The licensee has performed tests to demonstrate that these design modifications will automatically start all three EFW pumps and open the EFW regulating valves in the affected unit on loss of all main feedwater to any of the three units. The NRC resident inspector has verified the test results and we have concluded that the tests are satisfactory.

Based on the above, the NRC staff concludes that the licensee has complied with the requirement for automatic starting of the interconnected EFW system so that all three pumps will receive a start signal from any affected unit.

It was also ordered that the licensee "....test the system for stability." The licensee performed flow stability tests with two units operating and two EFW pumps available. The EFW flow to one of the steam generators was stopped

by closing a manual valve in the discharge line and the reactor operators inside the control room manually adjusted the flow to the other three steam generators to prescribed flow rates. The manual control of the emergency feedwater flow did not result in any flow instability or operator problems. We conclude that these tests are satisfactory and that the licensee has complied with the requirement to perform a stability test. However, we require the licensee to perform additional startup tests on the EFW system when all three units are operational to demonstrate acceptable flow rates and flow stability with only two operating EFW pumps. The test plan must be reviewed by the NRC staff prior to the tests, which will be witnessed by an NRC inspector.

It was also ordered that:

> "Administrative controls have been established so that in the event of loss of both main feedwater pumps on an affected unit, that unit's EFW pump will start automatically, backed up by remote manual start from the control room. If the pump fails to start automatically, the operator stationed at that pump will start the pump locally, and has been trained to do so.
>
> In addition, the other two available EFW pumps will be started remotely from their unit's control room or locally if required to provide two more sources of feedwater to the affected unit."

The licensee has stationed an operator at each of the EFW pumps to start the pumps locally if required. The procedures associated with the EFW system (discussed in item (b)) have provisions for verifying that all EFW pumps start automatically with local initiation if required. The operators stationed locally at the pumps have been trained to start the pumps and have procedures for accomplishing this task. The implementation of this requirement has been verified by an NRC inspector. The actuation of the EFW pump locally is fairly simple. Depending on the fault, it may require bleeding off air to open the steam admission valve to the turbine (these are turbine driven pumps) before opening the governor valves with a local controller. Members of the NRC staff at the site have estimated it would take less than 5 minutes to actuate the pumps locally even allowing for several false starts. The operator stationed locally is also provided with piping and instrumentation diagrams (P&IDs) of the EFW pumps, descriptions of the governor valves, and auxiliaries such as the lube oil and cooling water. If necessary, the local operator can take remedial actions if subsequent problems with pump operation arise.

Based on our inspection at the site, we conclude that the licensee has complied with the requirement to have a trained operator stationed at each EFW pump and has administrative procedures covering his actions if required.

It was also ordered that:

> "Emergency feedwater flow to the steam generators will be assured by the control room operator who has been trained to maintain the necessary level."

The licensee has modified his procedures on the EFW system as discussed in item (b) of this evaluation. These procedures include verification of flow and manual control of steam generator water level. The regulating valves in the individual EFW discharge lines open automatically upon signal of need from the affected unit(s); it is only later that the valves are controlled by the operator. The licensee has installed flow orifices in each of the EFW discharge lines with indicators in the control room.

The NRC staff at the site has verified that control room operators is properly trained to carry out these procedures. We conclude that the licensee has complied with the requirement that a trained control room operator shall maintain the necessary steam generator water level.

Based on the above, we conclude that the licensee has complied with the requirements of item (a) of paragraph (1) of the NRC Order, with the provision for additional flow tests when all three units are operational.

Item (b)

By letter from W. Lee (DPC) to H. Denton (NRC) dated April 26, 1979, the licensee committed to develop and implement operating procedures for initiating and controlling EFW independent of ICS control. With the installation of the modified EFW system, discussed in item (a), the licensee has bypassed the previous piping and valve alignments that were controlled by the ICS. As a result, the present EFW system is totally separate from the ICS. The order requires the licensee to:

"Develop and implement operating procedures for initiating and controlling EFW independent of Integrated Control System control."

The licensee has revised his emergency procedures related to the use of the EFW system to reflect the separation of the EFW system from the ICS. The key operator actions resulting from the system modification are to verify that all EFW pumps are actuated either automatically or manually and to maintain the steam generator water level at prescribed values which depend on whether the reactor coolant pumps are operating. These procedures would be implemented by the operator whenever there is a loss of all main feedwater caused by feedwater system problems or loss of offsite power. For all other events, the steam generator water level would be controlled by the ICS using the startup valves and the main feedwater pumps.

We have reviewed the revised procedures for the EFW system to assure that there is sufficient guidance to the operator to activate the system if the automatic initiation fails and to manually control the steam generator water level to specified values.

The review of the procedures included consideration of verifying readings of certain key parameters by using alternate instrumentation and specification of parameter values that must be controlled by the operator. Our comments on the procedures were incorporated by the licensee and verified by the NRC resident inspector. The licensee has committed to provide double verification of the restoration of equipment following surveillance tests or maintenance on the EFW system.

NRC staff at the Oconee site walked through the EFW procedures with Oconee operators to evaluate whether the procedures were functionally adequate. In addition, the NRC staff audited a sample of Oconee operators to determine if they were familiar with the revised procedures and would implement them correctly. Based on the NRC staff audit, we conclude that the revised procedures and operator training are satisfactory.

The procedures reviewed addressed the following emergency conditions:

1. Loss of Main Feedwater Pumps
2. Loss of Main Feedwater Pumps and Emergency Feedwater
3. Loss of Station Power and/or Loss of Instrument Air
4. Loss of Reactor Coolant Flow both with and without Station power and instrument air
5. Steam line break inside the reactor building both with and without Station power and instrument air.
6. Steam line break outside the reactor building both with and without Station power and instrument air.

Based on our review and verification, we find that the licensee has complied with the requirements of item (b) of paragraph (1) of the Order.

Item (c)

The original Oconee design did not have any direct reactor trips that would be initiated by a malfunction in the secondary system (feedwater and steam). To obtain an earlier reactor trip (rather than delaying the trip until an operator

took action or when 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. (Letter from W. Lee (DPC) to H. Denton (NRC) dated April 26, 1979.) The Order requires that:

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

The purpose of this anticipatory trip is to minimize the potential for opening the Pilot Operated Relief Valves (PORVs) and/or the safety valves on the pressurizer. The licensee has estimated that the new anticipatory trip will result in a reactor trip 3 to 10 seconds earlier for loss of all main feedwater and turbine trip events, and the staff generally agrees.

The licensee has added new nonsafety-grade circuitry to Units 1, 2, and 3 which is designed to provide an automatic reactor trip when either the main turbine trips or all the main feedwater is lost.

The main turbine trip is sensed by an existing normally deenergized auxiliary relay in the main turbine Electro-Hydraulic control system. The relay is energized by a Class IE dc power supply upon any turbine trip signal on the main turbine trip bus. The relay provides two contact closures to energize two dc shunt coils (one in each of the two reactor trip ac circuit breakers) to open each of the breakers and trip the reactor. The shunt coil power supply is also from the Class IE dc source. Both ac circuit breakers must be opened to cause reactor trip. Provisions have been included to bypass the

turbine trip signal to the breaker (via a main control room switch) for startup and/or low power levels.

The total main feedwater loss is sensed by a normally energized auxiliary relay in the EFW actuation circuitry. The relay is deenergized by either a low pressure signal in the main feedwater discharge header or indication that both main feedwater pumps are tripped (indication is provided by feedwater pump turbine steam stop valve limit switches.) The relay provides a contact closure to energize each of the same two dc shunt coils energized in the event of a turbine trip to open the ac circuit breaker and trip the reactor. A loss of the 115 VAC Class IE power to this relay will cause a reactor trip and initiate EFW.

The turbine trip and main feedwater trip circuits are wired in parallel to each coil such that either signal will cause the ac circuit breakers to open.

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. This shunt coil is powered by a Class IE 125 VDC supply. It is not classified as part of the existing reactor trip system. However, it is separate and operates independently from the 120 VAC undervoltage trip coil of the same ac breaker. The reactor trip safety-grade signal deenergizes the 120 VAC undervoltage coil to produce a trip of the same ac breaker. The new cabling associated with the added circuitry is located in the cable room/control room area and is routed with the non-Class IE cabling throughout.

The licensee has committed to perform a monthly test on the added circuitry in order to demonstrate its ability to open the ac 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 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. The resident IE inspector has confirmed that the checkout tests for this circuitry were completed successfully.

On the basis of the above, we conclude that the new trip complies with the requirements of item (c) of paragraph (1) of the Order.

Item (d)

This item in the Order requires the licensee to:

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

By letter from W. Lee (DPC) to H. Denton (NRC) dated April 26, 1979, the licensee committed to providing the analyses and operating procedures of this requirement.

Babcock & Wilcox, the reactor vendor for the Oconee plants, submitted an analysis entitled, "Evaluation of Transient Behavior and Small Reactor Coolant System Breaks in the 177 Fuel Assembly Plant" and supplements to this analysis (References 1 through 5). The major parameters used in this generic study conservatively bound the Oconee plants. EG&G on NRC's request also performed two reactor coolant system small break calculations for the Oconee plant. The EG&G calculations were consistent with B&W results. The staff evaluation of B&W generic study has been completed and the results of the evaluation will be issued as a NUREG report prior to June 1, 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, high pressure injection system and operator action ensure adequate core cooling. The auxiliary feedwater 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 set point. 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. The ability to remove heat via the steam generators has always been recognized to be an important consideration when analyzing small breaks. Sensitivity analyses were performed with acceptable results 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. These are acceptable results

considering the ability to locally start the EFW pumps in five minutes as discussed under item (a) of this evaluation, assuming failure of automatic EFW actuation.

Another aspect of the studies was the assessment of recent design changes on the lift frequency of pressurizer safety and relief valves. The design changes included change in the setpoint of the pressurizer relief valve from 2255 psi to 2450 psi, change in the high pressure reactor trip setpoint from 2355 psi to 2300 psi and the installation of anticipatory reactor trips on turbine trip and on loss of feedwater. In the past, during turbine trip and loss of feedwater transients the pressurizer relief valves were lifted. With the new design these transients do not result in lifting of the relief valve. However, lifting of both relief and safety valves might occur in case of rod withdrawal and boron dilution transients, using the normally conservative assumptions found in the Chapter 15 safety analyses. The above design changes did not effect the lift frequency of the safety valves for these events.

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, or (b) a stuck open PORV consequentially does not result in core uncovery, provided either EFW or 2 HPI pumps are initiated within 20 minutes. Based on the acceptable consequences calculated for small break LOCAs and loss of all main feedwater events and the expected reliability of the EFW and high pressure injection systems, we conclude that the licensee has complied with the analysis portion of paragraph (1)(d) of the Order.

To support longer term operation of these facilities, requirements will be developed for additional and more detailed analyses of loss of feedwater and other anticipated operational transients. More detailed analysis 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 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 mechanical reliability of the pressurizer relief and safety valves of the Oconee facilities.

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 are necessary to take as soon as the problem is diagnosed. In order to carry out followup actions the operators must consult and follow instructions 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 in order to assist the utility staff of the operating B&W facilities to develop emergency procedures for the small break accident.

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 DPC staff at the Oconee Nuclear Station made substantial revisions to EP/O/A/1800/4, Loss of Reactor Coolant. This emergency procedure defines the required operator action in response to a spectrum of break sizes for a LOCA in conjunction with various equipment availability and failures. The procedure is divided into eight sections beginning with excessive reactor coolant system leakage without a reactor trip and concluding with a rupture in excess of the capability of three high pressure injection pumps. The latter case is the larger break accident in which the system depressurizes to the point of low pressure injection.

Six cases of small break accidents are considered in the procedure. The first one assumes that feedwater to the steam generators and the reactor coolant pumps is available but the reactor is not automatically tripped. The second case increases the break size to cause an automatic trip of the reactor. In both cases, the required operator actions are generally the same and a safe, cold shutdown of the plant is accomplished with normal cooldown procedures.

The other four small break procedures provide guidance to the operators for dealing with degraded conditions such as loss of feedwater and/or loss of reactor coolant pumps. If feedwater is lost, a heat removal path is established from the high pressure injection system through the break and 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 operators are directed to establish and verify natural circulation. Additional guidance is provided if natural circulation is not immediately achieved.

For all cases in which HPI is manually or automatically initiated, the operators are specifically instructed to maintain maximum high pressure injection flow unless two criteria are met. These criteria are:

1. Both LPI pumps are in operation and flowing 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 temperatures for the existing reactor coolant system pressure. If the 50 degrees subcooling cannot be maintained after high pressure injection cutoff, the HPI shall be reactuated.

The 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, Steam Generator Tube Rupture, Loss of Reactor Coolant Flow and Loss of Steam Generator Feedwater. Each of these procedures, in addition to the Loss of Reactor Coolant 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 the key parameter readings.

The Loss of Reactor Coolant 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. Two members of the NRC staff "walked through" the latest revision to this emergency procedure in the Oconee control rooms. 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 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 fourteen licensed operators and senior operators was conducted by the NRC staff to determine their understanding of the small break accident including how the operators are required to diagnose and respond to it. The Oconee staff has conducted special training sessions for the operators on the concept of and use of the emergency procedures, EP 1800/4. We found the operators had sufficient knowledge of the small break phenomenon and the general requirements of the emergency procedure. Each licensed individual has since received additional training on the approved procedure prior to assuming his shift duties.

The audit of the operators also included questioning about the TMI-2 incident and the resulting design changes made at Oconee. Our discussions with them 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 errors that were apparently made during the course of the incident. We found their level of understanding sufficient to be able to respond to a similar situation if it happened at Oconee. We can also conclude they have adequate knowledge of thermodynamic processes of subcooling and saturated conditions and are able to recognize each in the primary coolant system.

Based on the foregoing evaluation, we conclude that the licensee has complied with the requirements of item (d) of paragraph (1) of the Order.

Item (e)

By letter from W. Parker (DPC) to J. O'Reilly (NRC) dated May 4, 1979, the licensee has confirmed that all reactor operators and senior reactor operators assigned to the Oconee control rooms have completed the TMI-2 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 to six 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.

The licensed operators and senior operators have received in excess of 23 hours of training concerning the TMI-2 accident and followup actions. Nearly 30 percent of the licensed individuals on shift duty at the 3 units were interviewed by the NRC. The results were judged to be satisfactory with some generic deficiencies noted to their management. In order to correct these deficiencies, Duke Power Company has committed (letter from W. O. Parker (DPC) to H. Denton (NRC) dated May 16, 1979) to a written examination by their training services group. An individual must receive a grade of 90% before he will be utilized at the control board of an operating unit. For long term verification of the effectiveness of the training, Duke Power Company has contracted with B&W and General Physics Corporation to independently perform audits on the operators. The NRC staff will review all results and recommendations as part of our normal inspection function of their requalification program. We conclude that there is adequate assurance that the operators at Oconee 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 our interviews with a sample of the licensed operators and Duke Power Company's commitment to examine all of their operators, we conclude that the licensee has complied with the requirements of item (e) paragraph (1) 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) need not shut down Unit 1 as

described in Paragraph (4) and may restart Unit 2 and Unit 3 as provided by Paragraphs (2) and (3). Paragraph (2) of Section IV of the Order is related specifically to Oconee Unit No. 3, which is currently shutdown for a reload. Unit 3 is undergoing a reload review and cannot restart until NRC issues a license amendment related to the reload review in addition to meeting the requirements of Paragraph (2) of Section IV of the Order. Paragraph (5) 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.

Dated: May 18, 1979

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 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 T. M. Novak, 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 (PROV) 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.