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REMARKS

It has been noted that pages 1-12 were incomplete.

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

July 2, 1979

Docket No.: 50-346

MEMORANDUM FOR: Chairman Hendrie
Commissioner Gilinsky
Commissioner Kennedy
Commissioner Bradford
Commissioner Ahearne

FROM: Harold R. Denton, Director
Office of Nuclear Reactor Regulation

THRU: Executive Director for Operations TAR for L.V.G.

SUBJECT: DAVIS-BESSE 1

Enclosed is a revised Safety Evaluation for Davis-Besse 1, relative to the shutdown order of 16 May 1979. We have provided supplemental material at page 6, and following, to account for our evaluation of an LER, in the auxiliary feedwater system area, which we recently received. Also enclosed for your information are two letters on this subject; one is from NRR to Toledo Edison, dated June 29, 1979; the second letter is Toledo Edison's response.

H R Denton

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Enclosures:

1. Revised SER
2. Ltr dtd 6-29-79 Denton to Toledo Edison
3. Ltr dtd 6-29-79 Toledo Edison to Denton

cc w/enclosures: See next page

8000081

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EVALUATION OF LICENSEE'S COMPLIANCE

WITH THE NRC ORDER DATED MAY 16, 1979

TOLEDO EDISON COMPANY AND

THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

DAVIS-BESSE NUCLEAR POWER STATION, UNIT No. 1

DOCKET NO. 50-346

INTRODUCTION

By Order dated May 16, 1979, (the Order) the Toledo Edison Company and the Cleveland Electric Illuminating Company (TECO or the licensee) were directed by the NRC to take certain actions with respect to Davis-Besse Nuclear Power Station, Unit 1 (DB-1). 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 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 power-operated relief valve (PORV) setting.*

*[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.]

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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 letters dated April 27, 1979 and May 4, 1979, to perform promptly certain actions. The Commission found that operation of the plant should not be resumed until the actions described in Items (a) through (g) of paragraph (1) of Section IV of the Order are satisfactorily completed.

Our evaluation of the licensee's compliance with items (a) through (g) 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 in letters dated May 11, 18, 19, 22 (2), 23 (2), 26 (2), 29 and June 15 (2), 18, 21, 23 and 25, 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 DB-1 site. An audit of the training and performance of the DB-1 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:

> "Review all aspects of the safety grade auxiliary feedwater system to further upgrade components for added reliability and performance. Present modifications will include the addition of dynamic braking on the auxiliary feedpump turbine speed changer and provision of means for control room verification of the auxiliary feedwater flow to the steam generators. This means of verification will be provided for one steam generator prior to startup from the present maintenance outage and for the other steam generator as soon as vendor-supplied equipment is available (estimated date is June 1, 1979). In addition, the licensees will review and verify the adequacy of the auxiliary feedwater system capacity."

The auxiliary feedwater (AFW) system at DB-1 consists of two safety-grade AFW pumps capable of being actuated and controlled by safety-grade signals that ensure the availability of feedwater to at least one steam generator, under the assumed conditions of a single failure. In addition, the capability to manually actuate and control AFW is available in the control room. The sources of water include two condensate storage tanks (CST), the service water system and the fire protection system. The CSTs provide the normal supply (non-safety-grade) and the service water system is used as a backup safety-grade supply.

A low level in either CST is alarmed to the operator and a continuous level is displayed inside the control room. Low pressure switches on the AFW pump suction provide safety-grade signals to automatically shift suction for the pump from the CSTs to the backup service water supply. Additionally, the operator could also manually transfer the AFW suction to the fire water storage tank (FWST) in the fire protection system.

Both steam-driven auxiliary feedwater pump turbines at DB-1 are provided with a governor used for variable pump speed control. The governor is equipped with a small DC motor which changes the speed setpoint on the turbine control valve, thereby controlling steam flow which regulates the turbine and pump speed. This DC motor receives "raise-and-lower" pulses from the safety-grade steam generator level control system or the manual control switches (located in the control room), which change the turbine speed as required. Pulse length is automatically increased the further steam generator level deviates from its setpoint. These changes in pump speed alter the AFW flow and thus control the water level in the steam generators.

A "dynamic brake" feature has been added, which consists of a resistor and electrical contacts in parallel with the windings of the DC motor. When the control pulse is terminated, the braking resistor is placed in parallel with the motor windings, causing rapid dissipation of the energy associated with the motor momentum (thus reducing the amount of motor coast). This, in turn, reduces the amount of pump speed overshoot, thereby allowing fewer speed changes to match the AFW flow rate to the steaming rate of the steam generators.

The licensee has also added flow rate indication for both steam generator AFW inlet lines. Each inlet line has a pipe-mounted ultrasonic flow transducer and signal conditioner. These are located in the auxiliary building and are accessible during normal plant operations. The signal conditioners provide outputs both locally and in the control room on the AFW pump section of the main control console. Each device is designed to provide flow rate indication to each steam generator from 0 to 1000 gpm. The systems are powered from 120 VAC, 60 Hz buses which are fed by redundant non-Class IE station inverters. Functional testing of the installed auxiliary feedwater flow rate indication is to be conducted in conjunction with the functional testing of the dynamic braking modification of AFW pump turbine controls. The staff concludes that the dynamic brake and AFW flow rate indication modifications are acceptable contingent upon successful testing prior to restart.

We have reviewed the piping and instrumentation diagrams and have determined that no active failure of a mechanical component, such as a pump or valve, would preclude obtaining the required AFW flow rate. The licensee has previously performed tests of the manual and automatic level control system. The test results showed that the control system functioned as designed to control steam generator level. Verification of acceptable flow capacity for each of the two AFW pumps was based upon recorded steam generator level changes following a previous reactor trip. These data showed that each pump exceeded the design flow rate of 800 gpm at a steam generator pressure of 1050 psig. (The 800 gpm is the flow rate delivered to the steam generators and does not include the approximately 250 gpm recirculation flow rate.)

Additional information submitted by the licensee (letter from Lowell E. Roe (TECO) to Mr. Robert W. Reid (NRC) dated May 23, 1979) shows that a total minimum flow, to one or both steam generators, of 550 gpm is required to support the accident analyses. Based on these data and analyses, and the agreement by the licensee to perform checkout testing of the dynamic braking and flow rate indication modifications prior to restart, we conclude that adequate assurance exists that the AFW system will deliver the required flow rate upon demand.

By letter (Lowell E. Roe (TECO) to Mr. Robert W. Reid (NRC) dated May 23, 1979), the licensee provided results of a review of the operating history of the AFW system at DB-1. The largest number of failures* occurred during the initial operating and debugging phase of the facility. Fourteen (14) of the seventeen (17) reported failures occurred prior to January, 1978. Subsequent to implementing system design changes as a result of several of these failures, the systems failure rate has been reduced and its reliability enhanced. There have been 3 failures of AFW system components from January 1978 to date. (There were a total of 65 actuations of the AFW system in this time period.) Three different components in the AFW system were involved in these three failures: (1) the speed control circuit for #1 AFW pump turbine, (2) a faulty limit switch on an AFW discharge valve, and (3) two sticky AFW pump turbine steam supply valves. In each case, the licensee performed corrective actions and failures in these components have not reoccurred. A more recent letter

*[For the purpose of demonstrating improvement in the performance of the AFW system, the licensee has defined a failure of the AFW system to be any event for which at least one train of the AFW system is not delivering design flow to a steam generator.]

(Lowell E. Roe (TECO) to Mr. Robert W. Reid (NRC) dated June 29, 1979) addressed a series of pressure switch failures which were discovered on May 21, 1979, and which affected both AFW trains. An evaluation of these failures by the licensee concluded that both trains would have automatically actuated if required, but that one train would not have shifted automatically to the service water supply. The NRC staff has discussed these failures with TECO and has requested that an improved surveillance program for these pressure switches be initiated to determine the cause of the failures and the optimum calibration interval. The licensee has agreed to increased frequency of switch calibration. In addition, the licensee has made procedural changes, requested by the staff, to instruct the operator to manually shift to the alternate supply of water for the AFW pumps, when the CST level drops to three feet (if automatic switchover has not occurred). This procedure provides greater assurance that, even with failures of this nature, the AFW system is available during the longer term. The staff concludes, that the licensee has increased the reliability of the AFW system by implementing appropriate corrective actions and design modifications.

In addition, the licensee has revised the administrative procedure pertaining to valve alignment and control. These revisions to AD 1839.02 ("Operation and Control of Locked Valves") provide further assurance that mispositioning of AFW system valves would be detected.

Based on the above evaluation, the NRC staff concludes that the licensee has complied with the requirement of Item (a) of the Order.

Item (b)

It was also ordered that the licensee:

"Revise operating procedures as necessary to eliminate the option of using the Integrated Control System as a backup means for controlling auxiliary feedwater flow."

As indicated in Item (a), the DB-1 AFW system has been designed as a safety grade system and, as such, is separate from the integrated control system (ICS); however, the licensee has indicated that the AFW system is capable of being switched to the ICS mode for a backup means of control. As currently designed, the AFW system has three operational modes of controlling flow: "ICS control", "auto-essential" and "manual." We requested that the licensee consider a more positive means to assure the continued separability of the ICS control position of the mode selector switches. The licensee agreed (letter from Lowell E. Roe (TECO) to Mr. Robert W. Reid (NRC) dated June 15, 1979) to install a mechanical stop on these switches to further deter use of the ICS control position. The IE site inspector has verified the installation of this mechanical stop.

The licensee has revised SP 1106.06 ("Auxiliary Feedwater System"), which describes procedures for AFW system operation. This procedure specifically prohibits the use of the ICS control position on the mode selector switches. Procedural steps for placing the AFW system in service for plant startup require the operator to place the AFW mode selector switches in the auto-essential position. We have reviewed the revised procedure for AFW switch operation and conclude there is sufficient guidance to prevent use of the AFW system in the ICS mode of control.

Other plant procedures that made reference to the ICS control mode of AFW have been revised by the licensee to no longer authorize that mode of control. The staff has reviewed those procedures and concludes that those revisions are adequate. In addition, the NRC staff audit confirmed that the control room operators are aware that ICS control of AFW is prohibited.

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

Item(c)

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 DB-1 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 Lowell E. Roe (TECO) to H. Denton (NRC) dated April 27, 1979). The purpose of this anticipatory trip is to minimize the potential for opening of the power-operated relief valve (PORV) and/or the safety valves on the pressurizer. 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).

TECO has added control-grade circuitry to DB-1 which is designed to provide an automatic reactor trip when either the main turbine trips or there is a reverse differential pressure of 177 psid across both of the two main feedwater check valves (one check valve is located in the main feedwater discharge piping associated with each steam generator). The main turbine trip is sensed by a normally deenergized auxiliary relay associated with the main turbine generator master trip bus. The power for this bus is provided from a 24 volt DC source, which in turn is provided power (through rectifier circuitry) from a non-Class 1E inverter supplied 120 volt AC distribution panel. A contact from the above auxiliary relay is arranged into a 120 volt AC circuit containing four normally deenergized relays. Power for this 120 volt circuit is provided from a Class 1E inverter supplied distribution panel. The design for these four relays and appropriate associated circuitry conform to Class 1E requirements, including physical independence and provisions for testing. Each of these four relays provide one contact which is arranged in series with one of the four Class 1E undervoltage coils associated with one of the four AC reactor trip circuit breakers (one undervoltage coil associated with each AC reactor trip circuit breaker). When these relays are energized, power to the associated Class 1E undervoltage coils is interrupted so as to produce the desired reactor trip.

As indicated above, differential pressure switches across check valves, located in the main feedwater pump discharge piping, actuate upon sensing a reverse differential pressure across these check valves. Two contacts from these differential pressure switches are arranged into a 125 volt DC circuit, which is provided power from a Class 1E 125 volt distribution panel. This circuit contains two associated DC relays. Two contacts (one contact per relay) associated with these relays are arranged in series. This series contact arrangement is provided in parallel with the contact associated with the main turbine generator master trip bus. The remaining circuitry associated with this trip is identical and common (shared) to that described above for the turbine trip (including power supply identification).

Provisions have been included in the design to manually bypass and to reinstate the reactor trip feature associated with the main turbine generator trip. To supplement this feature, the design includes an annunciator which actuates whenever this reactor trip is bypassed and the reactor power level is above 15 percent. Access to this bypass switch will require a key which is under suitable administrative control. Operator verification of the bypass removal is required by procedure during power escalation. The NRC staff has reviewed these procedures and concludes that sufficient administrative control exists. No bypass features are included in the design for the reactor trip feature associated with the loss of main feedwater circuitry. During normal startup or shutdown, an electric auxiliary pump is used when the steam driven main feedwater pumps are not available.

The licensee has analyzed this additional circuitry with respect to its independence from the existing reactor trip system and to assure that the design and operation of this additional circuitry will neither degrade the reliability of the existing reactor protection system nor create any new adverse safety system interactions. Based on our review of the implementation of the added 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. In addition, the licensee has satisfactorily completed testing of this trip circuitry.

The licensee has committed to perform a monthly periodic test of the added circuitry to demonstrate its ability to open the AC reactor trip circuit breakers (tripping of the AC reactor trip circuit breakers via the under-voltage trip circuit). We conclude that there is reasonable assurance that the additional circuitry will perform its intended function.

Based on the above evaluation, 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: