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Docket No. 50-261

Carolina Power & Light Company
ATTN: Mr. J. A. Jones
Senior Vice President
336 Fayetteville Street
Raleigh, North Carolina 27602

Gentlemen:

SUBJECT: NRC STAFF EVALUATION OF CP&L RESPONSES TO IE BULLETINS 79-06A AND 79-06A, REVISION 1, FOR H.B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2.

We have reviewed the information provided by your letters dated April 23, June 28 and July 12, 1979 in response to IE Bulletins 79-06A and 79-06A, Revision 1, for H. B. Robinson Steam Electric Plant, Unit No. 2 (Robinson). We have also reviewed your August 28, 1979 letter which responded to our August 9, 1979 letter requesting additional information regarding the aforementioned bulletins. The enclosure provides our evaluation of your responses with respect to their specificity, completeness, and responsiveness to the bulletins. In this regard, we have found that you have taken appropriate actions to meet the requirements of IE Bulletins 79-06A and 79-06A, Revision 1.

It should be noted that the staff review of the Three Mile Island, Unit 2 accident is continuing. Consequently, other corrective actions may be required at a later date. For example, IE Bulletin 79-06C was issued on July 26, 1979, requiring new considerations for operation of the reactor coolant pumps following an accident. Our reviews of the Westinghouse Owners' Group response to Items 2 and 3 of Bulletin 79-06C (Westinghouse reports WCAP-9584 and WCAP-9600, respectively) are documented in NUREG-0623 and NUREG-0611, respectively. You will be kept informed regarding the requirements for Robinson resulting from these reviews by separate correspondence.

Sincerely,

Original signed by:
S. A. Varga

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing

Enclosure:
Evaluation of Licensee's Responses to IE Bulletins 79-06A and 79-06A, Revision 1

Completed TAE 13130

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

July 17, 1980

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Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing

Enclosure:
Evaluation of Licensee's Responses
to IE Bulletins 79-06A and 79-06A,
Revision 1

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EVALUATION OF LICENSEE'S RESPONSES TO IE BULLETINS
79-06A AND 79-06A (REVISION 1)

H.B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
DOCKET NO. 50-261

INTRODUCTION

By letters dated April 14, and April 18, 1979, we transmitted our Office of Inspection and Enforcement (IE) Bulletins No. 79-06A and 79-06A (Revision 1), respectively, to Carolina Power and Light Company (the licensee). These bulletins specified actions to be taken by the licensee to avoid occurrence of an event similar to that which occurred on March 28, 1979 at Three Mile Island, Unit No. 2 (TMI-2). By letter dated April 23, 1979, the licensee provided its response to the aforementioned bulletins for H.B. Robinson Steam Electric Plant, Unit No. 2 (Robinson). The licensee supplemented its response by letters dated June 28 and July 12, 1979, providing clarification and elaboration of certain of the Bulletin Action Items in response to our expressed concerns. Following our review of the three licensee submittals, we requested additional information regarding the licensee's responses in our August 9, 1979 letter. By letter dated August 28, 1979, the licensee provided the requested information. Our evaluation of the licensee's responses, as supplemented, is provided below.

EVALUATION

In this evaluation, the paragraph numbers correspond to the bulletin action items and to the licensee's response to each action item.

1. In Bulletin Action Item No. 1, licensees were requested to review the description of circumstances described in Enclosure 1 of IE Bulletin 79-05 (issued to all licensees with Babcock & Wilcox (B&W)-designed plants for

action, and to all other licensees for information) and the preliminary chronology of the TMI-2 accident included in Enclosure 1 to IE Bulletin 79-05A (same distribution as IE Bulletin 79-05).

(a) This review should be directed toward understanding: (1) the extreme seriousness and consequences of the simultaneous blocking of both auxiliary feedwater trains at the Three Mile Island Unit 2 plant and other actions taken during the early phases of the accident; (2) the apparent operational errors which led to the eventual core damage; (3) that the potential exists, under certain accident or transient conditions, to have a water level in the pressurizer simultaneously with the reactor vessel not full of water; and (4) the necessity to systematically analyze plant conditions and parameters and take appropriate corrective action.

(b) Operational personnel should be instructed to: (1) not override automatic action of engineered safety features unless continued operation of engineered safety features will result in unsafe plant conditions (see Section 7a.); and (2) not make operational decisions based solely on a single plant parameter indication when one or more confirmatory indications are available.

(c) All licensed operators and plant management and supervisors with operational responsibilities were to participate in this review and such participation was to be documented in plant records.

On April 23, 1979, an NRC briefing team provided a detailed review of the circumstances described in Enclosure 1 of IE Bulletin 79-05 and the preliminary chronology of the TMI-2 accident included in Enclosure 1 of IE Bulletin 79-05A to a majority of the licensed operators and plant management. The briefing team consisted of an IE Projects Inspector, an Operator Licensing Branch (OLB/NRR) representative, and the facility Principal/Resident Inspector. Attendance was documented by the NRC Principal/Resident Inspector. The NRC briefing also provided a detailed review of Items 1.a and 1.b of IE Bulletin 79-06A. We consider the NRC briefing to be an acceptable response to Bulletin Action Item No. 1.

2. Action Item 2 of the Bulletin requested licensees to review actions required by operating procedures for coping with transients and accidents, with particular attention to (a) recognition of the possibility for forming voids large enough to compromise core cooling capability, (b) action required to prevent the formation of such voids, and (c) action required to enhance core cooling in the event such voids are formed. Emphasis in (a) was placed on natural circulation capability.

The licensee stated in its June 28, 1979 letter that it had completed a thorough review of all transient and accident conditions based on insight gained from TMI-2 to (a) assure that action steps specifically warn of potential for voiding with a description of all instrumentation which might provide indication of potential or actual voiding, (b) specifically address operator actions, based on operational modes and instrument indications discussed above, for terminating conditions tending to lead to void formation and (c) provide operators with guidance for enhancing core cooling given the unexpected condition of actual voiding in the primary system.

In accordance with the licensee's review and original response to Item 2 of the NRC IE Bulletin 79-06A, plant procedure EI-1, Incident Involving Reactor Coolant System Depressurization, was revised to specifically address operator actions, based on operational modes, for terminating conditions tending to lead to void formation. Included in this revision were steps for the operator to assure that all automatic ESF equipment operated as required upon either automatic or manual SI initiation, whichever method was used to initiate ESF. A method for reducing RCS temperature and pressure consistent with conditions which prevent exceeding the 50° subcooled temperature curves was added. A reference saturation temperature and 50° subcooled temperature curve along with a list of the minimum indications which the operator should use as a basis for his determinations and operational decisions was added to the plant procedures. In addition to changes to caution the operator of the potential for void formation and actions to eliminate conditions tending to form voids, the procedure was revised to specifically include a section which provides guidance for enhancing core cooling with actual voiding in the primary system.

The licensee has procedures regarding loss of reactor coolant flow to provide the operator with the actions required to establish and maintain natural circulation in case that total forced reactor coolant flow is lost. The Office of Inspection and Enforcement will verify that these procedures are acceptable.

All operators have been instructed to immediately trip the reactor coolant pump as specified by IE Bulletin 79-06C.

In addition, the licensee participated, as a member of the Westinghouse Owners Group, in the effort to develop generic guidelines for emergency procedures. In our November 5 and December 6, 1979 letters to the Owners Group, we approved the Westinghouse generic guidelines regarding small break LOCAs for implementation by licensees with Westinghouse-designed reactors. The Owners Group, in conjunction with Westinghouse, has also developed generic guidelines for emergency procedures regarding natural circulation. These generic guidelines were submitted on December 28, 1979, as part of the Owners Group response to the requirements of Item 2.1.9 of NUREG-0578 regarding inadequate core cooling. In order to satisfy NUREG-0578 requirements, the licensee should have incorporated the guidelines into the Robinson procedures (small break LOCA guidelines by January 1, 1980 and inadequate core cooling guidelines by January 31, 1980). The Office of Inspection and Enforcement will verify that acceptable guidelines have been properly implemented. Procedures based on these generic guidelines represent an acceptable method of complying with Bulletin Action Item No. 2.

We find that the licensee has provided an acceptable response to Bulletin Action Item No. 2.

3. Bulletin Action Item No. 3 requested that licensees with facilities that used pressurizer water level coincident with pressurizer pressure for automatic initiation of safety injection into the reactor coolant system trip the low pressurizer level setpoint bistables such that, when the pressurizer pressure reached the low setpoint, safety injection would be initiated regardless of the pressurizer level. The pressurizer level bistables could be returned to their normal operating positions during the pressurizer pressure channel functional surveillance tests.

   In its April 23, 1979 response, the licensee stated that the pressurizer level bistables which input to safety injection initiation had been placed in the trip mode using an Abnormal Procedure (AP). On May 24, 1979, we issued Amendment No. 38 to the Robinson operating license. This license amendment approved a design change to the safety injection initiation logic which the licensee had proposed. This design change consisted of modifying the safety injection initiation system logic so that safety injection will be initiated on a two-out-of-three low pressurizer pressure condition regardless of the pressurizer level. We consider the licensee's response to Bulletin Action Item No. 3 acceptable.

4. Bulletin Action Item No. 4 requested that licensees review the containment isolation initiation design and procedures, and implement all changes necessary to permit containment isolation, whether manual or automatic, of all lines whose isolation would not degrade needed safety features or cooling capability, upon automatic initiation of safety injection.

   Initiation of safety injection at Robinson by automatic or manual actuation signal actuates Phase A isolation of containment. Phase A isolates all non-essential process lines, but does not affect safety injection, containment spray, component cooling, or steam and feedwater systems. Therefore, Phase A isolation does not degrade needed safety features or cooling capability including the operation of reactor coolant pumps. Phase B isolation of containment is actuated by high-high containment pressure or by manual initiation of containment spray. Phase B isolation isolates all remaining process lines except safety injection, containment spray, and auxiliary feedwater.

We find that the licensee's response has adequately addressed the concerns expressed in Bulletin Action Item No. 4.

5. In Bulletin Action Item No. 5, licensees with facilities at which the auxiliary feedwater system is not automatically initiated were requested to prepare and implement immediately procedures which required the stationing of an individual (with no other assigned concurrent duties and in direct and continuous communication with the control room) to promptly initiate adequate auxiliary feedwater to the steam generator(s) for those transients or accidents, the consequences of which could be limited by such action.

The auxiliary feedwater system at Robinson is automatically initiated, with no operator action required in order to ensure adequate flow. Therefore, Bulletin Action Item No. 5 does not apply to this plant.

6. Bulletin Action Item No. 6 requested that licensees prepare and implement immediately procedures which:

(a) Identified those plant indications (such as valve discharge piping temperature, valve position indication, or valve discharge relief tank temperature or pressure indication) which plant operators could utilize to determine that the pressurizer power-operated relief valve(s) are open, and

(b) Directed the plant operators to manually close the power-operated relief block valve(s) if the reactor coolant system pressure had been reduced to below the set point for normal automatic closure of the power-operated relief valve(s) and the valve(s) remained stuck in the open position.

The licensee reviewed the applicable Robinson procedures and determined that no changes or revisions were needed to comply with Bulletin Action Item No. 6.a.

In response to Bulletin Action Item No. 6.b. the licensee revised emergency procedure EI-1 to ensure compliance with the requirements. Based on our review, we find that the licensee's response to Bulletin Action Item No. 6 is acceptable.

7. In Bulletin Action Item No. 7, licensees were requested to review the action directed by the operating procedures and training instructions to ensure that:

(a) Operators do not override automatic actions of engineered safety features, unless continued operation of engineered safety features would result in unsafe plant conditions. For example, if continued operation of engineered safety features would threaten reactor vessel integrity, then the high pressure injection (HPI) system should be secured (as noted in b(2) below).

(b) Operating procedures currently, or are revised to, specify that, if the (HPI) system had been automatically actuated because of a low pressure condition, it must remain in operation until either:

(1) Both low pressure injection (LPI) pumps are in operation and flowing for 20 minutes or longer at a rate which would assure stable plant behavior, or

(2) The HPI system has been in operation for 20 minutes, and all hot and cold leg temperatures are at least 50 degrees Fahrenheit below the saturation temperature for the existing RCS pressure. If 50 degrees subcooling cannot be maintained after HPI cutoff, the HPI shall be reactivated. The degree of subcooling beyond 50 degrees and the length of time HPI has been in operation shall be limited by the pressure/temperature considerations for the vessel integrity.

(c) Operating procedures currently, or are revised to, specify that, in the event of HPI initiation with reactor coolant pumps (RCPs) operating, at least one RCP shall remain operating for two-loop plants and at least two RCPs shall remain operating for 3 or 4 loop plants, as long as the pump(s) is providing forced flow.

(d) Operators are provided additional information and instructions to not rely upon pressurizer level indication alone, but to also examine pressurizer pressure and other plant parameter indications in evaluating plant conditions, e.g., water inventory in the reactor primary system.

In response to Bulletin Action Item No. 7.a, the licensee reviewed the appropriate Robinson plant emergency procedures and they prohibit overriding engineered safety features unless the continued operation would result in unsafe conditions. This constitutes an acceptable response to Bulletin Action Item No. 7.a.

In response to Bulletin Action Item No. 7.b, the licensee participated in the effort by the Westinghouse Owners Group, in conjunction with Westinghouse, to develop generic guidelines for emergency procedures. In our November 5 and December 6, 1979 letters to the Owners Group, we approved generic guidelines for emergency procedures regarding small break LOCAs for implementation by licensees with Westinghouse-designed operating plants. These approved guidelines include the following criteria (taken from the enclosure to our letter of December 27, 1979) for termination of safety injection:

(1) The reactor coolant system pressure is greater than 2000 pounds per square inch gauge and increasing, <u>and</u>

(2) The pressurizer water level is greater than the programmed no-load water level, <u>and</u>

(3) The reactor coolant indicated subcooling is greater than (insert plant-specific value, which is the sum of the errors for the temperature measurement system used and the pressure measurement system translated into temperature using the saturation tables), <u>and</u>

(4) The water level in at least one steam generator is stable and increasing, as verified by auxiliary feedwater flow to that unit. Auxiliary feedwater flow to the unaffected steam generator should be greater than <u>(a value in gallons per minute sufficient to remove decay heat after 20 minutes following reactor trip</u>) until the indicated level is returned to within the narrow range level instrument.

Details of our evaluation of this issue are included in the report (NUREG-0611) of our generic review of Westinghouse-designed operating plants.

Our Office of Inspection and Enforcement will verify that the approved Westinghouse generic safety injection termination criteria have been properly incorporated in the Robinson plant procedures. Pending such verification, we find that the licensee's actions with regard to this bulletin action item are acceptable.

Another issue on which the Westinghouse Owners Group worked, in conjunction with Westinghouse, to achieve resolution with the staff was the matter of reactor coolant pump operation following a small break LOCA (Bulletin Action Item No. 7.c). On July 26, 1979, IE Bulletin 79-06C superseded Action Item No. 7.c of Bulletin 79-06A. Bulletin 79-06C required that, as a short-term action, licensees were to trip all reactor coolant pumps after an initiation of safety injection caused by low reactor coolant system pressure. In its August 28, 1979 response to Bulletin 79-06C, the licensee stated its conformance with this requirement. This action was to remain in effect until the results of analyses specified in Bulletin 79-06C had been used to develop new guidelines for operator action.

We have completed our review of the reactor coolant pump trip issue with the Owners Group. The generic guidelines for emergency procedures regarding small break LOCAs, which we approved in our November 5 and December 6, 1979 letters to the Owners Group, contain the approved pump trip criteria for Westinghouse-designed operating plants. Basically, they are as follows:

(1) Stop all reactor coolant pumps after high pressure safety injection pump operation has been verified, and when the wide range reactor pressure is at (plant-specific pressure derived from secondary system relief capacity, primary-to-secondary system pressure difference, and instrument inaccuracies).

Appropriate cautions have been included in the guidelines regarding isolation of component cooling water to the reactor coolant pumps and maintaining seal injection flow to preclude pump damage due to inadequate cooling. The details of our review of the pump trip issue are reported in NUREG-0623.

Pending confirmation by our Office of Inspection and Enforcement that the licensee has incorporated the pump trip criteria as specified in the approved Westinghouse generic guidelines into the Robinson plant instruction to operators, we find the licensee's response to Bulletin Action Item No. 7.c acceptable.

In response to Bulletin Action Item No. 7.d, the licensee issued a instructions to Robinson operations personnel which cautioned against overreliance on pressurizer level indication, and recommended examination of other plant parameters in assessing water inventory and plant conditions. In its April 23, 1979 letter, the licensee identified the specific plant parameters to be used in assessing water inventory and plant conditions. We find these actions to be an acceptable response to Bulletin Action Item No. 7.d.

8. Bulletin Action Item No. 8 required that licensees review alignment requirements and controls for all safety-related valves necessary for proper operation of engineered safety features. In response, the licensee stated that the required review was conducted by reviewing valve positions concurrently with the procedures that check or manipulate the valves. In its responses, the licensee noted that valve lineups on safety-related systems were completed prior to startup. Locked valves on safety-related systems are verified and documented with respect to their proper position. Safety-related valves that have position indication in the control room are verified to be in their proper positions on a shift turnover check list which meets the requirements of Item 2.2.1.c of NUREG-0578, "Shift and Relief Turnover Procedures."

Based on our review, we find the licensee's response to Bulletin Action Item No. 8 acceptable.

9. In Bulletin Action Item No. 9, licensees were requested to review their procedures to assure that radioactivity will not be inadvertently released from containment. Particular emphasis was placed on the resetting of engineered safety features (ESFs) and the effects of this action on valves controlling the release of radioactivity.

In its June 28 and July 12, 1979 supplemental responses, the licensee listed all systems which are designed to transfer potentially radioactive fluids from containment, indicated those systems for which high radiation interlocks exist, and identified the means by which the operability of each system listed is assured. Information pertaining to the resetting of ESFs and its effect on valves controlling the release of radioactivity was provided in the licensee's December 31, 1979 response to Item 2.1.4 of NUREG-0578. In brief, once Phase A Containment Isolation has been initiated by a safety injection signal, the automatic isolation valves can be opened only upon manual reset of the actuating signal and deliberate remote manual operation of the individual valve.

We find that the licensee has adequately addressed the concerns expressed in Bulletin Action Item No. 9.

The staff's implementation of Item 2.1.4 of NUREG-0578 provides further assurance that the inadvertent release of radioactivity from containment upon resetting of ESFs will be precluded. Our review of NUREG-0578 Item 2.1.4 implementation will be reported in a separate document.

10. Action Item No. 10 of Bulletin 79-06A required that licensees review and modify, as necessary, maintenance and test procedures for safety-related systems to ensure that they require that: (a) redundant systems are operable before a system is taken out of service, (b) systems are operable when returned to service, and (c) operators are made aware of the status of these systems.

In its June 28 and August 28, 1979 supplemental responses, the licensee provided additional information regarding this bulletin action item. The operability of redundant pieces of equipment in safety-related equipment is verified prior to removal of any safety-related component from service consistent with the minimum equipment lists as developed from the Limiting Conditions for Operation listed in the unit Technical Specifications. Operability is verified by test or by visual inspection. The visual inspection consists of, as a minimum, a review of the equipment status on the control board. Applicable tests or inspections are specified in the individual Operating Work Permit and/or the applicable equipment trouble and work report.

All Operating Work Permits have been reviewed and corrected as applicable to verify that all equipment is properly aligned prior to returning the equipment to operable status following maintenance.

The transfer of information about the status of safety-related systems at shift change will be accomplished according to the requirements of Item 2.2.1.c of NUREG-0578.

Based on our review, we find that the licensee's response to Bulletin Action Item No. 10 is acceptable.

11. Bulletin Action Item No. 11 requested licensees to review their prompt reporting procedures for NRC notification to assure that the NRC is notified within one hour of the time the reactor is not in a controlled or expected condition of operation. Further, at that time, an open, continuous communication channel shall be established and maintained with the NRC.

    In its April 23, 1979, letter, the licensee committed to meet these requirements. We find the licensee's commitment to Bulletin Action Item No. 11 acceptable.

    The actions specified in Action Item No 11 of IE Bulletin 79-06A have subsequently been incorporated in the requirements of Section 50.72 of 10 CFR Part 50, immediately effective upon issuance February 29, 1980.

12. In Action Item No. 12, licensees were requested to review operating modes and procedures to deal with significant amounts of hydrogen gas that may be generated during a transient or other accident that would either remain inside the primary system, or be released to the containment.

    In response to this bulletin action item, the licensee reviewed the existing Robinson procedures to assure adequate post-accident containment sampling circulation, and means of hydrogen removal which may be released to the containment during a transient or accident condition.

In addition, in its April 23, 1979 response, the licensee identified the various methods covered by existing procedures for removing hydrogen gas from the reactor coolant system.

Based on our review, we find that the licensee has provided an adequate response to Bulletin Action Item No. 12.

13. This bulletin action item requested licensees to propose changes, as required, to those plant Technical Specifications which had to be modified as a result of implementing Bulletin Action Item Nos. 1 through 12, and to identify design changes necessary in order to effect long-term resolution of these items.

    By letter dated May 18, 1979, the licensee requested a license amendment to the Robinson Technical Specifications necessitated by actions required by this bulletin. This change was required to implement two-out-of-three low-low Pressurizer Pressure Safety Injection actuation (from Bulletin Action Item No. 3) and was approved by the NRC on May 24, 1979. The licensee's letter of July 12, 1979, documents a design change to the containment isolation valves from the PRT and RCDT to the gas analyzer. This change will be completed during the next refueling outage.

    We find the licensee's response to Bulletin Action Item No. 13 acceptable.

CONCLUSIONS

Based on our review of the information provided by the licensee, we conclude that the licensee has correctly interpreted IE Bulletins 79-06A and 79-06A, Revision 1. The actions taken demonstrate the licensee's understanding of the concerns arising from the Three Mile Island, Unit No. 2 accident in relation to their implications on its own operations, and provide added assurance for the protection of the public health and safety during plant operation.