EVALUATION OF LICENSEE'S RESPONSES TO IE BULLETINS

79-06A AND 79-06A REVISION 1

AND NUREG-0737 ITEM II.K.1

JOSEPH M. FARLEY NUCLEAR PLANT, UNIT NO. 1

DOCKET NO. 50-348

INTRODUCTION

By letters dated April 12 and April 18, 1979, we transmitted our Office of Inspection and Enforcement (IE) Bulletins No. 79-06A and 79-06A (Revision 1), respectively, to the Alabama Power 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 24, 1979, the licensee provided its response to the aforementioned bulletins for Joseph M. Farley Nuclear Plant, No. 1. By letter dated May 22, 1979 we provided to all licensee's our preliminary review of the licensee responses. In that letter we scheduled and held a meeting on May 30, 1979 with owners having Westinghouse designed nuclear supply systems. Subsequently, the licensee supplemented its response by letters dated June 22 and August 28, 1979, providing clarification and elaboration of certain of the Bulletin Action Items in response to our expressed concerns. By letter dated January 14, 1981 the licensee included the response to NUREG-0737, Item II.K.1, which is the identical item of concern. 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.

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(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 June 7, 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 Section Leader, Nuclear Reactor Regulation Operator Licensing Branch representative, and the facility Principal IE Inspector. Attendance was documented with any missing personnel being required to review a video tape of the initial presentation at a later time. The NRC briefing also provided a detailed review of Items 1.a and 1.b of IE Bulletin 79-06A. In addition, a follow-up IE Inspection Report #79-26 dated July 25, 1979 documented the inspector's review of the licensee administered training program relating to the TMI incident. No items of noncompliance were disclosed in this inspection. We consider the NRC briefings 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. By letter dated April 24, 1979 the licensee advised that Emergency Operating Procedure (EOP) 1 - Loss of Reactor Coolant would be revised with guidance specified for the operator for these accidents prior to return to power operation from the first refueling outage (restarted about November 6, 1979).

In response to the requirements of Item 2.1.3.b of NUREG-0578, the licensee, while shutdown during March to November 1979, installed saturation meters which calculates the margin to saturation using redundant safety grade pressure and temperature inputs. Our letter of April 3, 1980 provided the evaluation of Category "A" TMI-2 Lessons Learned items including use of these meters.

Procedures have been developed by the licensee to (1) alert the operator of the potential for voiding and address actions for terminating conditions which lead to voiding, and (2) enchance core cooling should voiding take place in the reactor coolant system, taking into account both forced and natural circulation cooling.

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 incorporated appropriate guidelines into the Farley Unit No. 1 procedures. 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 24, 1979 response, the licensee stated that the pressurizer level bistables which input to safety injection initiation had been placed in the trip mode. Trip status lights on the control board confirm that the action has been completed. On June 11, 1979, we issued Amendment No. 12 to the Farley Unit No. 1 operating license. This license amendment approved the design change to the safety injection initiation logic which the licensee had proposed on May 11, 1979. 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. By letter of June 22, 1979 the licensee advised that operating procedures were revised on May 18, 1979 to require operators to manually initiate a reactor trip or safety injection if their setpoints and automatic actuation does not occur. IE Inspection Report 79-26 dated July 25, 1979 verified that Procedure AP-16 requires such operator action.

We consider the licensee's action taken and the 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 Farley Unit No. 1 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, service water, 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 a high-high containment pressure signal. Phase B isolation isolates all remaining process lines except safety injection, containment spray, and auxiliary feedwater. Although operation of the reactor coolant pumps cannot continue for very long when Phase B isolation stops component cooling water to the pump seals and motor bearings, the high containment pressure or need for containment spray would indicate a large rapid blowdown of the primary system. In that event, the reactor coolant pumps would not be of any use until after longer term reflooding had taken place. We, therefore, find this response acceptable.

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 Farley Unit No. 1 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.

Operating Procedure EOP-1 Loss of Reactor Coolant was revised on May 18, 1979 to list specific plant parameters which are indications of a leaking or open relief valve. IE Inspection Report #79-26 dated July 25, 1979 verified the action. The operator is to close the relief valve and its block valve if the PORV appears stuck open.

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 Safety Injection System (HPSI) should be secured (as noted in b(2) below).

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

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

(2) The HPSI 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 reactor coolant system pressure. If 50 degrees subcooling cannot be maintained after HPSI cutoff, the HPSI shall be reactivated. The degree of subcooling beyond 50 degrees and the length of time HPSI 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 HPSI initiation with reactor coolant pumps (RCPs) operating, at least two RCPs shall remain operating for a three loop plant such as Farley Unit No. 1, as long as the pumps are 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 applicable Farley Unit No. 1 operating procedures to ensure that overriding engineered safety features is prohibited, except in cases of spurious actuation, or the initiating cause is alleviated, or unless continued operation of engineered safety features 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. Those 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.

We find that the licensee's actions with regard to this bulletin action item are acceptable when the approved Westinghouse generic guidelines have been properly incorporated in the Farley Unit 1 plant procedures.

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 RCP trip criteria for Westinghouse-designed operating plants. We will complete our review of the concern as part of TMI Task Action Plan Item II.K.3.5.

In response dated April 24, 1979 to Bulletin Action Item No. 7.d, the licensee advised that it would revise emergency operating procedures which deal specifically with conditions similar to that which occurred at TMI. Procedures will identify certain specific plant parameters to be used in assessing water inventory and plant conditions. Operators were cautioned in the necessity to monitor all the listed parameters and not to solely rely on one parameter in detection of accident.

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 dated April 24, 1979, the licensee noted the following relating to its review of valve position (alignment) requirements:

   (a) Operating Procedures have valve check lists establishing initial valve positions by sign-off for start of system operation.

   (b) Surveillance Test Procedures require sign-off of valve alignments for required flow-paths.

   (c) Positive controls (locks on valves, etc.) are by sign-off steps in procedures.

   (d) Following major outages, significant maintenance or when returning systems to service from maintenance or off-normal operation, valve check lists or flow path verifications have been plant policy. This policy was to be incorporated into administrative procedures.

   (e) Shift relief includes a walk-down of main control room boards to check proper alignment of remotely-operated equipment.

   Subsequently, by letter of June 22, 1979 the licensee reported its review of associated procedures had been completed prior to exceeding Mode 5 operation.

   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 April 24, 1979 response, the licensee listed all systems which are designed to transfer potentially radioactive fluids from containment. The licensee indicated that these systems have either high radiation interlocks or that the systems cannot be operated until containment isolation is reset and valves are manually actuated. One exception is the Post Accident Venting System where the containment isolation valves are locked closed.

   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 is reported in our letter of April [illegible], 1980.

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.

    The Farley 1 Technical Specifications specify the surveillance requirements that must be completed to confirm the operability of safety-related systems. A subsystem or equipment is removed from service for preventive or corrective maintenance according to maintenance operating procedures. When a subsystem fails or is removed from service, this event must be approved on an appropriate form or procedure. When maintenance has been completed, the controlling procedure (AP-2) ensures that testing of the subsystem equipment is performed to determine operability.

    The licensee in its April 24, 1980 letter noted specific actions to be included in AP-52, Equipment Status Control and Maintenance Authorization which would require verification, by test or inspection, that the redundant subsystem/train is operable before removal of a portion of the other subsystem/train.

    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. In addition, the licensee modified procedure AP-16, Conduct of Operations - Operations Group, to clarify that the Operator-at-the-Controls log the removal and return to service of all safety related systems. Procedure AP-52 (noted above) was also modified to require explicit notification of the operator by the Shift Foreman when a safety-related system is removed from service or found inoperable and returned to service.

    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.

    The Farley Unit No. 1 Emergency Implementing Procedures were clarified to specify that the NRC be notified within one hour of the time the reactor is not in a controlled or expected condition of operation. Provisions are included for establishing and maintaining a continuous open channel of communication with the NRC. IE Inspection Report #79-26 dated July 25, 1979 identified the specific procedures revised.

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.

We find the licensee's action in response to Bulletin Action Item No. 11 acceptable.

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 Farley Unit No. 1 procedures regarding removal of hydrogen gas from the containment. The licensee listed several methods for hydrogen removal in the primary system and committed to incorporate these methods into existing procedures. The licensee described the existing hydrogen recombiner system for removal of hydrogen from the containment, indicated the secondary method for hydrogen removal through use of a controlled vent through charcoal filters, and stated that operating modes and procedures were reviewed.

    In addition, the licensee participated in the Westinghouse Owners Group effort to develop generic guidelines for emergency operational procedures in response to the requirements of Item 2.1.9 of NUREG-0578. On December 28, 1979, the Owners Group submitted generic guidelines regarding natural circulation and post-LOCA long-term cooldown for staff review. Treatment of noncondensible gas in the reactor coolant system was considered in the development of these guidelines. Our review of NUREG-0578 implementation will be reported in a separate document.

    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.

    In its April 24, 1979 letter, the licensee advised that proposed changes to the Joseph M. Farley Plant Technical Specifications required by this bulletin would be submitted by May 14, 1979. On May 11, 1979 the licensee proposed a change in the control logic deleting the low pressurizer level from the safety injection actuation logic (discussed above in Action Item No. 3). Other Technical Specification changes have not been required.

    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. Also, Action Item II.K.1 of NUREG-0737 is considered complete with this action. 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.

DATE: May , 1981