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

August 27, 1980

Docket No.: 50-334

Mr. C. N. Dunn, Vice President
Operations Division
Duquesne Light Company
435 Sixth Avenue
Pittsburgh, PA 15219

Dear Mr. Dunn:

Subject: NRC STAFF EVALUATION OF DUQUESNE RESPONSES OT IE BULLETINS 79-06A AND 79-06A, REVISION 1, FOR BEAVER VALLEY POWER STATION, UNIT NO. 1.

We have reviewed the information provided by your letters dated April 30, May 14, May 17, and July 12, 1979 in response to IE Bulletins 79-06A and 79-06A, Revision 1 for Beaver Valley Power Station, Unit No. 1. We have also reviewed your September 21, 1979 letter which responded to our September 13, 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 the Beaver Valley Unit 1 plant resulting from these reviews by separate correspondence.

Sincerely,

Marshall Grotenhuis

for S. Varga, Chief
Operating Reactors Branch #1
Division of Operating Reactors

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)

BEAVER VALLEY POWER STATION, UNIT NO. 1
DOCKET NO. 50-334

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 Duquesne 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 30, 1979, the licensee provided its response to the aforementioned bulletins for Beaver Valley Power Station, Unit 1 (Beaver Valley 1). The licensee supplemented its response by letters dated May 14, May 17, 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 four licensee submittals, we requested additional information regarding the licensee's responses in our September 13, 1979 letter. By letter dated September 21, 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 2, 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, an Operator Licensing Branch (OLB/NRR) representative, and the facility Principal Inspector. Attendance was documented. 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 No. 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.

In its July 12, 1979 submittal, the licensee summarized revisions that had been implemented in the Beaver Valley Power Station Operating Procedures related to the potential for void formation, i.e., ECCS actuation, loss of reactor or secondary coolant, steam generator tube rupture, loss of feedwater, loss of offsite power, total loss of reactor coolant system flow or inadvertent safety injection.

In recognition of the possibility of forming void, operators are to give special attention to the pressure and temperature of the reactor coolant and to be aware of the results of void formation.

A chart with saturation and 50F degrees subcooling and saturation conditions is provided and the operator is required to determine if safe conditions exist on the basis of four parameters: Reactor Coolant System Pressure and Temperatures (hot), and the levels of the Pressurizer and Steam Generator.

Actions to be taken to prevent formation of voids include maintaining programmed levels for all steam generators, and assuring that safety injection is available for water inventory makeup and core cooling. If forced flow of the Reactor Coolant System is lost, specific instructions are provided to enhance natural circulation and guidance is provided to ensure that natural circulation exists by means of the Reactor Coolant temperatures and Steam Generator Pressure.

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 Beaver Valley 1 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 30, 1979 letter the licensee stated that the pressurizer level conincident bistables would be placed in the trip mode whenever the Reactor Coolant System pressure was greater than 2000 psig. Subsequently, in its July 12, 1979 letter, the licensee reported that procedures that require manual initiation of safety injection at the pressurizer pressure setpoint of 1845 psig had been implemented. By letter of June 26, 1979, the licensee proposed Technical Specification revisions to initiate Safety Injection and Feedwater Isolation on low pressurizer pressure only. This single actuation is achieved by removing the pressurized level signal from each of the pressurizer level pressure channel trips and converting the system to a two-out-of-three logic based on the pressurizer low pressure trips. We find that these actions provide acceptable responses to Item 3.

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.

The Beaver Valley 1 design provides for automatic initiation of containment isolation upon safety injection actuation, as called for in the bulletin. This aspect of the licensee's response is therefore acceptable.

Containment isolation consists of a Phase A and a Phase B isolation. Phase A involves closure of automatic valves in all non-essential process lines; Phase B isolates all remaining process lines, except for those related to engineered safety features. The reactor coolant pump seal water return line is isolated upon a Phase A signal. The seal water supply is not provided with isolation valves. The component cooling water supply and return lines for the reactor coolant pumps are isolated by a Phase B signal. The reactor coolant pumps do not trip automatically on either isolation signal. Therefore, the pumps must be manually tripped following a Phase B isolation, since component cooling water to the motor coolers and thermal barriers is lost.

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 Beaver Valley 1 is automatically initiated upon receipt of either loss of heat sink indication or safety injection actuation. 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.

   Present Beaver Valley-1 procedures assure that adequate indicators are available to the operator for: the position of the power operated relief valve (PORV); temperature of the PORV and Safety Valve Relief Lines; and three parameters (pressure, level, and temperature) for the pressurizer relief tank. The procedures also provide the direction required in

Item 6(b); therefore, we find that the licensee's response to Item 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 revised the applicable Beaver Valley 1 plant procedures in July 1979 to prohibit overriding engineered safety features 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. 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 50°F, <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 (a value in gallons per minute sufficient to remove decay heat after 20 minutes following reactor trip) 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.

The Beaver Valley 1 procedures include Criteria 1, 2, and 3 as written. Criterion 4 has been modified to the extent that water level in one steam generator is verified by being visible in the narrow range span (5%) or in the wide range span at a level sufficient to assure coverage of the "U" tubes (i.e., 55%).

Our Office of Inspection and Enforcement will verify that the approved Westinghouse generic safety injection termination criteria have been properly incorporated in the Beaver Valley-1 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:

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 Beaver Valley 1 plant procedures, 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 revised its operating procedures to require that the operator observe seven indications in addition to pressurizer level when determining the condition of the plant. In its July 12, 1979 letter, the licensee identified these parameters. We find this action to be an acceptable response to this Action Item.

8. Bulletin Action Item No. 8 required that the licensees review alignment requirements and controls for all safety-related valves necessary for

proper operation of engineered safety features. In its July 12, 1979 letter the licensee stated that such a review of alignment requirements and procedures for controlling manipulation of safety-related valves had been made. In its September 21, 1979 supplemental letter, the licensee added that schedules for subsequent surveillance are available in the technical specification and station procedures. These schedules have been confirmed by the Office of Inspection and Enforcement. We find this response to be 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 July 12, 1979 supplemental response, the licensee listed all systems which are designed to transfer potentially radioactive gases and liquids out of containment. These systems do not have interlocks with high radiation levels. In both its April 30, 1979 and supplemental responses, these systems are stated to isolate on Containment Isolation (CI) - Phase A and to fail closed. Manual operator action is required to open the flow paths of these systems after CI - Phase A has been reset. We find this response to be acceptable.

   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 letter of April 30, 1979, the licensee stated that station procedures require that a licensed senior operator verify that redundant safety-related equipment or systems will not be affected by the removal of any safety-related system from service. The licensee stated in both its April 30 and July 12, 1979 letters that it relies primarily on prior operability verification within the limits of its Technical Specification surveillance interval. In its additional supplemental response on September 21, 1979 the licensee stated that the station procedures were revised to provide for a visual check of the redundant system or subsystem status within eight hours prior to removing an Engineered Safety Feature system or subsystem from service. No activities may be done on the redundant system subsequent to the visual check. This satisfies the intent of Action Item 10.a.

Existing procedures require verification of the operability of safety-related systems when they are returned to service following maintenance and testing.

In its April 30, 1979 letter, the licensee stated that present procedures require notification of reactor operating personnel prior to removing from or returning to service any safety-related system. In its supplemental letter of July 12, 1979, the licensee states that the level of authority for these actions is the Nuclear Shift Supervisor. Any requirement for demonstration of operability of redundant equipment is carried over to each shift by means of the Shift Supervisor's log until the equipment has been returned to service.

Based on our review we find the licensee's responses to Bulletin Items No. 10a, 10b, and 10c to be 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 letters of April 30 and July 12, 1979, the licensee stated that station procedures require one hour notification to the NRC in the event that the reactor is not in a controlled or expected condition of operation. The DPX network linking Beaver Valley 1 and the NRC Operations office and Region I office will be used for this purpose. We find these actions acceptable.

The actions specified in Action Item No. 11 of IE Bulletin 79-06A have subsequently been incorporated into the requirements of Section 50.72 of 10 CFR Part 50, effective immediately 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 Beaver Valley 1 procedures regarding removal of hydrogen gas from the containment using installed hydrogen recombiners.

In its September 21, 1979 supplemental response, the licensee stated that a procedure had been incorporated into the Beaver Valley 1 operating manual to encompass all modes that are available for the removal of a hydrogen bubble from the reactor coolant system. These modes had been described in the licensee's supplemental submittal of July 12, 1979, and include: (1) stripping hydrogen from the reactor coolant to the pressurizer vapor space and venting to the pressurizer relief tank; (2) removing hydrogen from the reactor coolant system via the letdown line and stripping it in the volume control tank and venting through the waste gas system; and (3) in the event of a LOCA, venting hydrogen with steam into containment.

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.

    The licensee identified the one change to the Beaver Valley 1 Technical Specifications necessitated by actions required in this bulletin. This modification would delete the level coincidence of the pressurizer safety injection logic and alter the pressure conicidence to a 2 out of 3 low pressure logic. The Beaver Valley-1 Technical Specifications were revised to implement this change by means of Amendment No. 19 to License DPR-66.

    We find the licensee's response to 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.