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

50-320

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Docket No. 50-346

Mr. Lowell E. Roe
Vice President, Facilities
Development
Toledo Edison Company
Edison Plaza
300 Madison Avenue
Toledo, Ohio 43652

Dave Bense

Dear Mr. Roe:

We have completed a review of the information you have provided in letters dated April 11, 1979, and April 16, 1979, in response to IE Bulletin 79-05A. Enclosure 1 provides an evaluation of your responses and discusses them with respect to their specificity, completeness, or the interpretation you have given to the requirements of this bulletin. The evaluation also discusses areas of concern to the staff additional to those contained in the bulletin. In several places in Enclosure 1, we have identified additional action or information that is needed. These items have been summarized in Enclosure 2.

We request that responses to the items of Enclosure 2 be forwarded within 30 days of receipt of this letter.

Sincerely,

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Enclosures:
1. Evaluation Report
2. Request for Additional Information

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Roisman 79-98

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Enclosure 1

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EVALUATION OF LICENSEE'S RESPONSES TO IE BULLETIN 79-05A

TOLEDO EDISON COMPANY

AND

CLEVELAND ELECTRIC ILLUMINATING COMPANY

DAVIS-BESSE NUCLEAR POWER STATION - UNIT 1

DOCKET NO. 50-346

Introduction

By letter dated April 1, 1979 we transmitted to the Toledo Edison Company (TECo or the licensee), IE Bulletin No. 79-05. This bulletin specified actions to be taken by the licensee to preclude occurrence of an event similar to that which occurred at Three Mile Island, Unit No. 2 (TMI-2) on March 28, 1979. This bulletin was expanded and revised by IE Bulletin 79-05A (the Bulletin). By letter dated April 11, 1979, supplemented by letter dated April 16, 1979, the licensee provided his responses in conformance with the requirements of the Bulletin.

Our evaluation of these responses is given below. In performing this evaluation we have utilized the additional clarification provided in IE Bulletin 79-06A issued on April 14, 1979. We have also evaluated the licensee's responses relative to those additional requirements specified in Bulletin 79-06A which are applicable to Davis-Besse Nuclear Power Station - Unit 1 (Davis-Besse - 1).

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EVALUATION

Item 1

As a result of his review of the preliminary sequence of events of the incident at TMI-2 described in the IE Bulletin 79-05A, the licensee, assisted by Babcock and Wilcox Company, has identified the same six statements of concern outlined in the "Description of Circumstances" of the Bulletin. He has also related these statements of concern to the Davis-Besse-1 plant. We have reviewed his responses to these statements and find, in general, a satisfactory understanding of the sequence of events and their relationship to his facility. It is not clear, however, that the licensee's review has been sufficiently broad-based with respect to the personnel involved in the review. Therefore, to assure that all appropriate levels of the utility personnel have participated in this review, we will request a commitment that all licensed operators and plant management and supervisors with operational responsibilities participate in this review and such participation be promptly concluded and documented in plant records.

In response to Statement 1 concerning both auxiliary feedwater trains being valved out of service, he states that he is implementing additional valve verification (including independent verification) and positioning controls to ensure auxiliary feedwater operability. We will request the licensee to identify the procedures and provide

a schedule of completion of the procedures for verification of valve positioning.

In reference to Statement 2 concerning the pressurizer electromatic relief valve (ERV), he states that, in case of a failure of this valve to close, emergency procedures would govern operator action including isolation of the pressurizer ERV. We have also been informed that this valve position indication is provided in the control room and that a temperature monitor downstream of this valve is alarmed to provide indication that the valve is open.

We believe the temperature monitor alone may not be a valid indication of relief valve position, particularly after initial operation, and that other instrumentation (such as those that indicate the ERV discharge relief tank parameters) should also be used to determine the status of the ERV. The procedures and instructions to the operators concerning the ERV should be outlined and should emphasize the role of the block or isolation valve for the ERV. We will request the licensee to commit to prepare and implement immediately procedures which:

a. Identify those plant indications (such as valve discharge piping temperature, valve position indication, or valve discharge relief tank temperature, pressure, or level indication) which plant operators may utilize to determine that pressurizer power operated relief valve(s) are open; and

b. Direct the plant operators to manually close the power operated relief block valve(s) when reactor coolant system pressure is reduced to below the set point for normal automatic closure of the power operated relief valve(s) and the valve(s) remain stuck open.

In the licensee's response to Statement 3 in Item 1, we have a concern here as elsewhere throughout his response regarding the identification of the operating procedures which were reviewed and revised. We will request that the licensee provide the list by number and title of all procedures reviewed, revised, or prepared. If additional aids are provided the operator (reactor coolant system pressure/ temperature curves, for example), we will request that these aids be clearly described and that the appropriate procedures and training instruction be referenced.

Statement 4 of Item 1 relates to the possibility of pumping highly radioactive water out of the containment. The licensee indicates that containment isolation for his facility is initiated by 1600 psig low reactor coolant pressure (High Pressure Injection initiation) or by four psig, high reactor containment pressure. We fi that the containment should isolate early in a transient before any significant release of radioactivity would occur. However, Item 9 of the Bulletin also requires the licensee to review operating modes and procedures to further

assure that undesired pumping of radioactive liquids and gases out of the containment will not occur inadvertently. This is addressed in our evaluation of the response to Item 9 given below.

Statement 5 of Item 1 of the Bulletin relates to the adverse effect of intermittent operation of the High Pressure Injection (HPI) system at TMI-2 to maintain pressurizer level. The licensee states that operating procedures are being revised to identify the criteria and concurrence required prior to overriding automatically activated safety systems. Our evaluation of this matter is addressed below in our consideration of Items 4a, b and d.

In response to Statement 6 of Item 1, the licensee states that modifications to procedures are being prepared in response to Bulletin Item 4 which will preclude "premature" termination of reactor coolant pump operation.

Although not specifically detailed in Item 1 of the Bulletin, we will request further information and commitments from the licensee concerning the review of operating modes and procedures to deal with significant amounts of hydrogen that may be generated during a transient or other accident that would either remain inside the primary system or be released to the containment.

Item 2

In response to Item 2 of the Bulletin, the licensee stated that he had reviewed all transients which had occurred at Davis-Besse-1 that had been initiated by loss of feedwater flow or excessive feedwater flow. He identified five transients relating to this item including four which occurred before the final tuning of the Integrated Control System (ICS) was completed.

These transients will, during the staff's on-going evaluation of the TMI-2 incident, be reviewed to determine whether further changes or modifications may be desirable to provide added assurance that a TMI-2 accident will not be repeated.

The response to Item 2 was limited to feedwater flow events. The licensee will be requested to review all events relating to the TMI-2 event such as events relating to the ERV, lockout of the auxiliary feedwater valves, or shutoff of the HPI. We will also request that the licensee inform us of any procedures that exist to guide operator response to any of the various modes of failure of the ICS.

Item 3

We have reviewed the licensee's response and find it acceptable except that the length of time for completion of the operator training appears unduly protracted. We believe that the 120 days allowed for completion of the training is much too long and that a significant reduction in the time allowed should be made.

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Item 4a

The licensee's response to this item states that no safety systems will be bypassed without approval of the Station Management if there are questionable conditions or any sign of a RCS leak. The licensee has indicated that procedures are being modified to restrict the operators to the manipulation of only specified components. Also the licensee has indicated that other systems may be bypassed, with approval, if plant conditions are stable after a transient.

The licensee will be requested to commit to Item 7a of Bulletin 79-06A which supplements Item 4a of Bulletin 79-05A, and will be requested to indicate presently anticipated criteria under which engineered safety features would be shutdown or restarted to prevent an unsafe plant condition.

Item 4b

Item 7b of Bulletin 79-06A provides a revision and clarification of Item 4b of Bulletin 79-05A. We generally concur in the action taken by the licensee in response to this item. However, we will request the licensee to provide a basis for throttling of the HPI flow whenever

total flow exceeds 900 gpm. Also, we will request that he list all procedures reviewed, including appropriate emergency procedures, and identify those that were revised as a result of this review. In addition, we will request that he list the engineered safety feature systems included in his review.

Item 4c

The licensee's response to this item states that in the event of HPI initiation with reactor coolant pumps (RCP) operation, at least one RCP per loop shall remain operating. We find this acceptable. However, we will request the licensee to expand his procedures in accordance with Item 7c of Bulletin 79-06A which requires the operation of RCPs as long as the pumps are providing forced flow and to identify the parameters that indicate forced flow.

Item 4d

The licensee states that the operators have received supplemental and implicit training in all aspects of the TMI-2 incident. It is not clear from this response the nature of the direction provided to the reactor operators to utilize pressure/temperature relationships nor the form of pressure/temperature relationships that will assist the operator in evaluating abnormal plant conditions. Also, additional indications may be available to the operators to identify a decreasing reactor coolant system water inventory (for example, pressurizer relief tank parameter indications).

We will request that the licensee provide details regarding the nature of the direction given to the operators in response to this item and the form of the pressure/temperature relationships provided to the operator to assist in evaluating abnormal plant conditions. We will also request the licensee to identify the procedures and training instruction which were reviewed and modified.

Item 5

The licensee's response indicates that review of all safety related systems resulted in the establishment of procedures to restrict and control the manipulation, testing, maintenance and the independent verification of the line up of valves. We will request the licensee to identify safety related systems under this administrative control. Also, we will request the licensee to discuss their procedures to assure the operability of other safety related components and systems such as pumps, electrical power, and electrical instrumentation and controls. We will request the licensee to indicate the frequency of checks to insure that locked-open valves in the auxiliary feedwater system are in their correct positions.

Item 6

The licensee states that containment isolation at Davis-Besse-1 occurs at either 1600 psig reactor coolant pressure or 4 psig reactor building pressure. The licensee has proposed to override isolation for RCP seal injection and seal return. The licensee has also proposed to override isolation for reactor coolant system (RCS) makeup. This would allow the RC makeup pumps to continue supplying water to RCS after HPI pumps have been

started at RCS pressures less than 1600 psig. We find both the above acceptable. The licensee has further requested amendment to his license which would allow isolation of the RCP seal injection and seal return and the RCS makeup at an RCS pressure of 400 psig. This request is still under review. We will request the licensee to clarify whether overriding the valve closing signal will affect all valves or only the individual valves in the systems indicated above. We will request the licensee to clarify whether both containment isolation and HPI initiation comes from the same signal. We will request the licensee to describe their depth of review of containment isolation including all procedures affecting containment isolation of manual and remote manual valves, and to identify all lines not required for safety that are not isolated on containment isolation.

Item 7

We have reviewed the licensee's response and find that he has adequately addressed the item as stated in the Bulletin. We will request the licensee to identify all procedures which have been prepared or modified and his schedules for implementation of the procedures. We will also request the licensee to identify any pneumatically operated valves in the auxiliary feedwater system and include them in such procedures which assure their proper lineup.

Item 8

The licensee has stated that when at least one 100% capacity flow path is not available the reactor will be in Mode 3 within one hour and in Mode 4 within 12 hours or the reactor will be shutdown at the maximum safe rate. We find this acceptable and will request the licensee to confirm that Mode 4 operation does not rely on steam generators for cooling.

Item 9

This item was modified by IE Bulletin 79-06A. The licensee will be requested to complete his response by responding to Item 9c of IE Bulletin 79-06A. The licensee will also be requested to identify the operating modes and procedures which were reviewed.

The subject of the containment isolation valves being opened to allow purging during operation is presently under staff review. In our letter of November 29, 1978, we requested that the licensee provide a justification for continued purging at his facility and to limit purging to an absolute minimum, not to exceed 90 hours per year, pending the NRC staff review of his justification. The licensee provided a justification for purging in his letter of December 13, 1978, and his response is under review.

Item 10

The licensee has stated that procedures are being modified to require the shift foreman to provide the verification identified in Item 10. We will request the licensee to list the safety systems which were reviewed. We will request the licensee to identify the procedures reviewed and the procedures which are being modified and the implementation schedule for these modifications. We will request a description of how notification of reactor operating personnel and the Shift Foreman will be accomplished. We will request from the licensee a list of the Safety Related Systems covered under the status notification procedure. We will request the licensee to adopt the position that,not withstanding prior operability verification within the current surveillance interval,operability should be further verified at least by inspection immediately prior to removing the redundant unit from service. We will also request the licensee to confirm that procedures explicitly provide for the returning of a safety related system to its normal operating configurations following maintenance. Also, we will request the licensee to state whether any confirmatory checks exist in the verifications of the configurations.

Item 11

The licensee has indicated that operation, maintenance and instrumentation and control personnel have been briefed on the TMI-2 event. We will request the licensee to include appropriate supervisory and management

personnel in such training and to provide a schedule for completion of the training for all operations, maintenance and appropriate supervisory and management personnel.

Item 12

The response from the licensee discusses procedural controls for NRC notification. We note that no reference is made by the licensee to the guidance and requirements established in Regulatory Guide 1.16 "Reporting of Operating Information - Appendix A Technical Specifications," Regulatory Guide 1.101, "Emergency Planning for Nuclear Power Plants," and 10 CFR 20. We will request that the licensee clarify his response as to the incorporation of the requirements and guidance provided by the regulations and regulatory guides.

However, we believe additional emphasis should be placed by the licensee on the concern addressed in the Bulletin regarding very early notification of the NRC of serious events. We will request that the licensee prepare and implement immediately necessary actions and procedures to assure than an open, continuous communication channel can be established and maintained with NRC and that the NRC is notified within one hour of the time the reactor is not in a controlled or expected condition of operation.

CONCLUSION

Based on our review of the information provided by the licensee to date, we conclude that, while we have identified certain areas where additional information or action is needed, the licensee has correctly interpreted IE Bulletin No. 79-05A. The actions taken demonstrate his understanding of the salient concerns arising from the Three Mile Island incident in reviewing their implications on his own operations, and provide added assurance for the protection of the public health and safety during plant operation.

Technical Specifications should be reviewed to determine necessary changes because of the requirements of the Bulletin. We will request that the licensee propose changes, as required, to those Technical Specifications which must be modified as a result of implementing any revised procedures resulting from Bulletin 79-05A.

DAVIS-BESSE NUCLEAR POWER STATION - UNIT I

DOCKET NO. 50-346

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RESPONSE TO IE BULLETIN 79-05A

ADDITIONAL ACTIONS REQUESTED OF THE LICENSEE

1. Ensure that all appropriate levels of utility personnel have participated in the review required by Item 1 of IE Bulletin 79-05A. All licensed operators, plant managers, and supervisors with operational responsibilities should participate in this review, and such participation should be promptly concluded and documented in plant records.

2. Identify and provide a schedule of implementation of the procedures which provide for the verification of auxiliary feedwater system valve positioning.

3. Prepare, and implement immediately, procedures which:
   a. Identify those plant indications (such as valve discharge piping temperature, valve position indication, or valve discharge relief tank temperature, pressure, or level indication) which plant operators may utilize to determine that pressurizer power operated relief valve(s) are open; and

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b. Direct the plant operators to manually close the power operated relief block valve(s) when reactor coolant system pressure is reduced to below the set point for normal automatic closure of the power operated relief valve(s) and the valve(s) remain stuck open.

4. In all cases where reference has been made to a review or revision of existing procedures, or a preparation of new procedures, provide a list of the procedures by number and title. If additional aids are provided to the operator (e.g., reactor coolant system pressure/temperature curves), these should be fully described, and the appropriate procedures supporting the use of these aids should be referenced.

5. Review operating modes and procedures that deal with significant amounts of hydrogen as that may be generated during a transient or other accident that would either remain inside the primary system or be released to the containment.

6. Identify any procedures that exist to guide the operator response to any of the various modes of failure in the Integrated Control System (ICS).

7. Provide significant reduction in the 120 day period allowed for operator training.

8. Identify presently anticipated criteria under which engineering safety features would be shutdown or restarted to prevent an unsafe plant condition.

9. Provide a commitment that the operators do not override automatic actions of engineered safety features, unless continued operation of engineered safety features will result in unsafe plant conditions.

10. In the response to Item 4b of IE Bulletin 79-05A you have indicated that HPI flow will be throttled whenever total flow exceeds 900 gpm. Provide a basis for this action.

11. In response to Item 4a and 5 provide a list of all procedures which you have reviewed including appropriate emergency procedures and identify those that were revised as a result of this review. Provide a list of the engineered safety features for your plant.

12. With respect to Item 4c expand your procedures in accordance with Item 7c of Bulletin 79-06A which requires the operation of RCPs as long as the pumps are providing forced flow. Identify the parameters that indicate forced flow.

13. Provide details concerning the nature of the direction given to operators to utilize pressure/temperature relationships that will assist them in evaluating abnormal plant conditions. Also, describe the form of the pressure/temperature relationships that are available to the operator in the control room in this regard.

14. Identify the procedures and training instructions which were reviewed and modified in relationship to Item 4d.

15. With respect to Item 5, indicate the frequency of checks to insure that locked open valves in the auxiliary feedwater systems are in their correct positions.

16. Propose changes, as necessary, to those technical specifications which must be modified as a result of implementing any revised procedures resulting from IE Bulletin 79-05A.

17. Provide clarification whether overriding the SFAS close signal will affect all valves or cnly the individual valves in the overriding of the RCS make up and the RCP seal injection and seal return.

18. Describe your depth of review of containment isolation including all procedures affecting containment isolation of manual and remote manual valves and to identify all lines not required for safety that are not isolated on containment isolation.

19. Indicate whether Davis-Besse-1 has pneumatically operated valves in the auxiliary feedwater system and if it does, include them in such procedures which assure the proper line up of valves.

20. Confirm that Mode 4 operation does not rely on steam generators for cooling with respect to Item 8.

21. With respect to Item 9 of the IE Bulletin 79-05A complete the response by responding to Item 9c of IE Bulletin 79-06A. Also, with respect to Item 9, identify the operating modes and procedures which were reviewed.

22. With respect to Item 10 provide the following:

a. List of systems which were reviewed.

b. List of procedures which were reviewed and a list of procedures which are being modified or have been modified as a result of the review.

c. Schedule for implementing new procedures or modified procedures.

d. Description of how notification of operating personnel and Shift Foreman will be accomplished.

e. List of safety related procedures covered under the statute notification procedures.

f. Adoption of the position that,notwithstanding prior operability verification within the current surveillance interval,operability should be further verified at least by inspection, immediately prior to removing a redundant

unit from service.

g. Confirmation that procedures explicitly provide for the returning of a safety-related system to its normal operating configuration following maintenance. Also, state whether any redundancies exist in the verification of the configuration (for example, two man checks on valve positions).

23. Prepare and implement necessary actions and procedures to assure that an open, continuous communication channel can be established and maintained with NRC and that the NRC is notified within one hour of the time the reactor is not in a controlled or expected condition of operation. Also, clarify your response as to the incorporation of guidance provided by the regulations and the regulatory guidance.