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

50-313

DRAFT

Docket No. 50-313

Mr. William Cavanaugh, III
Executive Director, Generation
and Construction Department
Arkansas Power and Light Company
P. O. Box 551
Little Rock, Arkansas 72203

ANO-1

Dear Mr. Cavanaugh:

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 submitted within 30 days of receipt of this letter.

Sincerely,

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Enclosures:
1. Evaluation
2. Request for Additional Information

(21)

7907160373

Roisman 79-98 [redacted]

[redacted]

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Enclosure 1

UNITED STATES
NUCLEAR REGULATORY COMMISSION
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EVALUATION OF LICENSEE'S RESPONSES TO IE BULLETIN 79-05A

ARKANSAS POWER AND LIGHT COMPANY

ARKANSAS NUCLEAR ONE - UNIT NO. 1

DOCKET NO. 50-313

Introduction

By letter dated April 1, 1979 we transmitted to Arkansas Power and Light Company (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 letters dated April 11, and 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 licensees responses relative to those additional requirements specified in Bulletin 79-06A which are applicable to Arkansas Nuclear One - Unit No. 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 Arkansas Nuclear One - Unit No. 1 facility (ANO-1). 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; however, it is not clear that all affected personnel have participated in this review therefore we will request the licensee to document participation in this review.

In response to Concern 1 of Item 1, the licensee states that he has reviewed all appropriate Operating Procedures at ANO-1 and has found that adequate and appropriate instructions are given to assure the proper positioning of the Emergency Feedwater (EFW) valves. The licensee further states that in the course of his review, one valve was discovered that was not previously included in the procedure, but which has now been added. In addition he states that he has reviewed the present line up of all EFW valves and the line up that would result from loss of power and has found that all EFW valves were properly lined up and would be properly lined up in the event of loss of power. Based on these actions the licensee states that they have assured themselves of the adequacy of their procedures for preventing

incorrect alignment of EFW valves at ANO-1. A review of these procedures by representatives of the NRC Regional Office identified certain errors/ommisions in the procedures which were communicated to the licensee. Our evaluation relative to this concern is presented in the discussion of Items 5, 7 and 8, below.

In response to Concern 2 of Item 1, regarding failure of the pressurizer electromatic relief valve (ERV) to close upon reduction of pressure below the actuation level, the licensee states that he will revise the Emergency Procedures to require closing of the ERV block valve upon automatic ES (safety injection) actuation.

We believe 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) but the valve(s) remain stuck open.

We will also request the licensee to identify any conditions, other than b, above, when the ERV block valve would be closed.

In the licensee's response to Concern 2 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 a list by number and title of all procedures reviewed, revised or prepared.

Concern 3 of Item 1 relates to possibly erroneous indication of pressurizer level. Our evaluation of the licensee's response to this concern is given in Item 4.a.

Concern 4 of Item 1 relates to the inadvertent transfer of radioactive water from the Reactor Building by automatic initiation of a transfer pump because the Reactor Building was not isolated by high pressure injection (HPI) initiation. Our evaluation of the licensee's response to this concern is given in Item 6 and 9.

Concern 5 of Item 1 of the Bulletin relates to the adverse effect of intermittent operation of the HPI system at TMI-2 to maintain pressurizer level. The licensee states that recent revisions to the operating procedures at ANO-1 as requested by Items 4a and b of the Bulletin will preclude occurrence of a similar event at ANO-1. Our evaluation of this matter is addressed below in our consideration of these items.

Our evaluation of the licensee's response to Concern 6 of Item 1, which relates to premature tripping of the reactor coolant pumps is given in our discussion of Item 4c.

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 transients similar to those referenced, which had occurred at ANO-1, including Loss of Offsite Power, Loss of Feedwater, Turbine Trip, Load Rejection and Reactor Trip. He further stated that based on his

review, ANO-1 performed as expected except for a transient which occurred in December 1974 during startup testing of the facility. During this transient, which involved reactor trip from 100% power, the licensee states that there was a loss of pressurizer level indication for a period between 20 and 40 seconds. He also states, however, that analysis indicates that the pressurizer level dropped only about 8 inches below the zero level reading, so that approximately 96 inches of water actually remained in the pressurizer. The licensee states that this problem occurred before the Integrated Control System (ICS) had been fine-tuned and that subsequent to completion of fine tuning, pressurizer level indication has not been lost on any transient.

An added event, not cited under this item, was the instance of a stuck ERV which also occurred during startup testing in 1974 (cited under Item 1, Concern 2). The licensee states that minor modifications were made to correct the condition and that following modifications, the valve was retested and performed satisfactorily. 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 give added assurance that a TMI-2 accident will not be repeated.

In connection with these transients we will request that the licensee inform us of any procedures that exist at ANO-1 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 4 1/2 months allowed for completion of the training is much too long and that a significant reduction in the time allowed should be made.

Item 4

The licensee's response to this item was framed in terms of Bulletin 79-05A. Subsequent to issuance of 79-05A we also published Bulletin 79-06A, which in Item 7, provided additional clarification of the intent of Item 4 in Bulletin 79-05A. Our evaluation of this Item, therefore, is based on the guidance contained in Item 7 of Bulletin 79-06A.

Item 4a

To the extent that the licensee's commitments conform to the guidance contained in Item 7.a of Bulletin 79-06A, we find the licensee's response acceptable. This guidance makes it clear that engineered safety features (ESF) are not to be overridden unless failure to do so could result in an unsafe plant condition

To evaluate the licensee's intentions in this regard we will request the licensee to describe the presently anticipated criteria under which an ESF would be shutdown or restarted to prevent an unsafe plant condition. We will also ask him to explain what is meant by a "Caution Note" to be inserted in the ANO-1 Emergency Procedures and to summarize the "clarifying steps" which will be added to procedures to aid operators in identifying spurious actuation. We will further request that he list for this and all other items 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 ESF systems for his plant.

Item 4b

As in Item 4a, above, we find the licensee's response to this item acceptable to the extent that it conforms to Item 7b of Bulletin 79-06A. Specific deviations identified by the licensee are as follows:

1) In paragraph 4.b.(1), the deletion of the requirement that two low pressure injection pumps are running, and

2) The addition of paragraph 4.b.(3) without including a requirement that the specified conditions prevail in a stable condition for at least 20 minutes.

However, we will request the licensee to conform to all provisions of paragraph 7b of Bulletin 79-06A.

In addition, as regards paragraph 4.b.(2) of this response we will expect the licensee to observe the clarifying guidance relative to vessel integrity provided in Item 7.b.(2) of Bulletin 79-06A.

Thus, we find the licensee's response acceptable only insofar as it provides strict adherence to the provisions of Item 7.b of Bulletin 79-06A.

Item 4.c

Subject to the clarification provided in Item 7.c of Bulletin 79-06A regarding forced flow operation, we find the licensee's response basically acceptable. We will request the licensee to prepare his

procedures in accordance with Item 7c of Bulletin 79-06A which requires operation of reactor coolant pumps (RCPs) as long as the pump(s) is providing forced flow and to identify the parameters that indicate forced flow. However, the licensee proposed to trip the reactor coolant pumps (RCPs) under certain conditions, when according to the licensee, continued operation would worsen the conditions (e.g., Main Steam Line Break). We will request the licensee to identify the events and conditions under which he proposes to trip the RCPs. We will request that he identify the instrument indications available to the operator to unambiguously define the occurrence of these events.

Item 4.d

The licensee's response is acceptable.

Item 5

The licensee states that he has reviewed all safety-related valves, their positioning requirements and "fail positions" for ten plant systems. He also states that he has reviewed the related procedures and concludes that they are adequate to assure the proper positioning of safety-related valves. While the controls described by the licensee to assure proper valve positioning following maintenance and testing operations appear to be generally acceptable, we will request the licensee to provide a

list of the engineered safety features reviewed and identify by number and title, the operating, emergency, maintenance and testing procedures reviewed and/or modified. We will also request the licensee to indicate the extent to which independent verification is utilized to assure correct positioning.

Item 6

The licensee states that containment isolation occurs on either a High Reactor Building Pressure ($\geq$ 4 psig) or an a Low Reactor Coolant System Pressure ($\leq$ 1500 psig). He also identifies which fluid lines are isolated on each of these signals. In order to provide a greater margin of safety the licensee is proposing to increase the number of lines isolated on Low Reactor Coolant System Pressure (coincident with HPI initiation). Upon reviewing the specific lines involved, we find this proposal, including the schedule for implementation, acceptable. However, we will request the licensee to provide the following additional information:

1) Justification for not isolating the Intermediate Cooling Water to the Control Rod Drives.

2) Identify the procedures which were revised to assure that items 9 through 16 in the licensee's list are verified closed following HPI initiation pending implementation of the modification.

3) Assuming the generic small break LOCA analysis is based on a 1600 psig setpoint for HPI and LPI initiation, justify your selection of a 1500 psig setpoint.

Item 7

We have reviewed the licensee's response and find that he has adequately addressed the item as stated in the Bulletin. However, we will request that he revise his maintenance and surveillance procedures as necessary to assure the operability of the two pneumatically operated valves in the auxiliary feedwater system, and that he provide information concerning the position indication provided for these valves. We will also request that the licensee indicate his commitment to verify the correct positioning of the manual valves addressed in this item which are locked, sealed or otherwise secured.

Item 8

In the licensee's response to this item in his letter of April 16, 1979, he cites various surveillance procedures designed to assure that two independent auxiliary feedwater paths, each with 100% capacity, are operable at any time required to meet the requirements of his Technical Specifications. We note, however, that normally open cross-over valves are present in the cross-over line between the two auxiliary feedwater pump discharge lines.

Thus, with these valves open, in the event of a single passive failure, such as a pipe rupture in one of the discharge lines, both feedwater paths could be rendered inoperable. The open valves are necessary for the present ANO-1 design, however, because only the turbine driven emergency feedwater pump is started automatically, and thus the open crossover line provides the means for serving both steam generators. Redundant valving with independent power supplies is provided to protect against a single failure of either a valve or a power source. We conclude that this arrangement provides the necessary independence, even though operator action would be required to shut the valves in the event of a break in one of the auxiliary feedwater pump discharge lines.

We consider this aspect of the response to this item to be acceptable. We will, however, request that the licensee provide information regarding the indications available to the control room operator that would signal the requirement to shut the crossover valves, and cite the applicable operating procedures controlling his actions. We will also require that the licensee prepare and implement prior to startup from the current refueling outage, procedures which require the stationing of an individual (with no other assigned concurrent duties and in direct and continuous communication with the control room) to promptly initiate the non-automatic source of emergency feedwater to the steam generator(s) for those transients or accidents the consequences of which can be limited by such action.

The licensee's present Technical Specifications governing the operability requirements for the emergency feedwater trains do not conform to the position stated in the Bulletin. To remedy this condition, the licensee states that he is modifying his plant procedures such that he will be in conformance with the bulletin prior to operation following the current refueling outage. He also describes the procedures and states that by May 16, 1979, he will submit proposed changes to the Technical Specifications which conform to these revised procedures.

Based on our review of the revised procedures we find that while they are generally in conformance with the Bulletin guidance, he has certain exceptions, including:

1) In addressing the case of one flow path being inoperable, the licensee's use of the term "cold shutdown" does not make it clear that he is not relying on the steam generator for cooling.
2) Also, in the above case, the licensee does not conform to the Bulletin position that the cooldown shall be accomplished in 12 hours because he qualifies his commitment by stating "12 hours or at the maximum safe rate."

3) The licensee's procedure states that if both of the emergency feedwater pumps and the auxiliary feedwater pump are inoperable, the reactor will not be shutdown but rather, power will be reduced to less than 5% using the main feedwater pumps and maintained there until one of the emergency feedwater pumps or the auxiliary feedwater pumps is made operable. Following that, the reactor will be placed in cold shutdown within 12 hours or at the maximum safe rate.

We conclude that the licensee should conform to the provisions of the Bulletin as stated until he has provided sufficient justification for these proposed deviations from the Bulletin guidance.

Item 9

The licensee responded to this item by stating that the isolation valves in all systems designed to transfer potentially radioactive gases and liquids out of the Reactor Building are normally closed and also receive a close signal upon attainment of a high reactor building pressure (4 psig). To provide an additional margin of safety, however, the licensee is preparing to provide an additional close signal when a Low Reactor Coolant System Pressure condition is reached. In addition,

he states that resetting of the Engineered Safeguards channel will not result in the automatic opening of any closed valve.

We find that this response acceptably addresses Item 9 of the bulletin with the exception that, consistent with Item 9.c of Bulletin 79-06A, we will request the licensee to provide the basis upon which continued operability is assured for these systems.

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 January 30, 1979, and his response is under review.

Item 10

The licensee states that Plant Quality Control procedures require that a Job Order be issued for any maintenance of safety-related ("Q") systems. He also states that the Job Order form is currently being revised to specifically identify all Pre-maintenance and Post-maintenace requirements, including verification that the requirements are met, prior to declaring a system or component "operable" after maintenance. We find

that additional information is needed before we can complete our evaluation of this item. This information includes: a list of the safety-related ("Q") systems, a list of the procedures which were reviewed or modified relative to this item, a description of the procedure used for explicitly notifying operations personnel concerning the plant status and a description of the method of verifying the operability of systems prior to and following maintenance. In addition, we will request the licensee to apply for a revision of the facility technical specification which will require verification of the operability of redundant components when removed from service for testing as well as for maintenance.

Item 11

The licensee states that training sessions on the Three Mile Island Unit No. 2 incident will be completed for all active operations and maintenance personnel during the current refueling outage. He states that this training emphasizes the seriousness and consequences of simultaneous blocking of both emergency feedwater trains and discusses the sequence of events and the plant parameter curves generated during the incident. It is not clear, however, that the licensee's program has been sufficiently broadbased with respect to the personnel involved in the training sessions. Therefore, to assure that all appropriate

levels of utility personnel have participated in this training, we will request a commitment from the licensee that all licensed operators and plant management and supervisors with operational responsibilities participate in this training and that such participation be promptly concluded and documented in plant records.

Item 12

The response from the licensee outlines the procedural controls that have been established for NRC notification of serious events. We note that although reference is made by the licensee to the guidance and requirements established in Regulatory Guide 1.101, "Emergency Planning for Nuclear Power Plants," and 10 CFR 20, no mention is made specifically of following the guidance of Regulatory Guide 1.101 and 1.16. We will request that the licensee clarify his response as to the incorporation of the guidance provided by R. G. 1.101 and 1.16.

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 that an open, continuous communication channel can be established and maintained with NRC and that the NRC is notified within our 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.

Enclosure No. 2

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ARKANSAS NUCLEAR ONE - UNIT NO. 1
DOCKET NO. 50-313
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. 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.

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3. 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.

4. Review operating modes and procedures that 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.

5. Provide a significant reduction in the 4 1/2 month period allowed for operator training.

6. Verify that all procedures, including appropriate emergency procedures, have been reviewed with respect to continued operation of any engineered safety features system if it is automatically initiated. Provide a list of all engineered safety feature systems included in this review, and describe the presently anticipated criteria under which operation of engineered safety features would be terminated or restarted to prevent unsafe plant conditions.

7. Review all procedures used for maintenance or testing to ensure that all valves involved in engineered safety feature systems are returned to their correct positions following necessary manipulations.

8. 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.

9. Provide a discussion of the safety implications of not isolating the cooling lines to the reactor coolant pumps and control rods during containment isolation.

10. Revise maintenance and surveillance procedures as necessary to ensure the operability of the two pneumatically operated valves in the auxiliary feedwater system. Describe the position indication provided for these valves.

11. Review procedures to ensure that the operability of a redundant safety-related system, not just the redundant component, is verified prior to removing a safety-related system from service.

12. Indicate the extent to which independent verification is utilized to assure correct positioning of safety-related valves.

13. Revise your procedures as necessary to ensure that all reactor operating personnel are explicitly notified wheneve a safety-related system is removed from or returned to service.

14. Provide information regarding the indications available to the control room operator that would signal the requirements to shut the crossover valves in the auxiliary feedwater system and cite the applicable operating procedures controlling his actions.

15. Clarify whether or not systems of transferring potentially radioactive gases or liquids from the containment are interlocked with a high radiation signal and provide the basis upon which the continued operability of the interlocks and isolation signals are assured.

16. Identify any conditions other than HPI initiation when you would close the ERV block valve following opening of the ERV.

17. With respect to Item 4.a, explain what is meant by a "caution note" to be inserted in the ANO-1 Emergency Procedures and summarize the "clarifying steps" which will be added to procedures to aid operators in identifying spurious actuation.

18. Expand your procedures in accordance with Item 7c of Bulletin 79-06A which specifies the operation of RCPs as long as pumps are providing forced flow. Identify the parameters that indicate forced flow. Also identify the events and conditions under which you proposed to trip the RCPs.

19. Relative to Item 6, provide the following information:

    1) Identify the procedures which were revised to assure that items 9 through 16 in the list are verified closed following HIP initiation, pending implementation of the modification which will provide automatic closure.

    2) Assuming the small break LOCA analysis for ANO-1 is based on a 1600 psig setpoint for HPI and LPI initiation, justify your selection of a 1500 psig setpoint.

20, With respect to Item 7, indicate your frequency of checks of the correct positioning of the manual valves addressed in this item which are locked, sealed or otherwise secured.

21. With respect to Item 8, indicate your commitment to prepare and implement prior to startup from the current refueling outage, procedures which require the stationing of an individual (with no other assigned concurrent duties and who is in direct and continuous communication with the control room) to promptly initiate the non-automatic source of emergency feedwater to the steam generator(s) for those transients or accidents the consequences of which can be limited by such action.

22. Although we require you to meet the provisions of Item 7c of Bulletin 79-06, identify the other events and conditions under which you believe the RCPs should be tripped. Also identify the instrument indications available to the operator to unambiguously define the condition or event.

23. With respect to Item 10, provide the following information and commitments:

    1) Provide a list of the safety related systems reviewed.

    2) Identify the procedures which were reviewed or modified in connection with responding to this item.

3) Describe the procedure used for explicitly notifying involved reactor operating personnel whenever safety-related systems are removed from or returned to service.

4) Briefly describe the procedures to be used to verify operability prior to and following removal of safety-related systems, including the emergency feedwater system, from service for maintenance or testing.

24. Identify any procedures that exist to guide the operator response to any of the various modes of failure in the Integrated Control System.

25. Indicate your commitment to prepare and implement immediately necessary actions and procedures to assure than an open, continuous communication channel can be established and maintained with the 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.

26. Clarify the response to Item 12 of Bulletin 79-05A as to the incorporation of the guidance provided by Regulatory Guide 1.101 and 1.16.