TERA

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

June 13, 1980

Docket No. 50-313

Mr. William Cavanaugh, III
Vice President, Generation
and Construction
Arkansas Power & Light Company
P. O. Box 551
Little Rock, Arkansas 72203

Dear Mr. Cavanaugh:

We have reviewed the information provided by your letters dated April 11 and 16, and May 4 and 21, 1979, in response to IE Bulletins 79-05A and 79-05B for Arkansas Nuclear One, Unit No. 1. The enclosure provides our evaluation of your responses with respect to their specificity, completeness, and responsiveness to the intent of said bulletins. In this regard, except for the issues described below, we have found that you have taken the appropriate actions to meet the requirements of IE Bulletins 79-05A and 79-05B.

In your response to Item 8 of IE Bulletin 79-05A, you have taken certain exceptions including:

1) In addressing the case of one auxiliary feedwater (AFW) flow path being inoperable, your use of the term "cold shutdown" does not make it clear that you are relying on the steam generator for cooling.

2) Also, in the above case, you do not conform to the Bulletin position that cooldown shall be accomplished in 12 hours because you qualify your commitment by saying "12 hours or at a maximum safe rate."

3) Your procedures state that if both EFW and the AFW 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 in this condition until one of the EFW pumps or the AFW pump is made operable. Following that, the reactor will be placed in cold shutdown within 12 hours or at the maximum rate.

We request that you commit to conform to the provisions of the Bulletin as stated or provide sufficient justification for these proposed deviations from the Bulletin guidance.

With regard to Item 10 of IE Bulletin 79-05A, we request that you propose Technical Specifications which would require verification of the operability of redundant components when removed from service for testing and maintenance.

8006300319 Q

With regards to Item 4 of IE Bulletin 79-05B, we request you to provide further and sufficient justification for not providing a manual trip on loss of offsite power or conform to the original request of Item 4.

It should be noted that the results of reviews of separate issues resulting from the TMI-2 accident are provided in NUREG-0560, "Staff Report on the Generic Assessment of Feedwater Transients in Pressurized Water Reactors Designed by the Babcock and Wilcox Company", NUREG-0565, "Generic Evaluation of Small Break Loss of Coolant Accident Behavior in Babcock and Wilcox Designed 177 FA Operating Plants", and NUREG-0623, "Generic Assessment of Delayed Reactor Coolant Pump Trip During Small Break Loss of Coolant Accidents in Pressurized Water Reactors". These reports provide additional recommendations, some of which have been implemented and others which may eventually be implemented as part of the NRC Action Plans developed as a result of the TMI-2 accident.

Sincerely,

Thomas M. Novak, Assistant Director
for Operating Reactors
Division of Licensing

Enclosure: Evaluation of
Licensee's Responses to
IE Bulletins 79-05A and
79-05B

cc w/enclosure:
See next page

Arkansas Power & Light Company

cc w/enclosure(s):

Mr. David C. Trimble
Manager, Licensing
Arkansas Power & Light Company
P. O. Box 551
Little Rock, Arkansas 72203

Mr. James P. O'Hanlon
General Manager
Arkansas Nuclear One
P. O. Box 608
Russellville, Arkansas 72801

Mr. William Johnson
U. S. Nuclear Regulatory Commission
P. O. Box 2090
Russellville, Arkansas 72801

Mr. Robert B. Borsum
Babcock & Wilcox
Nuclear Power Generation Division
Suite 420, 7735 Old Georgetown Road
Bethesda, Maryland 20014

Troy B. Conner, Jr., Esq.
Conner, Moore & Corber
1747 Pennsylvania Avenue, N.W.
Washington, D.C. 20006

Arkansas Polytechnic College
Russellville, Arkansas 72801

Honorable Ermil Grant
Acting County Judge of Pope County
Pope County Courthouse
Russellville, Arkansas 72801

Director, Technical Assessment
Division
Office of Radiation Programs
(AW-459)
U. S. Environmental Protection Agency
Crystal Mall #2
Arlington, Virginia 20460

U. S. Environmental Protection Agency
Region VI Office
ATTN: EIS COORDINATOR
1201 Elm Street
First International Building
Dallas, Texas 75270

Director, Bureau of Environmental
Health Services
4815 West Markham Street
Little Rock, Arkansas 72201

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

EVALUATION OF LICENSEE'S RESPONSES

TO

IE BULLETINS 79-05A AND 79-05B

ARKANSAS POWER AND LIGHT COMPANY

ARKANSAS NUCLEAR ONE UNIT NO. 1

DOCKET NO. 50-313

Introduction

On March 29, 1979 the Three Mile Island Nuclear Power Plant, Unit 2 (TMI-2) experienced core damage which resulted from a series of events which were initiated by a loss of feedwater transient. Several aspects of the accident have generic applicability at operating Babcock and Wilcox (B&W) reactors. On April 1, 1979, IE Bulletin 79-05 was sent to all B&W operating plant licensees. The purpose of the bulletin was to provide information concerning the accident at TMI-2 and to request certain actions be taken by licensees to preclude a similar occurrence at their facilities. This bulletin was superseded and expanded by IE Bulletin 79-05A dated April 5, 1979, and by IE Bulletin 79-05B dated April 21, 1979. By letters dated April 11 and 16, and May 4 and 21, 1979, the Arkansas Power and Light Company (AP&L or the licensee) provided responses in conformance with the requirements of the bulletins (References 1, 2, 3 & 4).

Information became available to the NRC, subsequent to the issuance of IE Bulletin 79-05B, which required modification to Item 4.c of IE Bulletin 79-05A. On July 26, 1979, IE Bulletin 79-05C was issued superseding Item 4.c of IE Bulletin 79-05A. A separate evaluation of responses to IE Bulletin 79-05C is provided in NUREG-0623, "Generic Assessment of Delayed Reactor Coolant Pump Trip During Small Break LOCA in Pressurized Water Reactors".

Subsequent to the issuance of IE BUlletins 79-05, 79-05A and 79-05B, the Commission issued an Order dated May 17, 1979, which confirmed the licensee's commitment to make certain modifications to plant equipment and procedures, and to complete specific operator training and analyses of plant behavior prior to restart of the facility. Due to the overlap in the requirements of the Order and the bulletins, the Order is referenced several times in this evaluation. In addition, the NRC's Lessons Learned Task Force completed its report in July 1979 (NUREG-0578) detailing short-term recommendations that are to be implemented for all operating reactor plants in light of the accident at TMI-2. This report has also been referenced in this evaluation. The NRC staff's evaluation of the licensee's compliance with the short-term portion of the Commission Order was issued on May 31, 1979 (Reference 6). Separate evaluations of the licensee's compliance with the long-term portion of the Commission Order and NUREG-0578 will be issued at a future date.

As a result of a discovery by the NRC resident inspector on June 2, 1979, that the Emergency Feedwater (EFW) system was locked out of automatic initiation mode, the licensee was required by a Commission Order dated June 2, 1979, to remain in cold shutdown until certain actions relating to the procedures for all modes of plant operation were satisfactorily accomplished. Having satisfactorily completed the required actions, the licensee was issued the Commission Order Authorizing Resumption of Operation dated June 14, 1979.

The NRC staff's evaluation of the licensee's responses to IE Bulletins 79-05A and 79-05B is provided below. Certain items in this evaluation will require further staff review during its evaluation of the licensee's compliance with the long-term portion of the Commission Order and the licensee's implementation of NUREG-0578. Where applicable, these issues are discussed under the appropriate bulletin item and a summary of the outstanding issues is provided at the end of this evaluation.

Evaluation of Responses to IE Bulletin 79-05A

Item 1 "In addition to the review of circumstances described in Enclosure 1 of IE Bulletin 79-05, review the enclosed preliminary chronology of the TMI-2, 3/28/79, accident. This review should be directed toward understanding the sequence of events to ensure against such an accident at your facility(ies)."

The licensee has reviewed Enclosure 1 to IE Bulletin 79-05 and the preliminary sequence of events enclosed with IE Bulletin 79-05A. The licensee, assisted by B&W, assessed the adequacy of Arkansas Nuclear One, Unit No. 1 (ANO-1) to safely sustain transients such as the one which occurred at TMI-2. Its review identified the same six human, design and mechanical failures which resulted in the core damage and radiation releases at TMI-2, as are described in the "Description of Circumstances" portion of IE Bulletin 79-05A. Details of this review are documented in Reference 1 to this evaluation. The staff has reviewed this document and finds the licensee has a satisfactory understanding of the sequence of events. Specific staff comments concerning the licensee's responses are provided below.

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 EFW valves. The licensee further stated 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 position and failure mode of all EFW valves and found that all EFW valves were in their proper positions with the proper failure mode. 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 additional errors/omissions in the procedures which were communicated to the licensee. Our evaluation relative to this concern is presented in the discussion of Items 5, 7 & 8 below.

The description of circumstances in IE Bulletin 79-05A states that the pressurizer electromatic relief valve (PORV), which opened during the initial pressure surge, at TMI-2, failed to close when the pressure decreased below the actuation level. In Reference 1, the licensee stated that the Emergency Procedures would be revised to require closing of the PORV block valve upon automatic ES (safety injection) actuation. We believe that procedures should relate to the status of the PORV and therefore a direct means of monitoring the PORV position should be available.

In response to Item 2.1.3.a of NUREG-0578 the licensee has installed an acoustical monitoring system to monitor the position of the PORV and safety valves. This item is discussed in our Evaluation of the Licensee's Compliance with Category "A" Items of NRC Recommendations Resulting from TMI-2 Lessons Learned dated March 10, 1980. (NRC letter to AP&L dated March 10, 1980.)

The description of circumstances in IE Bulletin 79-05A states that because the containment did not isolate on high pressure injection (HPI) initiation, the highly radioactive water from the PORV discharge was pumped out of the containment by the automatic initiation of a transfer pump. The licensee stated in Reference 1 that at ANO-1 the line from the Reactor Building sump to the Auxiliary Building sump is a normally closed gravity flow line which requires manual operation. The two isolation valves receive a Reactor Building isolation signal. Item 6 of this Bulletin requires the licensee to review containment isolation initiation design and to implement changes necessary to cause containment isolation of all lines whose isolation does not degrade core cooling capability upon automatic initiation of safety injection. The licensee's response to this item also addressed isolation of the Reactor Building Sump Line. Item 9 of this Bulletin requires that the licensee review operating modes and procedures to further assure that undesired pumping of radioactive liquids and gases out of the containment will not occur inadvertently. The issue of containment isolation is addressed in our evaluations of the responses to Items 6 & 9.

The description of circumstances in IE Bulletin 79-05A discusses the adverse effect of intermittent operation of the HPI system at TMI-2. The licensee stated in Reference 1 that recent revisions to the operating procedures at ANO-1, as required by Items 4a, b and d of IE Bulletin 79-05A, will preclude occurrence of a similar event at ANO-1. This matter is addressed in our evaluation of the responses to Item 4.

The description of circumstances in IE Bulletin 79-05A states that tripping of the reactor coolant pumps (RCPs) during the course of the transient, to protect against RCP damage due to vibration, led to fuel damage since the core was uncovered and voids in the reactor coolant system (RCS) prevented natural circulation. As discussed in References 1 and 2, the licensee would modify its procedures to assure at least one RCP per loop remained operating during a loss of reactor coolant/loss of RCS pressure transient, in accordance with NRC guidance in Item 4.c of IE Bulletin 79-05A. As discussed in the introduction to this evaluation, this requirement has been modified and superseded by the requirements of IE Bulletin 79-05C. This issue is also discussed under Item 4.c of IE Bulletin 79-05A.

The NRC staff finds that the licensee has been responsive to Item 1 of IE Bulletin 79-05A and that follow-up action on direct PORV position indications was handled under Section 2.1.3a of NUREG-0578. Therefore, the NRC staff considers the licensee's response to this item complete.

<u>Item 2</u> "Review any transients similar to the Davis-Besse event (Enclosure 2 of IE Bulletin 79-05) and any others which contain similar elements from the enclosed chronology (Enclosure 1) which have occurred at your facility(ies). If any significant deviations from expected performance are identified in your review, provide details and an analysis of the safety significance together with a description of any corrective actions taken. Reference may be made to previous information provided to the NRC, if appropriate, in responding to this item."

In response to Item 2 of the Bulletin, the licensee stated that transients similar to those referenced, which had occurred at ANO-1, had been reviewed. These include Loss of Offsite Power, Loss of Feedwater, Turbine Trip, Load Rejection and Reactor Trip. The licensee further stated that based on this 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. The licensee 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-open PORV 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 particular, the Commission's Order of May 17, 1979 required the licensee to submit a failure modes and effects analysis of the ICS. This report was submitted on August 17, 1979 (Reference 5) and is presently under joint review by the NRC staff and the Oak Ridge National Laboratory.

The NRC staff considers the licensee's response to Item 2 of IE Bulletin 79-05A complete.

Item 3 "Review the actions required by your operating procedures for coping with transients and accidents, with particular attention to:

a. Recognition of the possibility of forming voids in the primary coolant system large enough to compromise the core cooling capability, especially natural circulation capability;

b. operator action required to prevent the formation of such voids; and

c. operator action required to enhance core cooling in the event such voids are formed."

As a result of the TMI-2 accident, all licensed operators have received additional training to enable them to cope with transients and accidents. The specific training received to comply with subparagraphs a, b and c above are documented in Reference 1 to this evaluation. Specific staff comments on this training are provided below.

To accomplish this training, the licensee stated all licensed operators would receive a training program using the B&W simulator, which would simulate the TMI-2 incident. The licensee stated that this training would be completed by September 1, 1979. The staff considered the length of time for completion of this training was unduly protracted. However, simulator training for all licensed operators was required by the Commission's Order of May 17, 1979 and this training was completed for all licensed operators by May 24, 1979. Reference 6 documents the NRC staff's evaluation of this training.

As part of the response to subparagraph c above, the licensee stated that it had modified operating procedures, as required in Item 4.b of IE Bulletin 79-05A, to assure continued operation of at least one RCP per loop in an emergency to assist in core cooling during accident conditions. As discussed previously in this evaluation, IE Bulletin 79-05C requires licensees to immediately trip all operating RCPs in the event of a reactor trip and initiation of HPI caused by RCS low pressure. ANO-1 operators have been instructed in the requirements of IE Bulletin 79-05C.

Additional requirements in the area of emergency procedures for transients and accidents have been recommended in Section 2.1.9 of NUREG-0578. To comply with these requirements, the licensee is actively engaged in developing operator guidelines which cover inadequate core cooling and other abnormal transients.

The schedules for completing these items are also found in NUREG-0578. These requirements greatly expand the actions required by Item 3 of this Bulletin.

The NRC staff considers the licensee's response to Item 3 of IE Bulletin 79-05A complete.

Item 4 "Review the actions directed by the operating procedures and training instructions to ensure that:

a. operators do not override automatic actions of engineered safety features;

b. operating procedures currently, or are revised to, specify that if the HPI system has been automatically actuated because of low pressure condition, it must remain in operation until either;

(1) both low pressure injection (LPI) pumps are in operation and flowing at a rate in excess of 1000 gpm each and the situation has been stable for 20 minutes, or

(2) the HPI system has been in operation for 20 minutes, and all hot and cold leg temperatures are at least 50 degrees below the saturation temperature for the existing RCS pressure. If the 50 degree subcooling cannot be maintained after HPI cutoff, the HPI shall be reactivated;

c. operating procedures currently, or are revised to, specify that in the event of HPI initiation, with RCP operating, at least one RCP per loop shall remain operating.

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."

IE Bulletin 79-05B modified the actions required in subparagraphs a and b above to take into account pressure vessel integrity considerations. Evaluation of this matter is discussed under Item 2 of IE Bulletin 79-05B.

IE Bulletin 79-05C modified the actions required in subparagraph c above. The staff's evaluation of all pressurized water reactor (PWR) licensee's responses to IE Bulletins 79-05C and 79-06C has been documented in NUREG-0623 "Generic Assessment of Delayed Reactor Coolant Pump Trip during Small Break Loss of Coolant Accidents in Pressurized Water Reactors".

In regard to subparagraph "d" above, the licensee has installed two primary coolant saturation meters as recommended by NUREG-0578. These meters continuously display the margin between actual primary coolant temperature and saturation temperature.

We have also reviewed the licensee's procedures for loss of coolant and consider that adequate guidance is given to the operator and that many indications, not just pressurizer level alone, are available to assist the operator in assessing the RCS water inventory. In addition, Section 2.1.3.b of NUREG-0578 requires that licensees upgrade reactor instrumentation to provide the operator with an unambiguous indication of vessel water level and core cooling adequacy.

This issue is still under review. We will complete our review of this issue during the review of the Category "B" items of NUREG-0578.

The NRC staff finds that the licensee has been responsive to Item 4 of IE Bulletin 79-05A and that any further follow-up action on Item 4.c will be handled under the NRC staff review of IE Bulletin 79-05C. Therefore, the NRC staff considers the licensee's response to this item complete.

Item 5 "Verify that EFW valves are in the open position in accordance with Item 8 below. Also, review all safety-related valve positions and positioning requirements to assure that valves are positioned (open or closed) in a manner to ensure the proper operation of engineered safety features. Also review related procedures, such as those for maintenance and testing, to ensure that such valves are returned to their correct positions following necessary maniupulations."

The licensee has documented in Reference 1 to this evaluation that a review has been completed of all safety-related valves, their positioning requirements, their "fail positions" and related procedures. A review of related maintenance and testing procedures was also completed by the licensee. This matter is more fully discussed under Item 10 of this Bulletin.

The NRC staff finds the licensee's response to Item 5 of IE Bulletin 79-05A complete.

Item 6 "Review the containment isolation initiation design and procedures, and prepare and implement all changes necessary to cause containment isolation of all lines whose isolation does not degrade core cooling capability upon automatic initiation of safety injection."

The licensee states that containment isolation occurs on either a high reactor building pressure ($\geq$4 psig) or on a low RCS pressure ($\geq$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 RCS pressure (coincident with HPI initiation). However, those lines to the RCPs, control rod drives, reactor building coolers, letdown cooling, seal water coolers and lube oil coolers would isolate only on high reactor building pressure. The acceptability of not isolating these systems on low RCP pressure was determined in our review and evaluation of the licensee's compliance with Section 2.1.4 of NUREG-0578.

This item was completed under the requirements of Section 2.1.4 of NUREG-0578 and evaluated in our Evaluation of Licensee's Compliance with Category "A" Items of NRC Recommendations Resulting From TMI-2 Lessons Learned (NRC letter dated March 10, 1980). Therefore, the NRC staff considers the licensee's response to this item complete.

Item 7 "For manual valves or manually-operated, motor-driven valves, which could defeat or compromise the flow of Auxiliary Feedwater (AFW) to the Steam Generators (SGs), prepare and implement procedures which:

a. require that such valves be locked in their correct position; or

b. require other similar positive position controls."

The licensee has documented in Reference 2 to this evaluation that procedures have been modified to require that all manual valves in the EFW system that could compromise safety system performance if mispositioned, be locked, sealed or otherwise secured in their correct positions. The licensee has also indicated in Reference 2 of this evaluation that surveillance procedures have been established to verify that valves which could compromise the flow of EFW to the SGs are in their correct position.

Although not specifically addressed in the Bulletin, the staff was concerned about the operation and reliability of the two pneumatically-operated flow control valves in the ANO-1 EFW system. These valves are controlled by the ICS. As part of complying with the immediate actions of the Commission's Order of May 17, 1979, the licensee developed a procedure that allows operator control of the EFW independent of the ICS through the safety-grade bypass valves. This action has resolved the staff's concern about the pneumatically-operated valves.

The NRC staff finds the licensee's response to Item 7 of IE Bulletin 79-05A complete.

Item 8 "Prepare and implement immediately procedures which assure that two independent SG AFW flow paths, each with 100% flow capacity, are operable at any time when heat removal from the primary system is through the SGs. When two independent 100% capacity flow paths are not available, the capacity shall be restored within 72 hours or the plant shall be placed in a cooling mode which does not rely on SGs for cooling within the next 12 hours.

When at least 100% capacity flow path is not available, the reactor shall be made subcritical within one hour and the facility placed in a shutdown cooling mode which does not rely on SGs for cooling within 12 hours or at the maximum safe shutdown rate."

In the licensee's response to this item (Reference 2), various surveillance procedures designed to assure that two independent EFW paths, each with 100% capacity, are operable at any time required to meet the requirements of his Technical Specifications (TSs) were cited. We note, however, that normally open cross-over valves are present in the cross-over line between the two EFW 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 were necessary for the ANO-1 design prior to the TMI-2 event because only the turbine-driven EFW pump was started automatically, and thus the open cross-over line provided the means for serving both SGs.

However, as a requirement of the Order of May 17, 1979, the licensee provided an auto start capability for the motor driven EFW pump. This arrangement assures flow to the SGs in the event only one EFW pump starts even though operator action would be required to shut the valves in the event of a break in one of the EFW pump discharge lines.

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 EFW pump discharge lines.

These valves can be operated from the control room and the operator can verify flow to the SGs by observing flow rate indication, installed as part of the short-term requirements of the Commission Order of May 17, 1979. The staff concurs with the licensee's justification for keeping the cross-tie valves open during normal operation.

The licensee's present TSs governing the operability requirements for the EFW trains do not conform to the position stated in the Bulletin. To remedy this condition, the plant procedures were modified to be in conformance with the Bulletin prior to the startup following the last refueling outage. The licensee also describes the procedures and by letter dated May 16, 1980 (Reference 7), submitted proposed changes to the TSs which conform to these revised procedures.

Based on our review of the revised procedures, we find that they are generally in conformance with the Bulletin guidance while there are 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 the licensee is not relying on the SG 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 the licensee's commitment is qulaified by a statement "12 hours or at the maximum safe rate".

3) The licensee's procedures states that if both of the EFW pumps and the AFW 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 EFW pumps or the AFW 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 sufficient justification has been provided for these proposed deviations from the Bulletin guidance.

Item 9 "Review your operating modes and procedures for all systems designed to transfer potentially radioactive gases and liquids out of the primary containment to assure that undesired pumping of radioactive liquids and gases will not occur inadvertently.

In particular, ensure that such an occurrence would not be caused by the resetting of engineered safety features instrumentation. List all such systems and indicate:

a. whether interlocks exist to prevent transfer when high radiation indication exists; and,

b. whether such systems are isolated by the containment isolation signal."

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 provided an additional close signal when a low RCS pressure condition is reached as part of the requirements of NUREG-0578. In addition, the licensee states that resetting of the engineered safeguards channel will not result in the automatic opening of any closed valve.

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 request for a TS change which would limit purging at power operation to 90 hours per year by letter dated April 2, 1979 (Reference 9). That letter also provided justification for purging. By letter dated December 12, 1979, the licensee committed to maintain ANO-1 purge isolation valves closed whenever the reactor is not in a cold shutdown or refueling mode. This matter is presently under review by the NRC staff.

The NRC staff finds that the licensee has been responsive to Item 9 of IE Bulletin 79-05A and that further resolution of purging during power operation will be handled under the NRR Generic Issues Program. Therefore, the NRC staff considers the licensee's response to this item complete.

Item 10 "Review and modify as necessary your maintenance and test procedures to ensure that they require:

a. verification, by inspection, of the operability of redundant safety-related systems prior to the removal of any safety-related system from service;

b. verification of the operability of all safety-related systems when they are returned to service following maintenance or testing; and,

c. a means of notifying involved reactor personnel whenever a safety-related system is removed from and returned to service."

The licensee's response to Item 10 of this Bulletin is documented in Reference 2 to this evaluation.

The licensee states that plant quality control procedures require that a job order be issued for any maintenance of safety-related ("Q") systems. The licensee also states that the job order form is currently being revised to specifically identify all pre-maintenance and post-maintenance requirements, including verification that the requirements are met, prior to declaring a system or component "operable" after maintenance.

We will request the licensee to apply for a revision of the facility TS which will require verification of the operability of redundant components when removed from service for testing as well as for maintenance.

Further, as a result of the Order dated June 2, 1979, the licensee has evaluated existing procedures to assure that such procedures include all actions necessary for safety.

Except for the request for TS changes which we will request, we find the licensee's response to Item 10 of IE Bulletin 79-05A complete.

Item 11 "All operating and maintenance personnel should be made aware of the extreme seriousness and consequences of the simultaneous blocking of both AFW trains at the TMI-2 plant and other actions taken during the early phases of the accident."

The licensee has documented in Reference 2 to this evaluation that all active operations and maintenance personnel will be trained on the TMI-2 incident. During later discussions with the licensee, we have determined that this training was provided to most all personnel at the plant including all licensed operators at the plant, the maintenance personnel, and the plant management and supervisors with operational and maintenance responsibilities. We find the licensee's response to Item 11 of IE Bulletin 79-05A complete.

Item 12 "Review your prompt reporting procedures for NRC notification to assure very early notification of serious events."

The licensee has documented its responses to this item in Reference 2 to this evaluation. The response from the licensee outlines the procedural controls that have been established for NRC notification of serious events. The licensee states that the policy of NRC notification is not limited to items which are deemed reportable by the TSs. To further clarify the policy, the licensee has modified the administrative control manual and the emergency procedures to include notification to the NRC of items assessed such that they might endanger the health and safety of the public.

The NRC staff finds the licensee's response to Item 12 of IE Bulletin 79-05A complete; however, IE Bulletin 79-05B expands the licensee's responsibility in this area. Further discussion of this matter can be found under the staff's evaluation of Item 6 of IE Bulletin 79-05B.

EVALUATION OF RESPONSES TO IE BULLETIN 79-05B

Item 1 "Develop procedures and train operation personnel on methods of establishing and maintaining natural circulation. The procedures and training must include means of monitoring heat removal efficiency by available plant instrumentation. The procedures must also contain a method of assuring that the primary coolant system is subcooled by at least 50°F before natural circulation is initiated.

In the event that these instructions incorporate anticipatory filling of the OTSG prior to securing the RCPs, a detailed analysis should be done to provide guidance as to the expected system response. The instructions should include the following precautions:

a. maintain pressurizer level sufficient to prevent loss of level indication in the pressurizer;

b. assure availability of adequate capacity of pressurizer heaters, for pressure control and maintain primary system pressure to satisfy the subcooling criterion for natural circulation; and

c. maintain pressure/temperature envelope within Appendix G limits for vessel integrity.

Procedures and training shall also be provided to maintain core cooling in the event both main feedwater and AFW are lost while in the natural circulation mode."

The licensee has documented its response to this item in Reference 3 to the this evaluation. Reference 3 stated that procedures were being revised to provide greater detail in methods of establishing and maintaining natural circulation involving cases of unplanned total loss of forced circulation, anticipated loss of forced circulation and plant cooldown utilizing natural circulation. Procedures had been revised to provide greater detail in methods of establishing and maintaining natural circulation before startup following the last refueling outage.

IE reported in Reference 9 that the licensee's emergency procedures have been revised to include detailed methods for establishing and maintaining natural circulation flow.

The NRC staff finds the licensee's response to Item 1 of IE Bulletin 79-05B complete.

Item 2 "Modify the actions required in Item 4a of IE Bulletin 79-05A to take into account vessel integrity considerations:*

'4. Review the action directed by the operating procedures and training instructions to ensure that:

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

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

(1) Both LPI pumps are in operation and flowing at a rate in excess of 1000** gpm each and the situation had been stable for 20 minutes; or,

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

* [NOTE: Underlined portions are modications to, and supersede, IE Bulletin 79-05A]

** The flow rate at ANO-1 must be greater than 2300 gpm in each line.

The licensee's reply to this item is documented in Reference 3 to this evaluation. This item was revised and covered more completely in the licensee's response (AP&L letter dated December 13, 1979) to IE Bulletin 79-05C, "Tripping of RCP's". Our evaluation of this licensee's response is provided in NUREG-0565, "Generic Evaluation of Small Break Loss of Coolant Accident Behavior in Babcock & Wilcox Designed 177-FA Operating Plants". The staff finds that the licensee has taken the appropriate action to meet the requirements of IE Bulletin 79-05B. This item will be included for complete resolution in the TMI-2 Action Plan now under development.

Item 3 "Following detailed analysis, describe the modifications to design and procedures which you have implemented to assure the reduction of the likelihood of automatic actuation of the pressurizer PORV during anticipated transients. This analysis shall include consideration of a modification of the high pressure scram setpoint and the PORV opening setpoint such that reactor scram will preclude opening of the PORV for the spectrum of anticipated transients discussed by B&W in Enclosure 1. Changes developed by this analysis shall not result in increased frequency of pressurizer safety valve operation for these anticipated transients."

The licensee has documented its response to this item in References 3 and 10 to this evaluation. In these references the licensee has stated that they have reviewed and discussed this item with B&W and they have raised the PORV opening setpoint to 2450 psig and lowered the scram setpoint to 2300 psig. Reference 10 is a request for amendment to the ANO-1 operating license which would require the above setpoint to be in the TSs. This request is under staff review.

Verification of the setpoint changes were completed by IE and reported in Reference 9.

Item 4 "Provide procedures and training to operating personnel for a prompt manual trip of the reactor for transients that result in a pressure increase in the RCS. These transients include:

a. loss of main feedwater;

b. turbine trip;

c. main steam isolation valve closure;

d. loss of offsite power;

e. low OTSG level; and,

f. low pressurizer level."

The licensee has documented its response to this item in Reference 3 to this evaluation. The licensee stated that procedures and operator training has been provided to require a prompt manual trip of reactor for: (1) loss of main feedwater, (2) turbine trip, (3) main steam isolation valve closure, and (4) low OTSG level. We find this acceptable. However, the licensee has not implemented a manual reactor trip on loss of offsite power since the plant is designed to runback to approximately 15% power and supply internal house loads via the main generator through the auxiliary transformer. Experience has shown that to runback to approximately 15% power during a loss of offsite power is difficult to achieve.

Therefore, we will request the licensee to provide further and sufficient justification for not providing a manual reactor trip on loss of offsite power or conform to the original request of Item 4.

Also, the licensee has not implemented a reactor trip on low pressurizer level. The licensee states that such a procedure would be inconsistent with the general concern of an overpressurization event, as the low pressurizer level is associated with underpressurization rather than overpressurization. Further, the licensee has stated that a reactor trip in an underpressurization (overcooling) event only adds to the problem as it removes the primary heat source. Current procedures require increasing reactor coolant makeup flow on decreasing pressurizer level, isolating letdown flow, and increasing seal injection flow in order to return pressurizer level to normal. If pressurizer level continues to decrease, exhibiting loss of coolant accident (LOCA) indications, procedures require reactor trip.

In response to the Commission Order dated May 17, 1980, the licensee has implemented a control grade anticipatory reactor trip for the loss of feedwater and turbine trip. This precludes the action taken in response to the bulletin.

Except for the issue of a manual reactor trip for loss of offsite power, the NRC staff finds the licensee's response to Item 4 of IE Bulletin 79-05B complete.

Item 5 "Provide for NRC approval a design review and schedule for implementation of a safety-grade, automatic anticipatory reactor scram for loss of feedwater, turbine trip, or significant reduction in SG level."

In Reference 4 to this evaluation, the licensee provided a conceptual safety grade design for initiating reactor trips upon loss of main feedwater and/or turbine trip. The licensee did not commit to implementing an automatic anticipatory reactor trip from significant reduction in SG level.

The licensee provided an analysis which shows that such a trip would not be anticipatory. With regard to the schedule, the licensee stated in Reference 4 that installation of the modification would occur during an outage which is of sufficient length to accommodate the change but no later than the next refueling outage, which is now scheduled for approximately January 1981.

Subsequent to the issuance of IE Bulletin 79-05B, the Commission Order of May 17, 1979, was issued to the licensee. One of the immediate actions required of the licensee, based on this Order, was to install a control-grade reactor trip for loss of main feedwater and turbine trip. The Order requires that for continued long-term operation, the licensee must upgrade this circuitry to meet safety-grade criteria. A letter was issued to the licensee, dated September 7, 1979 (Reference 11), which forwarded a request for additional information on the proposed design. This information was needed before the staff could approve the proposed design for ANO-1. In addition, this letter requested that the licensee expedite its installation schedule such that installation and testing could be completed within about six (6) months following NRC staff approval of the design.

By letter dated October 31, 1979, supplemented by letters dated December 18, 1979, and January 15, 1980, the licensee provided a description of a proposed design for automatic initiation and control of the EFW system independent of the ICS. We reviewed the proposed design at a meeting on January 18, 1980, and found that the design satisfies the control-grade provision of Item 2.1.7a of NUREG-0578 (Automatic Initiation of the AFW). The licensee completed the proposed modifications during January 1980.

The licensee in Reference 12 to this evaluation committed to implement the safety-grade anticipatory reactor trip for loss of feedwater and turbine trip within six (6) following the NRC approval of the design.

In Reference 13 to this evaluation the licensee requested an amendment to the ANO-1 operating license which would provide limited conditions of operation related to the anticipatory reactor trips. Our review of this has not been completed.

The NRC staff finds that the licensee has been responsive to Item 5 of IE Bulletin 79-05B and that any further follow-up action on this matter will be handled under the long-term portion of the Commission Order of May 17, 1979. Therefore, the NRC staff finds the licensee's response to this item complete.

Item 6 "The actions required in Item 12 of IE Bulletin 79-05A are modified as follows:

> Review your prompt reporting procedures for NRC notification to assure that 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 and continuous communication channel shall be established and maintained with the NRC."

The NRC finds that the licensee was responsive to Item 6 of IE Bulletin 79-05B. Subsequent to the issuance of IE Bulletin 79-05B the NRC amended the regulation which sets forth requirements for the reporting of significant events at operating reactors. Paragraph (a) of §50.72 requires, in part, that the licensee notify the NRC Operations Center as soon as possible and in all cases within one hour by telephone of the occurrence of any significant event listed in the paragraph. Therefore, no additional action will be taken on this matter under IE Bulletin 79-05B.

Item 7 "Propose changes, as required, to those TSs which must be modified as a result of your implementing the above items."

In Reference 14 the licensee requests an amendment to the license for ANO-1 which would provide limiting conditions of operation for loss of emergency feedwater. In Reference 12 the licensee requests an amendment to the license for ANO-1 which relates to operability and surveillance requirements for the anticipatory reactor trips upon loss of feedwater and turbine trip.

These proposed TS changes are presently under review and will be the subject of a separate safety evaluation which is being developed to support the necessary license amendment.

The NRC staff finds the licensee's response to Item 7 of IE Bulletin 79-05B complete.

SUMMARY OF OUTSTANDING ITEMS

As a result of the staff's review of the licensee's responses to IE Bulletins 79-05A and 79-05B, the staff has identified certain items for which additional information must be obtained in order to resolve these matters. The list below summarizes these matters. A more detailed discussion of these items is provided under the appropriate IE Bulletin item of this evaluation.

IE BULLETIN 79-05A

Item 8 A separate safety evaluation to support the required license amendment will be prepared that will include all TS changes necessary as a result of the licensee's implementation of IE Bulletin 79-05A.

The licensee has taken some exceptions to the IE Bulletin 79-05A. The licensee should conform to the provisions of the Bulletin as stated until sufficient justification for those proposed deviations have been provided. We will request such conformance.

Item 9 The subject of containment purging during power operation is presently under staff review as part of the NRR Generic Issues Program (Task - B-24: "Venting and Purging of Containment While at Power Operation and Effects on LOCA"). Any additional information needed to resolve this matter will be developed under this generic activities task. Therefore, no additional action will be taken on this matter under IE Bulletin 79-05A.

Item 10 The licensee has revised Plant Quality Control procedures relating to maintenance of safety-related systems. We will request the licensee to apply for a revision of the TSs which will require verification of the operability of redundant components when removed from service for testing and maintenance.

IE BULLETIN 79-05B

Item 4 This item required the licensee to implement a reactor trip for a loss of offsite power. We will request the licensee to provide further justification for not providing a manual trip on loss of offsite power.

Item 5 This item required the licensee to submit a proposed design and schedule for a safety-grade, automatic anticipatory reactor trip. The Commission Order of May 17, 1979, requires that this feature be installed as part of the long-term requirements of the Order. Therefore, any additional information required of the licensee in this matter will be reviewed as part of the NRC staff's evaluation of the licensee's compliance with long-term portion of the Order and no additional action will be taken under IE Bulletin 79-05B.

Item 7 A separate evaluation to support the required license amendment will be prepared that will include the licensee's proposed modifications to the TSs. Therefore, no additional action will be taken on this matter under IE Bulletin 79-05B.

CONCLUSIONS

Based on our review of the information provided by the licensee in response to IE Bulletins 79-05A and 79-05B, and with the exception of the outstanding items identified above, we conclude that the licensee has acceptably responded to these Bulletins. The actions taken by the licensee demonstrate its understanding of the concerns and implications of the TMI-2 accident as they relate to ANO-1. These actions have resulted in added assurance for the continued protection of the public health and safety during plant operation.

REFERENCES

1. Letter from David C. Trimble (AP&L) to K. V. Seyfrit (NRC), dated April 11, 1979 providing responses to Items 1, 2, 3, 4a and 5 of IE Bulletin 79-05A.

2. Letter from David C. Trimble (AP&L) to K. V. Seyfrit (NRC), dated April 16, 1979, providing responses to Items 4b through 4d, and 6 through 12 of IE Bulletin 79-05A.

3. Letter from David C. Trimble (AP&L) to K. V. Seyfrit (NRC), dated May 4, 1979, providing response to IE Bulletin 79-05B.

4. Letter from David C. Trimble (AP&L) to K. V. Seyfrit (NRC), dated May 21, 1979, providing proposed design and schedule for anticipatory reactor trip on loss of feedwater and turbine trip.

5. Letter from J. H. Taylor (B&W) to D. F. Ross (NRC), dated August 17, 1979, forwarding B&W's generic report BAW 1564 entitled "Integrated Control System Reliability Analysis".

6. Letter from H. R. Denton (NRC) to William Cavanaugh, III (AP&L), dated May 31, 1979, permitting resumption of operation in accordance with the terms of the Order of May 17, 1979, and enclosing the "Evaluation of Licensee's Compliance with the NRC Order Dated May 17, 1979 - AP&L - ANO-1 - Docket No. 50-313.

7. Letter from William Cavanaugh, III (AP&L) to R. W. Reid (NRC), dated May 16, 1979, providing request for change for the TSs which would provide limited conditions of operation in the event of loss of emergency feedwater equipment.

8. Letter from William Cavanaugh, III (AP&L) to R. W. Reid (NRC), dated April 2, 1979, concerning TSs for limiting purging to 90 hours per year.

9. Letter from G. L. Madsen (NRC) to William Cavanaugh, III (AP&L) dated July 6, 1979, transmitting IE Inspection Report Nos. 50-313/79-10 and 50-368/79-10.

10. Letter from William Cavanaugh, III (AP&L) to R. W. Reid (NRC), requesting change to the TSs reflecting a PORV pressure setting of 2450 psig and high pressure scram setpoint of 2300 psig.

11. Letter from R. W. Reid (NRC) to all B&W Operating Plants, dated September 7, 1979, requesting additional information concerning the upgrade of the anticipatory trip (loss of feedwater and turbine trip).

12. Letter from David C. Trimble (AP&L) to R. W. Reid (NRC), dated October 8, 1979, providing response to letter of September 7, 1979 concerning anticipatory reactor trip.

REFERENCES

13. Letter from William Cavanaugh, III (AP&L) to R. W. Reid (NRC), dated June 6, 1979, requesting TS changes relating to anticipatory reactor trip.
14. Letter from William Cavanaugh, III (AP&L) to R. W. Reid (NRC), dated June 6, 1979, request for TS changes which would provide surveillance requirements and limited conditions of operation relating to anticipatory reactor trip.

Dated: June 13, 1980