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May 4, 1979

1-059-8

Mr. K. V. Seyfrit, Director
Office of Inspection & Enforcement
U. S. Nuclear Regulatory Commission
Region IV
611 Ryan Plaza Drive, Suite 1000
Arlington, Texas 76011

Subject: Arkansas Nuclear One - Unit 1
Docket No. 50-313
License No. DPR-51
IE Bulletin 79-05B
(File: 1510.1)

Gentlemen:

Attached is our response to IE Bulletin 79-05B.

Very truly yours,

David C. Trimble

David C. Trimble
Manager, Licensing

DCT:ew

Attachment

cc: Mr. John G. Davis, Director (w/a)
Nuclear Regulatory Commission
Office of Inspection and Enforcement
Division of Reactor Operations Inspection
Washington, D.C. 20555

Mr. W. D. Johnson
U. S. Nuclear Regulatory Commission
P. O. Bx 2090
Russellville, AR 72801

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Question:

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 reactor coolant pumps, 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;

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 auxiliary feedwater are lost while in the natural circulation core cooling mode.

Response:

Procedures are currently 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.

These procedures shall include the following guidance in assessing heat removal efficiency and natural circulation stability:

1. Verification that following loss of forced circulation from rated power, Reactor Coolant System (RCS) hot leg temperatures stabilize after reactor coolant pump (RCP) coastdown and then tend to decrease, and the RCS hot leg temperature and cold leg temperature difference tends to decrease (i.e., tend to converge).

2. Verification that primary-to-secondary heat rejection is occurring by observing that turbine bypass or atmospheric dump valve operation is required or steam generator (OTSG) steam safety valve operation is required to limit secondary pressure, and emergency feedwater (EFW) is required to maintain OTSG level.

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In the event conditions occur which force the operator to purposefully terminate all RCP operation (other than during a normal plant cooldown), procedures will be revised to direct that prior to stopping all RCP's:

1. Flow to each OTSG is verified by placing EFW in operation, cycling the feed valves to each OTSG, and noting the corresponding EFW pump discharge pressure change.

2. Pressurizer level is verified to be between 50 inches and 90 inches.

3. RCS is verified to be at least 50 degrees subcooled.

4. At least one bank of pressurizer heaters are demonstrated operational.

5. OTSG operate range level is to be slowly increased, commensurate with the ability to maintain RCS pressure and pressurizer level, to approximately 50% indication via normal feedwater (if normal feedwater is available and times permits).

Procedures currently require maintenance of RCS pressure-temperature relationship within Appendix G limits. Proposed revisions shall also contain such requirements.

Caution notes will be included in natural circulation procedures to require manual High Pressure Injection (HPI) initiation and operation of at least one RCP per loop (if operable) in the event heat rejection via the OTSG's is irreparably lost. (provided these actions do not create an unsafe condition)

Such procedure modifications and associated training on those modified procedures shall be complete prior to exceeding 5% rated power from the current refueling outage.

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Question:

2) Modify the actions required in Item 4a and 4b 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 actions 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 then the HPI should be secured (as noted in b(2) below).

b. Operating procedures currently, or are revised to, specify that if the high pressure injection (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 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 the vessel integrity"

Response:

Our responses to question 4a and b in our letter of April 11, 1979, are revised as follows.

4a. A caution note will be added to the ANO-1 emergency procedures instructing the operators not to override the automatic actions of the Engineered Safeguards (ES), unless continued operation will result in an unsafe plant condition, without first determining the consequences of that override and consulting with the shift supervisor. The procedures will also be modified to add clarifying steps to aid operators in recognizing a spurious actuation and provide for orderly termination of the suprious actuation.

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4b. The ANO-1 emergency procedures will be modified to specify the following actions. If the High Pressure Injection (HPI) system has been automatically actuated because of a low pressure condition it must remain in operation until:

1) Low Pressure Injection is in progress with a flow rate in excess of 2000 gpm 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 50F below the saturation temperature for the existing Reactor Coolant System pressure. If 50F degrees subcooling cannot be maintained after HPI cutoff, the HPI shall be reactivated. The degree of subcooling beyond 50F and the length of time the HPI is in operation shall be limited by the pressure/temperature considerations for the vessel integrity; or,

3) The RCS pressure returns to normal operating pressure with the temperature in the hot and cold legs being controlled with at least 50F subcooling by an operable steam generator.

2232 356

Question:

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.

Response:

We have discussed the opening of the Electromatic Relief Valve (ERV) during normal transients with B&W and believe the changes proposed in our April 24, 1979 letter are appropriate to preclude lifting of the ERV during normal transients.

During our current refueling outage and before exceeding 200F Reactor Coolant System (RCS) bulk system temperature (Tave) we will increase the relief setpoint of the ERV from 2255 psig to 2450 psig and decrease the high RCS reactor trip setpoint from 2355 psig to 2300 psig. These changes will preclude opening of the ERV during normal plant operating transients yet will not increase the expected frequency of safety valve operation.

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Question:

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 reactor coolant system. 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

   f. low pressurizer level

Response:

Prior to exceeding 200F Reactor Coolant System (RCS) bulk system temperature (Tave), we will provide procedures and operator training for a prompt manual reactor trip, as appropriate, for the transients listed below.

4a. Loss of Main Feedwater - As committed in our May 3, 1979 letter to Mr. Denton, we will provide a control grade anticipatory reactor trip upon loss of main feedwater. Therefore, procedures will require verification of reactor trip upon loss of main feedwater. The term verification as it will be used in this procedure requires the operator to take the necessary manual actions to trip the reactor if the automatic system has not already caused a trip.

4b. Turbine Trip - As committed in our May 3, 1979 letter to Mr. Denton, we will provide a control grade anticipatory reactor trip upon a turbine trip. Therefore, procedures will require verification of a reactor trip.

4c. Main Steam Isolation Valve Closure - A manual reactor trip will be initiated by the operator in the event of a main steam isolation valve closure.

4d. Loss of Offsite Power - Upon separation of the Arkansas Nuclear One - Unit 1 (ANO-1) turbine generator from the offsite grid, the plant is designed to runback to approximately 15% Full Power and supply internal house loads via the main generator through the auxiliary transformer. The auxiliary transformer is fed directly from the main generator and is the normal source of onsite power for house loads during operations. Should the turbine trip during runback of the plant, the reactor would trip. (See our response to Item 4b. above).

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We believe it would be more prudent to allow the plant to runback to approximately 15% power and supply house loads through the auxiliary transformer than to trip the reactor upon grid separation and thereby lose an available source of power.

By maintaining the turbine generator on line, we are able to run all Reactor Coolant Pumps, as well as all safety and secondary equipment without challenging the Diesel Generators and maintaining them as backup sources of power.

We therefore, state that we believe tripping the reactor upon separation of the turbine generator from the system grid is a reduction in the margin of safety and such a trip should not be incorporated on ANO-1.

4e. Low OTSG Level - OTSG startup range level indication varies in an approximately linear fashion with power level. As such, the steam generator low level setpoints would require a sliding function with power level. However, since loss of feedwater transients typically occur below 40% power, a low level steam generator trip of 15 inches will provide optimum protection in the power range of highest probability. Above 40% power, the reactor would be tripped on loss of feedwater indications, although the 15-inch administrative limit will remain active as a backup trip. Therefore, above 40% power, the reactor will be tripped upon OTSG level falling to 15 inches (indicated) or below.

4f. Low Pressurizer Level - Reactor trip upon low pressurizer level is inconsistent with this question's general concern for overpressurization events, as a low pressurizer level is associated with underpressurization rather than overpressurization. Further, 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 necessary, additional makeup flow can be obtained by opening the high pressure injection motor operated valves or starting an additional makeup pump. If pressurizer level continues to decrease, exhibiting loss of coolant accident indications, procedures require reactor trip. The reactor core will continue to be protected by the low RC pressure trip as analyzed in the FSAR.

2232 359

Question:

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 steam generator level.

Response:

As commented in our letter of May 3, 1979, to Mr. Denton, during our current refueling outage and before exceeding 200F Reactor Coolant System (RCS) bulk system temperature (Tave), we will implement a control grade anticipatory trip to trip the reactor on a loss of Main Feedwater (MFW) or on a Turbine Trip. Commensurate with the above time frame, we will submit a schedule for upgrading these two reactor trips to safety-grade trips.

2232 360

Question:

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 in not in a controlled or expected condition of operation. Further, at that time an open continuous communication channel shall be established and maintained with NRC.

Response:

Our response to question 12 in our letter of April 16, 1979, is hereby modified to include the following paragraph.

"In addition, upon notifying the NRC that such a condition exists, a telephone number will be provided to the NRC that can be reached on a continuous (i.e., 24 hours a day) basis until normal conditions are restored."

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Question:

7) Propose changes as required, to those technical specifications which must be modified as a result of your implementing the above items.

Response:

Our letter to Mr. R. W. Reid of April 24, 1979, proposed a change to the Arkansas Nuclear One-Unit 1 (ANO-1) Technical Specifications to incorporate our response to Item 3. We are currently evaluating the necessity of other changes to the ANO-1 Technical Specifications that will be required by implementing the above responses. If further changes are required, we will provide the appropriate changes by May 21, 1979.

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