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June 29, 1979

2-069-16

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 2
Docket No. 50-368
License No. NPF-6
IE Bulletin 79-06B
(File: 2-1510)

Gentlemen:

Attached is our revised response to I&E Bulletin 79-06B which incorporates comments both from I&E and NRR. This response supercedes our response of April 24, 1979.

Very truly yours,

John M. Griffin for

David C. Trimble
Manager, Licensing

DCT:JTE:vb

Attachment

cc: Mr. W. D. Johnson
U. S Nuclear Regulatory Commission
P. C Box 2090
Russellville, AR 72801

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Question:

1) Review the description of circumstances described in Enclosure 1 of IE Bulletin 79-05 and the preliminary chronology of the TMI-2 3/28/79 accident included in Enclosure 1 to IE Bulletin 79-05A.

a. This review should be directed toward understanding: (1) the extreme seriousness and consequence[illegible] the simultaneous blocking of both auxiliary feedwater [illegible]ns at the Three Mile Island Unit 2 plant and other actions taken during the early phases of the accident; (2) the apparent operational errors which led to the eventual core damage; (3) that the potential exists, under certain accident or transient conditions, to have a water level in the pressurizer simultaneously with the reactor vessel not full of water; and (4) the necessity to systematically analyze plant conditions and parameters and take appropriate corrective action.

b. Operational personnel should be instructed to: (1) not override automatic action of engineered safety features unless continued operation of engineered safety features will result in unsafe plant conditions (see Section 6a.); and (2) not make operational decisions based solely on a single plant parameter indication when one or more confirmatory indications are available.

c. All licensed operators and plant management and supervisors with operational responsibilities shall participate in this review and such participation shall be documented in plant records.

Response:

We have reviewed, in detail, Enclosure 1 of IE Bulletin 79-05 and the sequence of events included in Enclosure 1 to IE Bulletin 79-05A. We have worked closely with NRC, B&W, and Met-Ed in gathering, analyzing, and disseminating all data we have been able to obtain to date to assure that our understanding of the incident is as accurate as possible.

Our reviews have been specifically oriented toward those areas of concern addressed in IE Bulletin 79-06B and have included presentations to the operations staff by NRC, I&E Region IV personnel, and NRC Operator Licensing Branch personnel, in addition to thorough reviews and discussion by plant staff personnel. All licensed operators and plant management/supervisors with operational responsibilities have participated in these reviews. Documentation as to their participation has been kept.

Additional operator instruction and guidance addressing the specific concerns of item 1.b of IE Bulletin 79-06B are being revised into plant procedures as detailed in our response to item 6.

Question:

2) 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. Operation action required to prevent the formation of such voids.

   c. Operator action required to enhance core cooling in the event such voids are formed (e.g., remote venting).

Response:

2a) The Arkansas Nuclear One - Unit 2 (ANO-2) emergency procedures will be modified as a result of the Three Mile Island - Unit 2 (TMI-2) incident, to recognize the possibility of forming steam voids in the Reactor Coolant System (RCS). The operator's need to monitor the RCS parameters to detect conditions at or near saturation will be emphasized.

Natural circulation is established by termination of forced flow (tripping the Reactor Coolant Pumps). Operators will be instructed to maintain forced flow until an adequate margin to saturation is obtained to assure natural circulation will be established following termination of the RCPs or unless continued operation of the RCPs would create an unsafe plant condition.

Voids in the RCS may be recognized by:

   1) Oscillations in RCP amperage and Δ P

   2) Incore thermocouples indicating super heat

   3) Oscillations in nuclear instrument (due to reduced shielding as a result of the voids).

2b) The ANO-2 emergency procedures will be modified to minimize the formation of such steam voids by:

   1) requiring the operator to check the reactor coolant pressure and temperature during recovery from a reactor trip and other transients in order to achieve and maintain subcooling of the reactor coolant in the hot and cold legs. During "followup actions" the operators will take steps to maintain at least 50F subcooling in the RCS.

   2) providing for termination of operation of Engineered Safety Features (ESF) systems only when the conditions described in the response to Questions 6.b.1 and 6.b.2

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are met. This will aid in the prevention of steam void formation and in steam void elimination should they be formed and ensure continued core cooling.

2c) The ANO-2 emergency procedures will be modified to ensure that the core is cooled in the event that such voids are formed by:

1) providing for termination of operation of the ESF Systems when automatically actuated by low pressure conditions only when the conditions described in the response to Questions 6.b.1 and 6.b.2 are met. This will aid in both the prevention of steam void formation and in steam void elimination should they be formed, thus, ensuring core cooling.

2) providing for the continued operation of at least one reactor coolant pump per loop to assist in core cooling during accident conditions if it is advantageous for those accident conditions.

3) providing for operator action to open Pressurizer ECCS vents in case steam voids are formed in the RCS <u>and</u> RCS flow and heat rejection to the steam generators is inadequate.

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Question:

3) Review the containment isolation initiation design and procedures, and prepare and implement all changes necessary to permit containment isolation whether manual or automatic, of all lines whose isolation does not degrade needed safety features or cooling capability, upon automatic initiation of safety injection.

Response:

We have reviewed the Containment Isolation Actuation System (CIAS) design and procedures and have listed below the automatically actuated valves which provide penetration isolation (valve numbers in parenthesis):

Category I

1) Chemical and Volume Control System Letdown (2CV-4821-1 and 2CV-4823-2)

Category II

2) Chilled Water Supply to Containment Coolers (2CV-3852-1)

3) Chilled Water Supply from Containment Coolers (2CV-3850-2 and 2CV-3851-1)

4) Component Cooling Water to Reactor Coolant Pump Coolers (2CV-5236-1)

5) Component Cooling Water from Reactor Coolant Pump Coolers (2CV-5254-2 and 2CV-5255-1)

Category III

6) Containment Vent Header (2CV-2400-2 and 2CV-2401-1)

7) Reactor Coolant System and Pressurizer Sample (2SV-5833-1 and 2SV-5843-2)

8) Nitrogen Supply to Safety Injection Tanks (2CV-6207-2)

9) Quench Tank Liquid Sample (2SV-5878-1 and 2SV-5871-2)

10) Safety Injection Tank Sample (2SV-5876-2)

11) Quench Tank Makeup Water Supply (2CV-4690-2)

12) Containment Sump Drain (2CV-2060-1 and 2CV-2061-2)

13) Containment Purge Inlet (2CV-8289-1, 2CV-8284-2 and 2CV-8283-1)

14) Containment Purge Outlet (2CV-8291-1, 2CV-8236-2 and 2CV-8285-1)

15) Low Pressure Nitrogen Supply (2CV-6213-2)

16) Reactor Drain Tank Drain (2CV-2202-1 and 2CV-2201-2)

Category IV

17) Reactor Coolant Pump Controlled Bleedoff (2CV-4847-2 and 2CV-4846-1)

18) Steam Generator Sample (2CV-5852-2 and 2CV-5859-2)

19) Air Particulate Monitor in Hydrogen Purge System (2SV-8231-2, 2SV-8273-1 and 2SV-8271-2)

20) Air Particulate Monitor in Containment Atmosphere Sample (2SV-8261-2, 2SV-8265-1 and 2SV-8263-2)

21) Fire Water Supply (2CV-3200-2)

Item 1 (Category I) above is isolated upon receipt of a Safety Injection Actuation Signal (SIAS) or a Containment Isolation Actuation Signal (CIAS). SIAS is generated when Reactor Coolant System pressure is less than or equal to 1740 psia or when Containment Building pressure is greater than or equal to 18.4 psia. CIAS is generated when Containment Building pressure is greater than or equal to 18.4 psia.

Items 2 through 21 isolate upon receipt of a CIAS. The valves noted in Items 2 through 5 (Category II) are normally open during power operation since they are in systems which provide support to needed systems within the Containment Building. The valves noted in Items 6 through 16 (Category III) are normally closed during power operation and are only opened periodically by specific manual operation, i.e. there is no automatic opening of any of these valves. The valves noted in Items 17 through 21 (Category IV) are normally open during power operation but are not necessary to be open following receipt of a SIAS.

Since Items in Category II are providing support to systems within the Containment Building, the valves should stay open upon receipt of a SIAS to prevent unnecessary equipment damage. The systems represented in Category II contribute to a "normal", orderly cooldown following receipt of a SIAS.

Items in Category III are normally closed during power operation and specific manual operation is required to open them. Furthermore, to cause full opening of the penetration, specific manual operation of at least two valves is required, each of which requires a specific and deliberate action. Based on this fact, no changes to the Containment Building Isolation System are needed.

Items in Category IV are normally open during power operation and specific manual operation is required to close these valves following receipt of a SIAS. Each of the Category IV systems were reviewed and it has been verified that a direct connection between the Containment Building atmosphere and the Auxiliary Building atmosphere or the environment does not exist while these penetrations are open. Based on this fact, no changes to the Containment Building Isolation System are needed.

However, to further increase the margin of safety, a design change is being evaluated for items in Category III and IV to add a SIAS to those valves. This design change will provide an additional degree of assurance that no release path to the environs exists upon receipt of a SIAS without a concurrent CIAS. These modifications resulting from this evaluation will be implemented during the first available plant outage to cold shutdown of sufficient duration to accommodate the modification following completion of the design change package, but no later than the first refueling outage.

Question:

4) For facilities for which the auxiliary feedwater system is not automatically initiated, prepare and implement immediately 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 adequate auxiliary feedwater to the steam generator(s) for those transients or accidents, the consequences of which can be limited by such action.

Response:

This question is not applicable to Arkansas Nuclear One-Unit 2 (ANO-2) as the Emergency Feedwater System (EFW) is designed to automatically initiate as part of the Engineered Safety Features Actuation System (ESFAS). The system is designed to the latest revision (Rev. 1) of Branch Technical Position ASB 10-1.

Those Design Basis Events which will cause automatic Emergency Feedwater actuation are:

Steamline Break (Inside Containment)
Steamline Break (Outside Containment)
Loss of Main Feedwater

The variables which are monitored to indicate the above events are:

Steam Generator Pressure
Steam Generator Level

Further information on the ANO-2 EFW System is presented in the ANO-2 Final Safety Analysis Report (FSAR) Sections 7.3.1.1.11.8. and 10.4.9, and in Volume IX, Response to NRC Questions 020.35, 020.54, 222.22, and 222.90.

Question:

5) For your facilities, 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 or pressure 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.

Response:

Question 5 is not applicable to Arkansas Nuclear One-Unit 2 (ANO-2) as the ANO-2 design does not include power operated relief valves on the pressurizer. Overpressurization of the Reactor Coolant System is precluded by means of safety valves and the Reactor Protective System (RPS).

Information on the safety valves is presented in Sections 5.2.2 and 5.5.10 and Chapter 5A of the ANO-2 Final Safety Analysis Report (FSAR). Information on the RPS is presented in Section 7.2 of the ANO-2 FSAR.

Question:

6) 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 for 20 minutes or longer; at a rate which would assure stable plant behavior; or

(2) The HPI system has been in operation for 20 minutes, and all hot and cold leg temperatures are at least 50 degrees 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.

c. Operating procedures currently, or are revised to, specify that in the event of HPI initiation with reactor coolant pumps (RCP) operating, at least one RCP shall remain operating in each loop as long as the pump(s) is providing forced flow.

d. Operators are provided additional information and instructions to not rely upon pressurizer level indication alone, but to also examine pressurizer pressure and other plant parameter indications in evaluating plant conditions, e.g., water, inventory in the reactor primary system.

Response:

6a) A caution note will be added to the ANO-2 emergency procedures instructing the operators not to override the automatic actions of the Engineered Safety Features (ESF) 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 sp[illegible]us actuation.

6b) The ANO-2 emergency procedures will be modified to specify the following actions. If the High Pressure Safety Injection (HPSI) system has been automatically actuated because of a low pressure condition it must remain in operation until:

1) Low Pressure Safety Injection (LPSI) is in progress with a flow rate in excess of 2000 gpm and the situation has been stable for 20 minutes, or,

2) The HPSI 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 RCS pressure. If 50F degrees subcooling cannot be maintained after HPSI cutoff, the HPSI shall be reactivated; 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; or,

4) Unless continued operation would result in an unsafe plant condition.

6c) The ANO-2 "Loss of Coolant/RC Pressure" emergency procedure will be revised to specify:

For break sizes exceeding HPSI Capacity:

If HPSI initiation automatically occurs because of low Reactor Coolant pressure, and the reactor coolant pumps are in operation, then at least one RCP/loop will remain in operation until Low Pressure Safety Injection flow is established and verified.

For break sizes within HPSI Capacity:

If HPSI initiation automatically occurs because of low Reactor Coolant pressure and the reactor coolant pumps are in operation, then at least one RCP/loop will remain in operation until LPSI injection or decay heat is established and verified or continued operation of the RCPs would create an unsafe plant condition.

6d) ANO-2 plant procedures are being revised to require the operators to monitor RCS pressures and temperatures following transients to assure that adequate margin to saturation conditions is maintained.

To lessen the ambiguity of this parameter (margin to saturation), special scales for RCS temperature and pressure indicators and recorders are being constructed which correlate saturation pressures and temperatures (reduced by 50F) to the current indicated temperature and pressure scales. With these special scales, the operator can rapidly assess core conditions relative to voiding.

Question:

7) Review all safety-related valve positions, positioning requirements and positive controls to assure that valves remain 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, testing, plant and system startup, and supervisory periodic (e.g., daily/shift checks,) surveillance to ensure that such valves are returned to their correct positions following necessary manipulations and are maintained in their proper positions during all operational modes.

Response:

We have completed a review of the Engineered Safety Feature (ESF) valves and their positioning requirements. The ESF systems are:

a) Containment Isolation System (CIS)

b) Containment Spray System (CSS)

c) Containment Cooling System (CCS)

d) Safety Injection System (SIS)

e) Penetration Room Ventilation System (PRVS)

f) Main Steam Isolation System (MSIS)

g) Emergency Feedwater System (EFS)

h) Chemical and Volume Control System (CVCS)

i) Diesel Fuel Oil and Starting Air System

j) Emergency Boration Systems

k) Service Water

Based on this review, and our review of related procedures, we have concluded that our procedures are adequate to ensure that valves in ESF systems are maintained in their proper position, or are capable of being properly positioned in the event of an Engineered Safety Feature Actuation Signal (ESFAS).

The procedures reviewed are summarized as follows:

Maintenance - Prior to taking an ESF system out of service, the Control Room must be notified as required by procedure. The redundant train of the affected ESF system will be inspected to verify operability prior to taking the aforementioned system out of service. The inspection will include checking control board indications, MOV status, alarm status, and verification that the last

surveillance test was within the surveillance interval and demonstrated operability. The out-of-service system includes components for which maintenance is to be performed as well as the valve(s) used to isolate the component for maintenance. Tags are placed on the affected out-of-service equipment, both at the equipment proper and at Motor Control Center (MCC) breakers, if applicable. Additionally, the out-of-service equipment is entered into the station log. Following completion of maintenance, and removal of out-of-service tags, the system is re-aligned to its proper configuration by the operator, the Control Room is notified of system return to service, and entry is made in the station log. Surveillance tests are performed to verify the operability of the affected equipment.

Testing - All ESF systems are required by ASME Section XI and/or ANO-2 Technical Specifications to be tested to ensure operability. Test frequencies vary according to the component being tested, and the reason for testing. Upon completion of ESF system testing, for whatever reason, the subject system is verified as required by procedure to be properly aligned to allow the system to perform its safety function. The verification of lineup is done by the operator using sign-offs in the procedure.

During our review, all manually operated valves were found to be procedurally required to be in their correct position. The procedures further require the system lineups to be verified correct prior to declaring the system operable. However, several of these valves in systems not classified as ASME Codes 1, 2, or 3 were not subject to the "Category E" listing (i.e., required to be locked, sealed or otherwise secured in their proper position). These valves, in the Diesel Fuel Oil System and Diesel Starting Air System, though not classified as Class 1, 2, or 3 will be added to the "Category E" procedure list and as such are required to be and will be locked, sealed, or otherwise secured in their proper position during operation, thus further assuring proper valve positioning of all safety-related valves in their associated system. These procedural changes will be implemented by June 1, 1979.

Thus, based on procedural controls and this review, we feel assured that all safety-related valves are positioned in, or are capable of being positioned in their ESF position upon receipt of an ESFAS, thereby ensuring the required response of systems to postulated events.

Startup - All safety-related systems are required to be operable (valves in the correct position) prior to and/or during plant startup as appropriate.

Question:

8) 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, venting or other release 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.

c. The basis on which continued operability of the above features is assured.

Response:

The systems designed to transfer potentially radioactive gases and liquids out of the Containment Building at ANO-2 do not automatically discharge under any conditions. The systems that require specific manual operation are as follows (valve numbers in parenthesis):

1) Chemical and Volume Control System Letdown (2CV-4821-1 and 2CV-4823-2)

2) Containment Vent Header (2CV-2400-2 and 2CV-2401-1)

3) Reactor Coolant System and Pressurizer Sample (2SV-5833-1 and 2SV-5843-2)

4) Quench Tank Liquid Sample (2SV-5878-1 and 2SV-5871-2)

5) Safety Injection Tank Sample (2SV-5876-2)

6) Containment Sump Drain (2CV-2060-1 and 2CV-2061-2)

7) Containment Purge Inlet (2CV-8289-1, 2CV-8284-2 and 2CV-8283-1)

8) Containment Purge Outlet (2CV-8291-1, 2CV-8286-2 and 2CV-8285-1)

9) Reactor Drain Tank Drain (2CV-2202-1 and 2CV-2201-2)

10) Reactor Coolant Pump Controlled Bleedoff (2CV-4847-2 and 2CV-4846-1)

11) Hydrogen Purge System and Air Particulate Monitor (2SV-8231-2, 2SV-8273-1 and 2SV-8271-2)

12) Containment Atmosphere Sample and Air Particulate Monitor (2SV-8261-2, 2SV-8265-1 and 2SV-8263-2)

All of the above systems receive a CIAS to close. None of the systems will reopen following resetting of the CIAS without specific operation.

Item 1 receives a SIAS as well as a CIAS to close.

Items 2 through 9 are normally closed during power operation and require specific manual operation to open. They receive a CIAS to close and will not open following resetting of the CIAS without specific manual operation.

Items 10 through 12 are normally open during power operation. They receive a CIAS to close and will not open following resetting of the CIAS without specific manual operation.

Items 2 (Containment Vent Header) and 8 (Containment Purge Outlet) have a high radiation release interlock to close the systems automatically. Items 11 (Hydrogen Purge System and Air Particulate Monitor) and 12 (Containment Atmosphere Sample and Air Particulate Monitor) have radiation alarms in the system which will annunciate in the Control Room to alert operator of a high radiation release.

All of the above listed valves are periodically surveillance tested as required by the ANO-2 Technical Specifications per Section 4.0.5 (ASME Section XI testing) and Section 3/4 6.1 (Appendix "J" to 10CFR50 testing). The CIAS is verified operable per Section 3/4.3.2 of the Technical Specifications. The radiation monitoring instrumentation is verified operable per Section 3/4/.3.3 of the ANO-2 Technical Specifications.

Based on the above, the inadvertent release of radioactive gases or liquids by automatic means following resetting of the CIAS is not possible. Specific manual operation of the systems would be required to open the containment penetrations. Furthermore, automatic means for discharging radioactive gases or liquids does not occur even prior to initiation of CIAS. Transfer of contaminated fluids requires specific manual operation.

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Question:

9) Review and modify as necessary your maintenance and test procedures to ensure that they require:

   a. Verification, by test or 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.

   c. Explicit notification of involved reactor operational personnel whenever a safety-related system is removed from and returned to service.

Response:

Prior to initiating maintenance on safety-related systems, the redundant system will be inspected to verify operability. The inspection will include checking control board indications, MOV status, alarm status, and verification that the last surveillance test demonstrated operability and was within the surveillance interval. Plant Quality Control procedures require that a Job Order be issued for any maintenance of safety-related ("Q") systems. All Job Orders require authorization by the affected unit's Shift Supervisor prior to work commencing.

The Job Order form is currently being revised to specifically identify all Pre-maintenance and Post-maintenance requirements, and to include verification that those requirements are met prior to declaring a system or component OPERABLE after maintenance.

The revised Job Order Form will be developed and implemented before the ANO-2 core is made critical following our current outage.

When performing testing on a safety-related system, ANO-2 Tech Specs address operability of the redundant system. Documentation is required, by a testing procedure and/or Job Order, by specific checkoffs and signoffs that a safety-related system is returned to its proper operable condition following testing of that system.

All safety systems taken out of service are noted in the Station Log and noted on the safety system status board. Current operating Procedures require operators, coming on shift, to read all log entries back to their previous shift or for the previous 7 days whichever is shorter and each shift is required to review the station log, plant annunciators, system status board, and equipment tag out book.

Question:

10) 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 continuous communication channel shall be established and maintained with NRC.

Response:

It has been and will continue to be AP&L's policy to promptly notify the Nuclear Regulatory Commission of any unusual event at Arkansas Nuclear One. This policy is not limited to items which are deemed reportable per the Technical Specifications or federal regulations.

In the event of an emergency situation, our procedures for implementing the Emergency Plan require early notification of the Nuclear Regulatory Commission.

To further clarify the above policy and procedure, we will modify the ANO Administrative Controls Manual and Emergency Procedure 1202.34 for Personnel Response, to include the following statement:

> "Upon notification by the Shift Supervisor of an event at ANO, the Duty Emergency Coordinator will assess the situation as to its seriousness. If the assessment indicates that the health and safety of the public might be endangered or there might be a potential for significant public interest (e.g. radioactivity release, etc.), AP&L Management and NRC shall be immediately notified regardless of the reportability of the event as defined in the Technical Specifications or federal regulations."

Notification of NRC that such a condition exists, will be via the "hot line" phone which was recently installed by NRC.

Question:

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

Response:

Current ANO-2 emergency procedures (LOCA procedures) were reviewed concerning H_2 concentration control in containment and were found to provide sufficient guidance and procedural control to minimize the potential danger of accumulation of significant quantities of H_2. In addition, the ANO-2 Containment Building is equipped with H_2 Recombiners. Further discussion of this system is presented in Section 6.2 of the ANO-2 Final Safety Analyses Report (FSAR).

In the event of H_2 accumulation in the Reactor Coolant System (RCS) we believe additional measures provided in our response to Question 2 above are sufficient to preclude and/or control such an event. However, the following methods could be used to remove large noncondensible gas voids.

1) Dissolve and/or suspend noncondensible gas by use of forced flow (Reactor Coolant Pumps) and/or adjustments to RCS temperature and pressure;

2) Degas the pressurizer via the steam space sample line;

3) Degas via letdown through the vacuum degassifier

4) Degas via the T_{hot} (hot leg) sample lines to the Volume Control Tank.

Question:

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

Response:

No changes to the ANO-2 Technical Specifications were deemed necessary as a result of these responses.