Central Files

INDIANA & MICHIGAN POWER COMPANY

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NEW YORK, N. Y. 10004

May 1, 1979
AEP:NRC:00185

Donald C. Cook Nuclear Plant Units 1 and 2
Docket Nos. 50-315 and 50-316
License Nos. DPR-58 and DPR-74

Mr. J. G. Keppler, Regional Director
U. S. Nuclear Regulatory Commission
Office of Inspection and Enforcement
Region III
799 Roosevelt Road
Glen Ellyn, Illinois 60137

Dear Mr. Keppler:

The purpose of this letter is to provide responses to the Nuclear Regulatory Commission's IE Bulletin No. 79-06A, Revision 1 which we received on April 23, 1979. The responses to Items 1 through 12 of the Bulletin are contained in the attached enclosure.

We in the New York office as well as our Cook Plant management have reviewed the circumstances and chronology of the Three Mile Island Unit 2 loss of feedwater accident. The NRC IE Bulletins and meetings with the NRC Staff have been very helpful in our review and understanding of the Three Mile Island Plant accident. This event was indeed unfortunate, but we stand to benefit from the lessons learned from the incident.

We would like to discuss some of the major differences that exist between the Westinghouse (W) design employed at the Cook Plant and the Babcock & Wilcox (B&W) design of the Three Mile Island Plant.

First, the reactor would have tripped immediately upon the turbine trip. This occurs as long as either the reactor thermal power or the turbine power is above 10%. Operability requirements for these channels of the reactor protection system are part of our Technical Specifications. There are a number of reactor trip circuits that are initiated by a departure from normal feedwater flow, water levels in the steam generators, or a trip of the turbine generator. The turbine, the generator and the reactor are interlocked to trip each other.

A second difference is the Westinghouse steam generator design. Cook's four steam generators are of the recirculation, U-tube type, with steam being produced at saturated conditions. A conservative estimate tells us that the amount of latent heat needed to dry up our steam generators is

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approximately 4.5 times larger than the amount needed to do the same in the B&W design. This difference is partly responsible for the longer time needed to uncover the reactor core in the Westinghouse design.

Finally when a Phase A containment isolation is initiated (caused by Safety Injection signal derived from the Reactor Protection System and/or Containment Pressure - High at 1.2 psi), the sump pumps become isolated. After isolation, these pumps do not become operable automatically; that is, operator action is required after either Safety Injection or Containment Isolation is reset. Cook's design would have made more difficult the primary coolant transfer without operator knowledge that took place in TMI-2 from inside the containment to the Auxiliary Building.

We believe our technical staff is competent and well-prepared. We do, however, recognize the need to continuously seek further improvement. I consider the Cook Plant personnel of very high quality and quite capable of operating a nuclear plant. Your staff is familiar with the Cook Plant's daily operations; your resident inspectors have been helpful to us in achieving our present level of performance.

In closing, we would add our belief that nuclear power represents an acceptable and needed source of electric energy for at least the next two decades. Rather than discourage its growth or use because of the Three Mile Island incident, we would be better off to learn how to make nuclear power safer and of further service to our economic well-being.

Very truly yours,

John E. Dolan

John E. Dolan
Vice President

JED:em
Attachment

Sworn and subscribed to before me
this 1st day of May, 1979 in
New York County, New York

Kathleen Barry
Notary Public

York County
1981

cc: R. C. Callen
G. Charnoff
R. W. Jurgensen
D. V. Shaller - Bridgman
N. C. Moseley - NRC

DONALD C. COOK NUCLEAR PLANT

RESPONSE TO IE BULLETIN 79-06A, REV. 1

NRC ACTION

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 consequences of the simultaneous blocking of both auxiliary feedwater trains 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 7a.); 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.

AEP RESPONSE

On April 5, 1979, two members of the AEPSC Nuclear Engineering Division and the Cook Plant Operations Superintendent participated in a meeting at Westinghouse Electric Corporation offices, along with other utilities operating Westinghouse PWR's, to review the events which transpired during the Three Mile Island incident and their impact on Westinghouse PWR's.

Review meetings with the four operating groups or shifts were held by the Operations Superintendent on April 10, 11, 12 and 14, 1979. Even though these were conducted prior to the date of this IE Bulletin, the requested review material was used and it was reviewed in detail. Also reviewed was a letter from Westinghouse to AEP dated April 2, 1979. This letter contained preliminary recommendations related to the TMI incident.

Attending these meetings were all Licensed Shift Operators with the exception of two who were on vacation. Also attending were five Senior Licensed Personnel who are not assigned to the rotating shifts.

On April 18, 1979, Messrs. John Streeter and Robert Campbell of the NRC were at the Cook Plant and three meetings were held in which Mr. Streeter made a very detailed review of the sequence of events that occurred at TMI-2 including the NRC observations of importance to operating personnel. These meetings were attended by a total of 116 persons in the following categories:

Licensed Personnel Assigned to Rotating Shifts	29
Licensed Personnel Not Assigned to Rotating Shifts	11
Non-Licensed Operating Personnel Assigned to Rotating Shifts	34
Non-Licensed Operating Personnel Assigned to Training	9
Balance of Plant Personnel	24
Corporate Office, Ft. Wayne, Indiana	2
Corporate Office, New York, New York	4
NRC On-Site Inspectors	2
Westinghouse	1

All operations personnel, except two licensed and two non-licensed, assigned to rotating shifts attended the NRC review. Since then, onsite inspector Robert Masse has reviewed the events that occurred at TMI-2 with these four people.

On April 11 and 26, AEPSC personnel attended meetings called by the NRC in their Bethesda, Md. offices to discuss with Westinghouse and utilities operating Westinghouse NSSS plants, the impact of the Three Mile Island accident on design and operation of the Westinghouse NSSS plants.

A task force of AEPSC and Cook Plant personnel was formed on April 9, 1979 to study the loss of feedwater flow transient at the Three Mile Island Plant and to evaluate the Cook Plant design, procedures and training against the TMI incident for potential impact. The task force has had several meetings with AEP and Cook Plant management to discuss their findings to date.

1.a This entire paragraph was stressed in both sets of meetings at the Cook Plant on April 10-14 and 18 as follows: (1) During the chronological review when the Auxiliary Feed Pumps started without delivery of water it was explained what was thought to be the cause and why it took 8 minutes into the accident to establish flow. (2) The largest apparent errors during the transient were the tripping of Reactor Coolant Pumps, termination of HPI and the premature attempt to get on the RHR System. An explanation of why the open relief valve was not isolated until 2.3 hours into the accident could not be

established. (3) It was explained that a void forming anywhere in the Reactor Coolant System would have to displace water into the pressurizer and as such pressurizer level would not be a true indication of total system inventory. (4) It was explained that it is necessary for the operator to look at many confirmatory parameters before making decisions. The operator will use pressure and temperature readings to determine margin to the saturation point.

1.b Personnel were told (1) not to override automatic engineered safety features actions unless the cause is known and corrected, and (2) was a restatement of 1.a(4).

On April 7, 1979 a Standing Order was issued by the Cook Plant Operations Superintendent to the Shift Operating Engineers instructing them to manually initiate safety injection with an uncontrolled reactor coolant system pressure reduction when the pressurizer pressure reaches the low pressure setpoint for safety injection actuation regardless of pressurizer water level.

On April 26, 1979 the pressurizer low water level bistables were placed in the tripped position, after a complete study of the effects of doing so were made. Safety injection actuation will occur automatically on pressurizer low pressure regardless of the pressurizer water level signals. This matter is discussed further in the response to Item 3.

1.c Detailed review of the meetings with four operating shifts is documented and the attendance lists of the NRC review meetings are in the Plant files. Mr. Streeter also reviewed Bulletin 79-06A at the April 18, 1979 meetings.

NRC ACTION

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)

AEP RESPONSE

2.a The primary indication of void formation in the primary coolant system is when the pressurizer pressure falls below the hot leg saturation pressure. However, during a loss of coolant event, void formation in the primary coolant system would be expected with two exceptions: (1) the loss of coolant is being caused by a stuck open pressurizer relief valve which closes or is isolated before the system depressurizes to hot leg saturation, or, (2) the reactor coolant system reaches an equilibrium pressure above hot leg saturation, when the safety injection flow equals the break flow. Thus, in these two specific cases, confirmation of no voids in the system will be apparent by the pressure in the pressurizer. In the remaining cases the engineered safeguards system has been designed to cope with voiding. Thus, it is not necessary to be able to recognize void formation in those cases.

2.b These will be discussed as appropriate under Emergency Operating Procedures. Immediate actions, prior to diagnosis of the specific accident classification, which tend to prevent formation of voids, include:

   1. Verify that reactor trip and safety injection have occurred.

   2. Verify that residual heat is being dissipated; that is, reactor coolant temperature is not increasing.

   3. Verify that feedwater is being supplied to the steam generators.

   4. Operator action should be taken to maintain a water level in the pressurizer by charging and emergency makeup control, dissipate residual heat through the steam generators and maintain an indicated water level in all steam generators not directly affected by the accident.

For some LOCA cases, no operator action will prevent the formation of voids in the primary coolant system. The emergency safeguards system was designed to recover and cool the core following various degrees of primary coolant system voiding, depending on the break size and location.

2.c From the review of the TMI-2 events, all operators have been instructed as to the importance of keeping Reactor Coolant System pressure greater than the saturation temperature of the hottest spot to prevent void formation. This can either be accomplished by increasing pressure or decreasing temperature. This review also included the necessity of adequate feedwater level in Steam Generators for heat removal purposes.

All applicable operating procedures are under review to see that these instructions are contained and/or adequately stressed. The procedure review and any subsequent revisions will be followed with training sessions for all operators for explanation of the procedures in light of the events that occurred at TMI-2. Further information on this matter is contained in response to Item 7.b.

Procedure revision and training sessions will be accomplished before July 1, 1979.

NRC ACTION

3. For your facilities that use pressurizer water level coincident with pressurizer pressure for automatic initiation of safety injection into the reactor coolant system, trip the low pressurizer level setpoint bistables such that, when the pressurizer pressure reaches the low setpoint, safety injection would be initiated regardless of the pressurizer level. The pressurizer level bistables may be returned to their normal operating positions during the pressurizer pressure channel functional surveillance tests. In addition, instruct operators to manually initiate safety injection when the pressurizer pressure indication reaches the actuation setpoint whether or not the level indication has dropped to the actuation setpoint.

AEP RESPONSE

Both D. C. Cook Unit's safeguards logic utilize pressurizer water level coincident with pressurizer pressure for automatic initiation of safety injection. All Operations personnel were instructed to manually initiate safety injection during normal operation when the pressurizer pressure indication reaches the actuation setpoint, regardless of pressurizer level, on April 7, 1979. Following a review of the system to ensure that no other functions were associated with the low pressurizer level setpoint bistables, these bistables were placed in the trip position on April 26, 1979, on Unit 2. This was not done on Unit 1 as it is presently in refueling; however, all procedures for both units were revised to reflect this change and Unit 1 bistables will be placed in the trip position prior to entering the appropriate operating mode.

Procedural revisions include returning all the bistables to normal operating positions during channel functional surveillance testing of the pressurizer pressure channels. In addition, Technical Specifications on Unit 2 require that the low pressurizer pressure setpoint for safety injection be set at ≥ 1900 psig and the P-11 permissive be set at ≥ 1915 psig. With a 1% lockup on the P-11 bistables, the reset point for this bistable during unit cooldown is 1907 psig. This allows only a 7 psig increment, with no allowance for bistable drift, for manually blocking safety injection during cooldown. As we consider this situation unacceptable for Unit 2, the pressurizer level bistables will be returned to their normal operating position at a pressurizer pressure of ≤ 2000 psig during unit cooldown, until a change in the P-11 setpoint Technical Specification can be obtained.

These immediate changes are considered only a short term solution as the resulting condition would enable a single pressurizer pressure transmitter failure to initiate safety injection. Studies are presently underway to evaluate long term solutions to the problem, including logic revisions to initiate safety injection with a 2/3 pressurizer pressure signal rather than the existing 1/3 logic caused by tripping the low pressurizer level bistables.

NRC ACTION

4. 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.

AEP RESPONSE

Phase A containment isolation is automatically initiated by a safety injection signal derived from the Reactor Protection System and/or Containment Pressure - High at 1.2 psi. Phase A isolation can be manually initiated at any time. Isolation valves cannot be reopened by manipulation of control switches while the isolation actuation is tripped. The Phase A containment isolation actuation may be reset by the Operator by operation of the "Reset" pushbutton on the control panel. Resetting the containment isolation will not, by itself, reopen valves closed by the containment isolation actuation.

After the containment isolation actuation has been reset, a valve may only be reopened by first operating its control switch to the "closed" position and then to the "open" position.

Containment Isolation Phase A may be initiated by the operator at any time whether it had been previously reset or not.

Containment Isolation Phase B is initiated by containment pressure greater than 3 psi. All lines entering or leaving the containment are isolated except those required for reactor core cooling. Valves closed by Phase B isolation operate in the same manner as those closed for Phase A isolation. Further information is contained in our responses to items 7.a and 9.

Further review of containment isolation is underway since, in light of certain events at TMI-2, it may be desirable to operate additional equipment. This operation may not be possible now because either the equipment itself or a supporting system for that piece of equipment is isolated on Phase A or B.

NRC ACTION

5. 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.

AEP RESPONSE

The auxiliary feedwater system at the Donald C. Cook Nuclear Plant is automatically initiated upon receipt of an actuation signal that requires it to function. Therefore, action item 5 is not applicable.

NRC ACTION

6. 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.

AEP RESPONSE

6.a. Each Cook unit has three pressurizer power operated relief valves (PORV). Each of these three-inch valves can be isolated with a normally open three-inch motor operated block valve in series with it. The power operated relief valves and the motor operated block valves have individual control switches on the control panels in the Control Room which provide manual operation and also indication of open and close by limit switches.

The lines with each pair of valves come together into a common six-inch header where discharge temperature is measured. Temperature is displayed on the control panel in the Control Room and an alarm sounds on high temperature. This six-inch line is headered together with lines from the three safety valves which also have similar indications and alarm on discharge temperature.

The power operated relief valves and safety valves discharge through a common 12-inch header to the Pressurizer Relief Tank. This tank is monitored on the control panels in the Control Room for liquid temperature, pressure and level. Alarms are provided for excessive temperature, excessive pressure and high and low level. In addition, if a pressurizer power operated relief valve opens, the condition is alarmed.

6.b. All of these multiple indications of a PORV in an open position are currently in our procedures and Operators are directed to close the three-inch motor operated block valves.

NRC ACTION

7. 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 for two loop plants and at least two RCPs shall remain operating for 3 or 4 loop plants 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 along, but to also examine pressurizer pressure and other plant parameter indications in evaluating plant conditions, e.g., water, inventory in the reactor primary system.

AEP RESPONSE

7.a. Operators have always had instructions not to override automatic actions of engineered safety features until the cause has been determined, evaluated and plant conditions have stabilized.

Further, on January 4, 1979 in response to the NRC letter of November 28, 1978, we provided the results of our review of all safety actuation signal circuits which incorporate an override feature and provided additional steps taken to prevent inadvertent overriding of safety actuation circuits. Our response of January 4, 1979 (AEP:NRC:00114) describes these actions in more detail and subsequent conversations with members of the NRC Staff have adequately addressed the concerns expressed in the November 28, 1978 letter which also apply here. The response to Item 2 of Bulletin 79-06A reiterated the specifics of reset feature for Containment Phase A and Phase B isolation. Our January 4, 1979 letter describes the reset feature for all those ESF systems which incorporate it. It is noted that when switchover from the injection mode to the recirculation mode is required, following a design basis LOCA, the SI signal must be reset before this can be accomplished. Emergency Operating procedures provide adequate instruction to take these actions.

7.b. From the review of the TMI-2 events, all operators have been instructed not to terminate SI without reviewing multiple parameters.

All applicable procedures are under review and revisions will provide specific instructions for terminating SI, such as Reactor Coolant System pressure, 50^{o} sub-cooling of liquid below saturation temperature, adequate steam generator level and adequate pressurizer level. Procedure revision will be followed by a training session for all operators for explanation of the revised procedures.

These procedure reviews and training sessions will be accomplished before July 1, 1979.

7.c. From the review of the TMI-2 events, operators have been instructed that it is desirable to keep one or two RCP's operating if possible. All applicable operating procedures are under review and revisions included in these procedures will provide specific instructions as to pressure temperature relationship, system pressure, cooling water availability and pump vibration. This will be covered in greater detail in training sessions prior to July 1, 1979.

7.d. From the review of the TMI-2 events, all operators were instructed that depending on system conditions, a steam bubble could be formed elsewhere in the system which would displace water into the pressurizer. Under these conditions

pressurizer level indication would not be a true indication of total system inventory. Instructions are provided to compare all hot leg and cold leg temperatures to standard steam tables.

A curve has been developed from the steam tables with a line 50^{o} below the saturated pressure curve. This curve has been issued to the operators as a guide to determine system conditions.

This will be covered in greater detail in training sessions prior to July 1, 1979.

NRC ACTION

8. 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.

AEP RESPONSE

The valve position requirements for the safety-related systems at the Cook Plant were reviewed in light of the TMI-2 incident as follows:

EMERGENCY DIESEL GENERATORS

The Emergency Diesel Generators and all supporting systems are run frequently for required surveillance. Successful Emergency Diesel Generator operation is a clear indication that valves in the flow paths of all the supporting systems are properly aligned. Each Emergency Diesel Generator is checked for automatic starting and acceptance of blackout and ECCS loads during each refueling. The Technical Specification requirements provide further assurance that the Emergency Diesel Generators are operable. We consider this to be a properly aligned and proven system.

AUXILIARY FEEDWATER SYSTEM

This system has been thoroughly reviewed in all aspects such as automatic actuation requirements, flow requirements, power supplies, steam supply, valve positions and operable flow paths. All manual valves in the flow path are either locked or sealed in proper positions. The positions of all power operated valves are indicated to the Operator in the Control Room.

The auxiliary feedwater system is used frequently and Operators are completely familiar with its operation. This system is used during normal unit startup until main feedwater becomes available with steam being supplied from the Steam Generators to the Main Feedwater Pump Turbines. The auxiliary feedwater system is then placed in its Automatic Emergency Standby Mode. Upon a trip the auxiliary feedwater system is automatically actuated and its flow is required to bring the unit to the shutdown condition. Technical Specification requirements further assure that the auxiliary feedwater system is operable. We consider this to be a properly aligned and proven system.

ESSENTIAL SERVICE WATER AND COMPONENT COOLING WATER SYSTEMS

These are constant in-service operating systems. All possible flow paths are periodically used. The positions of all power operated valves are indicated to the Operator in the Control Room. We consider these to be properly aligned and proven systems.

HIGH HEAD SAFETY INJECTION

This system serves a dual function. For normal operation this system provides the normal Charging and Reactor Coolant Pump seal injection flow. Under an emergency condition this system provides the High Head Safety Injection flow with boron injection to the Reactor Coolant System. There are four manual flow distribution valves in this system that are pre-set and mechanically locked in position with valve operators removed. The positions of all power operated valves are indicated to the Operator in the Control Room. Technical Specification requirements further assure that this system is operable. We consider this to be a properly aligned and proven system.

INTERMEDIATE HEAD SAFETY INJECTION, LOW HEAD SAFETY INJECTION (RHR) AND CONTAINMENT SPRAY SYSTEM

These systems have been totally reviewed and verified that the valve alignment as required by procedure is correct. The actual valve alignment was then checked against the procedure and no errors were found. All manual flow path valves are either locked or sealed in the required position. This alignment check was only made on Unit 2 which is operating at power. On Unit 1, these systems will be aligned according to procedure prior to returning the Unit to service following refueling. Technical Specification requirements further assure that these systems are operable. We consider these to be fully checked and operable systems.

All surveillance test procedures for safety related systems require verification when a valve is taken out of normal position for test purposes and verification that it is returned to normal position at the completion of the procedure.

When a piece of safety related equipment is returned from maintenance, it is functionally tested and documented.

All power operated valves in the safety related systems are equipped with status lights and these lights are arranged in groups on the control panels. All normally lit lights are identified so that by a quick survey of the status light groups any mispositioned valve is immediately identified. This is a required check at the start of each shift with a test that all unlit lights have the ability to light.

NRC ACTION

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, 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.

AEP RESPONSE

Our current procedures and operating modes for systems which handle potentially radioactive gases and liquids between the containment and auxiliary building are under review.

In the Cook Plant there are several types of Containment Isolation systems, each with different operating features as described below:

a) Containment Isolation - Phase A

   1. Phase A isolates non-critical lines entering or leaving the Containment.

   2. Automatic initiation of Phase A isolation is caused by a Safety Injection signal derived from the Reactor Protection System and/or Containment Pressure-High at 1.2 psi.

   3. Phase A isolation can be manually initiated from the main control room at any time.

b) Containment Isolation - Phase B

   1. Phase B isolates Non-Essential Service Water to the containment coolers and Component Cooling Water to the Reactor Coolant Pump motor oil coolers.

2. Phase B is automatically initiated from a containment Pressure-High-High signal which is caused by containment pressure greater than 3 psi.

3. Phase B isolation can be manually initiated from the main control room at any time.

c) Containment Ventilation Isolation

1. This isolates all containment ventilation openings which includes purge supply and exhaust and containment pressure relief lines.

2. Containment Ventilation Isolation is initiated automatically by:

   a. Safety Injection Signal derived from the Reactor Protection System and/or Containment Pressure-High at 1.2 psi.

   b. High Containment Radiation derived from any of the following monitors:

      i) Containment Air Particulate Monitor
      ii) Containment Radiogas Monitor
      iii) Containment Area Monitor

3. Containment Ventilation Isolation can be manually initiated at any time from the main control room.

Valves closed by actuation of the above isolation circuits will remain closed if the isolation circuit is blocked or reset by the Operator. The valves may be reopened by manual operation of the valve control switch after isolation actuation is reset. During standing isolation actuation, the closed valve cannot be reopened by manipulation of the control switch. All isolation valves operate in this manner except those specifically identified in the systems listed below.

The specific response to 9.a and 9.b is contained in the following list of systems which could transfer potentially radioactive gases or liquids out of the containment.

1. Steam Generator Blowdown System

9.a This system is not interlocked to automatically isolate on a high radiation indication inside containment.

9.b This system is isolated on Phase A containment isolation and/or high radiation in the blowdown itself.

2. Containment Sump Pumps

9.a The containment sump pump header is not interlocked to automatically isolate on a high radiation indication inside containment.

9.b The containment sump pump header is isolated on Phase A Isolation. In addition, the containment sump pumps are tripped and locked out on closure of the isolation valves. The sump pumps will not automatically restart after the sump header discharge valves have been reopened by operator action without additional action required to reset the pump lockout and restart the pump.

3. Pressurizer Relief Tank Discharge

9.a The pressurizer relief tank discharge line is not interlocked to automatically isolate on a high radiation indication inside containment.

9.b The discharge line is isolated on Phase A containment isolation.

4. Letdown

9.a The letdown line from the Reactor Coolant System is not interlocked to automatically isolate on a high radiation indication inside containment.

9.b The letdown line is isolated on Phase A containment isolation.

5. Reactor Coolant Pump Seal Water

9.a The RCP seal water discharge line is not interlocked to automatically isolate on a high radiation indication inside containment.

9.b The discharge line is isolated on Phase A containment isolation.

6. Reactor Coolant Drain Tank (RCDT) Vents

9.a The Reactor Coolant Drain Tank vent lines are not interlocked to automatically isolate on a high radiation indication inside containment.

9.b The Reactor Coolant Drain Tank vent lines are isolated on a Phase A containment isolation.

7. Reactor Coolant Drain Tank Discharge

9.a The Reactor Coolant Drain Tank discharge line is not interlocked to automatically isolate on a high radiation indication inside containment.

9.b The Reactor Coolant Drain Tank discharge line is isolated on a Phase A containment isolation. The Reactor Coolant drain pumps are stopped by closure of either of the isolation valves.

8. Ice Condenser Fan Cooler Drains and the Containment Ventilation Unit Drains

9.a These drains are not interlocked to automatically isolate on a high radiation indication inside containment.

9.b These drains isolate on a Phase A containment isolation.

9. Miscellaneous Sample Lines (Gas and Liquid)

The following is a list of gas or liquid sample lines leaving the containment:

i) Pressurizer liquid space sample
ii) Pressurizer steam space sample
iii) Two RCS hot leg samples
iv) Four Accumulator samples
v) Reactor Coolant Drain Tank gas sample
vi) Pressurizer Relief Tank gas sample

9.a These sample lines are not interlock to automatically isolate on a high radiation indication inside containment.

9.b These sample lines are isolated on a Phase A containment isolation.

10. Containment Atmosphere Sampling System

9.a This system is not interlocked to automatically isolate on a high radiation indication inside containment.

9.b The containment gas sampling system is isolated by Phase A Containment Isolation. The gas sample valves may be reopened after Phase A Isolation without resetting the Phase A isolation actuation by placing the gas sampling selector switch in the "Bypass" position and then opening the valve by operation of the valve control switch for the purpose of obtaining a containment atmosphere sample.

11. Containment Purge System

9.a The containment purge supply, purge exhaust and pressure relief valves are interlocked to isolate on a high radiation indication inside containment. Containment Ventilation Isolation actuation, whose signals are derived from the containment radiation monitors, isolates these lines.

9.b These lines are also automatically isolated by a Safety Injection signal derived from the Reactor Protection System and/or Containment Pressure-High at 1.2 psi.

9.c Continued operability of the above mentioned features is assured through our procedures and operating modes. Compliance with our Technical Specifications further assures continued operability.

NRC ACTION

10. 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.

AEP RESPONSE

10.a. No additional testing beyond the established periodic surveillance and testing program is necessary. Periodic surveillance and testing provides a sufficient level of confidence that redundant systems will function as designed when necessary. Testing intervals are selected in recognition of the fact that certain equipment may be unavailable on a temporary basis due to maintenance, and restrictions are imposed on the length of time such equipment can be removed from service. Redundant safety related systems must be in an operable condition for compliance with Technical Specifications. Thus, when removing a safety related system from service, the operability of the redundant system is assured.

10.b. The operability of safety related systems when returned to service after maintenance or testing is verified. See the response to Item 8.

10.c. Reactor Operations Personnel are adequately notified when any safety related system equipment is removed from service or returned to service by either the Clearance Permit System or appropriate Maintenance procedures.

NRC ACTION

11. 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.

AEP RESPONSE

An administrative procedure has been issued requiring notification of the NRC Region III within one hour of any operating anomaly when the reactor is not in a controlled or expected condition of operation. A telephone line dedicated for the purpose of providing a continuous communication channel between the plant and the NRC will be installed by June 1, 1979.

NRC ACTION

12. 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.

AEP RESPONSE

Adequate Plant operating procedures for controlling the hydrogen that is produced in the containment following an accident will be provided. The design of the hydrogen recombiner system is based on Regulatory Guide 1.7, for the hydrogen produced by corrosion of the aluminum portions of the NSSS, by radiolytic decomposition of the core cooling and sump solutions, by the reaction of the zirconium fuel cladding with water and by the release of the hydrogen dissolved in the reactor coolant and contained in the pressurizer vapor space.

The hyrdrogen recombiner system consists of two redundant accident qualified electric recombiner units permanently installed within the reactor containment which are controlled from the Control Room. Plant operating instructions are under review to assure adequate post-accident containment sampling circulation, and means of hydrogen removal including venting and operation of recombiners. Technical Specifications also provide assurance that the hydrogen recombiner system is operable. Further information is contained in the response to Item 2.