Carl L. Newman
Vice President

Regulatory Docket File

Consolidated Edison Company of New York, Inc.
4 Irving Place, New York, N. Y. 10003
Telephone (212) 460-5133

November 22, 1974

Re Indian Point Unit No. 3
AEC Docket No. 50-286

Mr. James P. O'Reilly, Director
U. S. Atomic Energy Commission
Directorate of Regulatory Operations
Region 1
631 Park Avenue
King of Prussia, Penn. 19406

Dear Mr. O'Reilly

Pursuant to 10CFR50.55(e) of the Commission's Regulations, Mr. Folsom of Region 1 Regulatory Operations was notified by telephone on October 24, 1974 of a deficiency concerning residual Heat Removal (RHR) Pump No. 31 at Indian Point Unit No. 3.

The primary function of the Residual Heat Removal Pumps is to transfer heat energy from the core and Reactor Coolant System during plant shutdown and refueling operations. The residual heat removal pumps are also used to deliver water to the Reactor Coolant System from the refueling water storage tank during the injection phase of loss-of-coolant accident. After the injection phase, they are used to remove residual heat from the reactor core and containment vessel by recirculation of the containment sump water through the residual heat exchangers, in the event the recirculation pumps are unavailable.

On September 30, 1974, RHR Pump No. 31 was secured because of a leaking seal package. On October 18, 1974 as construction personnel were replacing a defective gasket which was the cause of the seal package leak, it was noted that the pump impeller retaining nut and lock nut were loose. The impeller, however, had not slipped on its shaft, nor was there any other indication that the pump would not have operated satisfactorily for an indefinite period of time. The pump was thoroughly inspected by a representative of the pump manufacturer and no other abnormalities were found. The impeller nuts were reinstalled and torqued and the defective gasket was replaced. The pump was reassembled and returned to service.

10684

Re Indian Point Unit No. 3
AEC Docket No. 50-286

The following corrective action was taken to prevent a similar occurrence:

a. RHR Pump No. 32 was disassembled and thoroughly inspected. No abnormalities were identified.

b. A review of the plant design indicates that there are no other pumps of similar construction.

Very truly yours

Carl L. Newman

Carl L. Newman
Vice President

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Central Files

POWER AUTHORITY OF THE STATE OF NEW YORK

10 COLUMBUS CIRCLE NEW YORK, N. Y. 10019
(212) 397-6200

April 26, 1979

United States Nuclear Regulatory Commission
Office of Inspection and Enforcement
Region I
631 Park Avenue
King of Prussia, Pennsylvania 19406

Attention: Mr. Boyce H. Grier, Director, Region I

Subject: Indian Point 3 Nuclear Power Plant
Docket No. 50-286
Response to IE Bulletin No. 79-07

Dear Mr. Grier:

On April 24, 1979 the Power Authority of the State of New York submitted its response to IE Bulletin No. 79-07. The enclosed computer code listings completes the information which was requested under item number 2 of the Bulletin.

Since the enclosed computer code listings are Arthur D. Little Inc. proprietary information, the Power Authority hereby requests that under the provisions of 10 CFR §2.790 the listings be withheld from public disclosure. In conformance with 10 CFR §2.790(b) there are attached two affidavits supporting the non-disclosure request, one from Arthur D. Little, Inc. and the other from United Engineers and Constructors, Inc.

Very truly yours,

Joseph R. Schmieder
Joseph R. Schmieder
Chief Engineer

A002
ccp

cc: U.S. Nuclear Regulatory Commission
Office of Inspection and Enforcement
Division of Reactor Operations Inspection
Washington, D.C. 20555
w/o att.

~~7906130122~~

STATE OF NEW YORK)
) ss.:
COUNTY OF NEW YORK)

A F F I D A V I T

1. I, REIMAR F. DUERR, am the Project Manager of United Engineers & Constructors, Inc. ("UE&C") for Indian Point Unit #3 owned and operated by Power Authority of the State of New York and am authorized to apply for the withholding from public disclosure of proprietary information.

2. This Affidavit is submitted under the provisions of 10 C.F.R. Section 2,790.

3. Pursuant to 10 C.F.R. Section 2.790 (b) (4), the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld:

a. The information sought to be withheld has been released by Arthur D. Little, Inc. ("ADL") to UE&C through a Licensing Agreement dated May 20, 1970 which imposes confidentiality requirements and other restrictions on UE&C.

b. The information sought to be withheld is of a type customarily held in confidence.

c. ADL has requested that UE&C take steps to prevent public disclosure of this information and has advised UE&C for the following reasons that there is a rational basis for such withholding:

(i) The use of such information gives ADL an advantage over its competitors. Withholding such information from disclosure protects ADL's competitive position.

(ii) The information is marketable to licensees. Withholding the information from disclosure protects ADL's ability to sell the information to licensees.

(iii) The information sought to be withheld is not available in public sources, but is only available to licensees through licensing agreements.

(iv) the disclosure of the information sought to be withheld is likely to cause substantial harm to the competitive position of ADL, based on:

(a) The value of the information to ADL.

(A) The use of the information gives ADL advantages over its competitors. It reduces costs by increasing the productivity of UE&C's engineers and other personnel. Public disclosure of this information would enable competitors to exploit advantages now possessed exclusively by ADL and its licensees.

(B) The public disclosure of this information would jeopardize ADL's ability to preserve existing and procure future licensing agreements.

(b) The difficulty entailed by others in duplicating this information.

d. The information has been submitted to the Commission in confidence, and under 10 C.F.R. Section 2.790, it is to be received in confidence by the Commission.

It is therefore respectfully requested that the aforementioned information be withheld from public disclosure.

Reimar F. Duerr

Reimar F. Duerr
Project Manager
United Engineers & Constructors Inc.

Subscribed and sworn to before
me this 26th day of April, 1979.

Edgar K. Byham

EDGAR K. BYHAM
Notary Public, State of New York
No. 31-4600492
Qualified in New York County
Commission Expires March 30, 1980

A F F I D A V I T

1. I am the Vice President and General Counsel of Arthur D. Little, Inc., ("ADL") and am responsible for reviewing proprietary information sought to be withheld from public disclosure, and am authorized to apply for its withholding on behalf of ADL.

2. This Affidavit is submitted under the provisions of 10 C.F.R. Section 2.790 and in conjunction with the application by the licensee required to report pursuant to IE 79-07.

3. I have personal knowledge of the criteria and procedures used by ADL to designate information as a trade secret or privileged or confidential commercial or financial information.

4. Pursuant to 10 C.F.R. Section 2.790(b)(4), the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld:

   a. The information sought to be withheld is owned by ADL and consists of source program listings in whole or in part of a computer program named ADLPIPE. It has been released to several clients through licensing agreements, all of which impose confidentiality requirements on the licensee and prohibit its use other than as authorized, but otherwise has been held in confidence by ADL.

   b. The information sought to be withheld is of a type customarily held in confidence by ADL. Among these types of information are computer program source listings which are made available to clients under the previously mentioned licensing agreements and computer program source listings which must be protected from unauthorized alterations.

c. There is a rational basis for ADL's holding this information in confidence:

(i) The use of such information gives ADL an advantage over its competitors. Withholding such information from disclosure protects ADL's competitive position.

(ii) The information is marketable to licensees. Withholding the information from disclosure protects ADL's ability to sell the information to licensees.

d. The information has been submitted to the Commission in confidence, and under 10 C.F.R. Section 2.790, it is to be received in confidence by the Commission.

e. The information sought to be withheld is not available in public sources, but is only available to licensees through licensing agreements.

f. The disclosure of the information sought to be withheld is likely to cause substantial harm to the competitive position of ADL, based on:

(i) The value of the information to ADL.

A. The use of the information gives ADL advantages over its competitors. It reduces costs by increasing the productivity of ADL's engineers and other personnel. Public disclosure of this information would enable competitors to exploit advantages now possessed exclusively by ADL and its licensees.

B. Between 1968 and 1979, ADL released this information to a number of licensees pursuant to licensing agreements. The public disclosure of this information would jeopardize ADL's ability to preserve existing, and procure future, licensing agreements.

(ii) The amount of effort and money expended by ADL in developing this information. This information has been developed over a period of twelve years at a cost of several hundreds of thousands of dollars.

(iii) The difficulty entailed by others in duplicating the information. A competitor of ADL would have to expend resources comparable to those expended by ADL to duplicate the information sought to be withheld.

It is therefore respectfully requested that the aforementioned information be withheld from public disclosure.

Richard T. Murphy, Jr.

Richard T. Murphy, Jr.
Vice President and General Counsel

COMMONWEALTH OF MASSACHUSETTS)
ss:
COUNTY OF MIDDLESEX .)

Before me, the undersigned authority, personally appeared Richard T. Murphy, Jr., who, by me duly sworn according to law, deposes and says that he is authorized to execute this Affidavit on behalf of Arthur D. Little, Inc., and that the averments of facts set forth in this Affidavit are true and correct to the best of his knowledge, information, and belief.

Sworn to before me this 20th day of April, 1979.

Martha Cotton

Notary Public

MARTHA COTTON, Notary Public
My Commission Expires May 29, 1981

CENTRAL FILES

POWER AUTHORITY OF THE STATE OF NEW YORK
INDIAN POINT NO. 3 NUCLEAR POWER PLANT
P. O. BOX 215 BUCHANAN, N. Y. 10511
TELEPHONE: 914-739-8200

April 26, 1979
IP-WDH-4660

Docket No. 50-286
License No. DPR-64

Mr. Boyce H. Grier, Director
Office of Inspection and Enforcement
Region I
U. S. Nuclear Regulatory Commission
631 Park Avenue
King of Prussia, Pennsylvania 19406

Subject: I.E. Bulletin No. 79-06A

Dear Mr. Grier:

Enclosed is our detailed response to Items 1 through 12 of I.E. Bulletin 79-06A.

Very truly yours,

J. P. Bayne
Resident Manager

WDH:ms

Attachment

cc: Office of Inspection and Enforcement
U. S. Nuclear Regulatory Commission
Division of Reactor Operations Inspection
Washington, D. C. 20555

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

<u>RESPONSE</u>

a. As a result of the TMI incident, the Power Authority of the State of New York personnel have met with <u>W</u> representatives to discuss the chronological information concerning the incident and the potential consequences to Indian Point 3. Specifically on April 5, 1979, Power Authority personnel, along with other utility representatives, attended a meeting with <u>W</u> representatives to discuss the events at TMI which were known at that time. A list of specific items was proposed to be discussed at a future meeting. On April 16 through April 18, 1979 Power Authority personnel met with <u>W</u> representatives to discuss these specific items as they related to Indian Point 3. Future meetings with <u>W</u> will be scheduled as the need arises.

   I.E. Bulletins 79-05 and 05A have been issued for review by licensed operators and plant management to (1) understand the events and actions at TMI, (2) understand that the potential exists for a false pressurizer level indication in some instances and (3) analyze available parameters and indications before taking appropriate corrective action.

   The licensed operators, plant management and corporate headquarters staff also attended an NRC briefing which discussed the chronological events at TMI. This briefing was video-taped to ensure that all licensed operators, who were unable to attend, will understand the chronological events at TMI.

   The TMI incident has also been reviewed by other plant personnel including engineering and support groups and the Plant Operating Review Committee as well as the Safety Review Committee. Additional information will be appropriately issued as it is received.

b. A memorandum has been issued to all operational personnel to instruct them to (1) not override automatic action of engineered safety features unless continued operation of engineered safety features will result in unsafe plant conditions and (2) not make operational decisions based solely on a single plant parameter indication when one or more confirmatory indications are available. Additionally the Administrative Procedure for "Shift Organization Requirements" is being revised to include these instructions and will be effective by 5/1/79.

c. A review system has been established to include all licensed operators, operations supervisors and plant management to review I.E. Bulletins 79-05, 5A, 6 and 6A as well as other pertinent information relating to TMI and its potential consequences to Indian Point 3. This system requires acknowledgement of receipt and allows participation in the review to improve plant design and operating procedures. The acknowledgement and responses will be documented in the permanent plant records.

Documentation is also provided for those who have attended the NRC briefing of the incident on 4/19/79.

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. Operator 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

We have reviewed the actions required by our operating procedures concerning the recognition of the possibility of forming voids. The existing procedures are written so as to instruct the operator on what actions to take to insure the maintenance of adequate core cooling and at the present time do not address the possibility of void formation due to sustained operation below saturation conditions. We are currently revising our procedures to address this concern both in a preventive sense by emphasing the need to always maintain the core in a sub-cooled state and in an anticipative sense by instructing the operator how to deal with potential void formations. These revisions will be completed by 5/15/79.

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.

RESPONSE

On receipt of I.E. Bulletin No. 79-06A on 4/14/79, the low pressurizer level safety injection bistables were placed in the tripped position under the direction of the Shift Supervisor. This action insures that safety injection is initiated when the pressurizer pressure reaches the Safety Injection actuation setpoint regardless of the indicated pressurizer level. It is our view that the imposition of this requirement is inconsistent with maintaining plant reliability and totally unwarranted. What occurred at TMI is well known by our operators and, if a similar condition were to exist, the operators have been instructed to institute a manual safety injection. Rather than place the plant in a position where a minor voltage disturbance could cause a plant trip concurrent with a safety injection actuation, we feel it would have been more prudent to direct our energies to implementing a new logic scheme for safety injection based upon low pressurizer pressure signals. This action would relieve the dependence of automatic initiation of safety injection on low pressurizer level without sacrificing the reliability of the plant. Therefore, to this end, our engineering department in concert with our NSSS vendor (Westinghouse) has designed a modification based on low pressurizer pressure. The design will allow for on line testing to insure the operability of the system. We currently are awaiting NRC approval to implement this modification and re-establish an acceptable level of plant reliability consistent with protection system design criteria.

Additionally, as a result of your I.E. Bulletin No. 79-06A Rev. 1 dated 4/18/79 and discussions with the commission, and until the above modification is implemented, we have instituted temporary procedure changes to our surveillance test procedures. These changes allow all three pressurizer low level S.I. bistable trip switches to be placed in their normal operating position during the performance of analog channel or safeguards logic tests and to return them to the tripped condition at the conclusion of the test. Untripping of the switches and the return to the tripped condition requires signature approval of a licensed operator.

Lastly, our procedure which covers the failure of a pressurizer relief valve to reseat, requires the operator to insure that safety injection is initiated either automatically or manually should pressurizer pressure indication reach the actuation setpoint regardless of the pressurizer level indication.

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.

RESPONSE

We have reviewed our containment isolation initiation design and procedures for all lines whose isolation does not degrade needed safety features or cooling capability upon automatic initiation of safety injection. Isolation is provided by both automatic valves and manual valves. The required automatic valves will close immediately on a safety injection signal. The manual valves are closed prior to going above cold shutdown except for those that are open continuously or intermittently for normal operation as required by plant Technical Specifications. These manual valves are required to be closed following a safety injection. For a more detailed explanation of our containment isolation system including both design and operational criteria, refer to the response for Item #9.

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.

RESPONSE

No response to this item is required since the auxiliary feedwater system is automatically initiated at our facility.

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.

<u>RESPONSE</u>

The existing Alarm Response procedure for a pressurizer relief line high temperature alarm requires the operator to verify the alarm and close the power operated relief valve which is leaking or stuck open and then close its respective motor operated stop valve. Additionally, an operating procedure was issued on 4/9/79 which identifies specific plant indications to allow an operator to determine whether a pressurizer power operated relief valve or safety valve is stuck open or if a break has occurred in the pressurizer vapor space which is unisolatable. The procedure directs the operator to manually close the power operated relief valves and their motor operated stop valves when reactor coolant system pressure is reduced below the setpoint for normal automatic closure of the power operated relief valves.

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 RCP's 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 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

a) Our Administrative Procedure entitled "Shift Organization Requirements" is being revised and will be effective 5/1/79, which specifically states that all operations personnel are not to override automatic action of engineered safety features unless and only if continued operation of engineered safety features will result in unsafe plant conditions. It should be noted that we have a specific procedure to override the safeguards signal when it is determined that it was spuriously actuated and the need for the engineered safety system clearly does not exist.

Additionally, a new procedure is being written and will be effective 4/27/79 which gives specific instruction on how to reset Phase A containment isolation without causing a change in valve positions. It should be noted that resetting the Phase A signal is a distinct action which is separate from resetting the Safety Injection signal.

b,c) We feel that to incorporate the instructions, per se, into our procedures on safeguards initiation is unwarranted at this time as these instructions are not germane to the entire spectrum of accidents requiring safeguards actuation. While the actions stipulated are proper with respect to the TMI occurrence, they are not necessarily correct for incidents initiated by other malfunctions or failures. Therefore, we are presently reviewing all of our procedures to insure that the actions required by them are complete and correct. It should be noted that our NSSS supplier concurs with this approach.

In view of this, a new procedure is being written and will be effective 5/15/79 which deals specifically with conditions similar to what occurred at TMI. Also, another procedure is being prepared which deals with the removal of a bubble should one be formed in the primary system. This procedure will likewise be effective by 5/15/79.

We feel this action will insure that the health and safety of the public will not be endangered in the event of similar accident conditions as experienced at TMI.

d) Our response to this item has been adequately dealt with in our response to item number 1b.

In addition the emergency procedures contain a section indicating many additional indications which should be considered before taking appropriate operator 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.

RESPONSE

All safety-related valve positions, which are accessible during normal power operation, have been reviewed to ensure their proper position with respect to proper operation of the engineered safety features. This has been documented on the check-off list procedures. All valves were found in their proper position. Critical safety-related line-ups which are accessible during normal power operation, are routinely verified and documented to ensure the equipment is in the proper mode of operation consistent with the status of the plant and as required by Technical Specifications. Supervisory alarms, which annunciate safeguards equipment out of normal position and safeguards valves out of normal position in the control room, alert the licensed operator to critical valves and equipment that are placed off normal position.

Startup and shutdown of the plant require valve and switch positioning requirements of safety-related equipment. These operations are specified within the appropriate procedures and documented on the check-off list procedures. A detailed review of these check-off list procedures and the positive controls governing their use is being performed and will be completed by 6/1/79.

Prior to performing maintenance on safety-related equipment, the equipment must be removed from service under the direction of a licensed reactor operator and an appropriate procedure followed to perform maintenance. Following maintenance the equipment is restored to service under the direction of a licensed reactor operator. The equipment is then tested in accordance with the retest program to ensure its operability. The administrative controls governing the sequences of these events are being reviewed. This review will be completed by 6/1/79.

A licensed reactor operator must authorize any safety related equipment surveillance test to be performed. All operating surveillance tests include steps to return the equipment to a normal mode of operation. We have reviewed these steps and verified that all valves are required for proper operation of engineered safety features. Several refueling surveillance tests require the licensed operator to designate the desired equipment line-up following testing. Since the plant is in a shutdown condition, this equipment may or may not be selected to remain in service. However prior to returning the plant to service, the check-off list procedures are completed which place the plant in the safe and proper operating condition in accordance with the plant Technical Specifications.

Any manipulation of safety-related equipment for testing, maintenance and surveillance checks is done under the direction of a licensed reactor operator.

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.

RESPONSE

A review of our operating modes and procedures for all systems designed to transfer radioactive gases and liquids from the vapor containment building show that no undesired pumping, venting or other releases of radioactive liquids or gases will occur inadvertently. These systems operate under one or more of the following restrictions:

1) Automatically interlocked with high radiation signals
2) Automatically isolated by a containment Phase A isolation signal
3) Automatically isolated by a containment Phase B isolation signal
4) Automatically isolated by a containment ventilation isolation signal
5) Manually isolated by procedure and are closed prior to going above cold shutdown except for testing and required operational activities as per Technical Specifications
6) Not isolated but required for operation during an accident condition

In general, a "phase A" isolation signal will isolate all lines which are not needed for operation during an accident except for component cooling to the reactor coolant pumps and seal return from the reactor coolant pumps. A containment ventilation isolation signal will isolate the containment purge supply and exhaust system as well as the containment pressure relief system.

A "phase B" isolation signal will isolate the component cooling to the reactor coolant pumps and the seal return for the reactor coolant pumps.

Phase A isolation and containment ventilation isolation will occur automatically whenever a safety injection signal is initiated. Additionally, both can be initiated manually from the control room. Phase B will be initiated automatically on a high-high containment pressure or can be initiated manually from the control room.

Resetting of the SI signal is required by PEP-ES-1 (during an actual SI condition). The resetting of the SI signal will not affect the position of any of the valves which are controlled by a phase A, phase B or containment ventilation isolation signal nor will it cause any other valves to change position which could inadvertently transfer radioactive fluid out of the primary containment.

It is important to note that resetting of the phase A isolation signal could result in a repositioning of some of the valves such that undesired release of radioactive liquid or gas from the vapor containment could occur. All procedures are being reviewed and if any procedure does require resetting the phase A signal, cautions will be inserted and the procedure revised to allow for resetting of a phase A signal without any undesired release of radioactive gas or liquid from the vapor containment building. These procedures will be effective by 4/27/79.

Resetting of a phase B signal does not automatically change the position of the valves as they are all motor operated valves and must be repositioned with their individual control switches.

Resetting of the containment ventilation isolation signal will not cause the purge supply and exhaust or the pressure relief valves to open. All these valves must be repositioned using their individual control switches after resetting ventilation isolation.

In as much as plant conditions could be so varied prior to an incident, we have reviewed all containment penetrations to assure they are properly addressed during accident conditions. The following is a listing of all lines penetrating containment exclusive of those required to be in service during accident conditions.

1) SJAE to Containment
2) Comp. Cooling from RCP Motor Coolers
3) Comp. Cooling to RCP
4) PRT to Gas Analyzer
5) Makeup Water to PRT
6) CVCS Letdown
7) RCP Seal Return
8) RCS Sample Line
9) Accumulator Sample
10) Primary System Vent Header and H_2 Supply
11) RCDT to Gas Analyzer
12) RCDT Pumps to Holdup Tank
13) RCP Cooling Water Out
14) Excess Letdown Ht.Ex.Cooling Wtr. In
15) Excess Letdown Ht. Ex. Cooling Wtr. Out
16) Containment Sump Pump Discharge
17) Containment Air Sample In

18) Containment Air Sample Out
19) Stm. Gen. Blowdown
20) Stm. Gen. Blowdown Sample
21) Purge Supply Duct
22) Purge Exhaust Duct
23) Containment Pressure Relief
24) Prz. Stm. Space Sample
25) Prz. Liquid Space Sample
26) O_2 Supply to Containment
27) H_2 Supply to H_2 Recombiner
28) Instrument Air/P.A. Venting Supply Line
29) 80 Foot Air Lock Solenoids
30) 95 Foot Air Lock Solenoids
31) Fuel Transfer Canal Gate Valve
32) High Head SI Pump Test Line
33) High Head SI Pump Test Line
34) Aux. Steam Supply to Containment
35) Aux. Steam Condensate Return
36) Recirculation Pump Sample Line
37) Recirculation Pump Sample Line
38) Station Air Supply to Containment
39) Station Air Supply to Containment
40) Post Accident Containment Sampling from 33 FCU
41) Post Accident Containment Sampling from 34 FCU
42) Post Accident Containment Sampling from 31 FCU
43) Post Accident Containment Sampling from 32 FCU
44) Post Accident Containment Sampling from 35 FCU
45) Post Accident Containment Sampling Return A
46) Post Accident Containment Sampling Return B
47) RHR Suction
48) RHR Sample
49) RHR Sample
50) RHR Sample
51) RHR Suction from Containment Sump
52) RHR Suction from Containment Sump
53) RHR Supply to High Head Pumps
54) RHR Supply to High Head Pumps
55) Containment Spray to RWST
56) BIT Bypass
57) Accumulator N2 Supply
58) Accumulator N2 Supply
59) Dead Weight Calibrator
60) O2 Supply to Containment
61) H2 Supply to H2 Recombiner
62) PACVS Exhaust Line
63) N2 Supply to PRT

64) H_2 Supply to H_2 recombiner, (4 lines)
65) Post Accident Venting Exhaust Line
66) Containment Leak Test Instrument Line
67) Containment Leak Test Air Line
68) Residual Heat Removal Loop Out Between Valves 730 and 732 (A-106)
69) Fuel Transfer Tube Blind Flange.
70) Containment Spray Header 31 Valves 868A and S-133, S-134
71) Containment Spray Header 32 Valves 868B, S-135 and S-136
72) Containment Air Sample In Between Valves 1234 and 1235
73) Containment Air Sample Out Between Valves 1236 and 1237
74) Air Ejector Discharge to Containment Between PCV-1229 and 1230
75) Purge Supply Duct Between Valves 1170 and 1171
76) Purge Exhaust Duct Between Valves 1172 and 1173
77) Pressure Relief Line Between 1190 and 1191 and 1191 and 1192
78) Supply to Post Accident Sampling Lines Between 1890F and 1890C. (1891A)
79) Post Accident Return Sample Lines Between 1890G and 1890H. (1891B)
80) O2 to Containment - Downstream of IV-2A and 1882A
81) N2 Supply to RCDT (Valve 1668)
82) 518 N2 Supply
83) 741 RHR Pumps to RHR Heat Exchanger
84) 867A To Containment Spray Header
85) 867B To Containment Spray Header
86) 1616 N2 Supply to RCDT
87) IA-39 Instrument Air To Containment
88) Equipment door
89) 80' Air Lock door (at least one)
90) 95' Air Lock door (at least one)
91) Fan Cooler Units inlet drain valves (5)

The above listed valves and/or systems which receive a phase A isolation signal are:
Item Number 1, 4-6, 8-12, 14-20, 24-30

The above listed valves and/or systems which receive a phase B isolation signal are:
Items Number 2, 3, 7, 13

The above listed values and/or systems which receive a containment ventilation isolation signal are:
Items Number 21, 22, 23

Additionally, the above valves and/or systems which are interlocked to prevent transfer when high radiation indication exists are:
Items Number 19, 20, 21, 22, 23

The above listed manual valves and/or systems which are procedurally closed or isolated prior to exceeding cold shutdown are:
Items #31-81, 88-91

Items #82-87 are containment isolation check valves which are checked for operability via the periodic testing as required by the Technical Specifications.

All other valves and/or systems not listed are required for operation during an accident condition and thus do not pose a threat to allow undesired pumping, venting or other release of radioactive liquids and gas being released from the vapor containment building.

Lastly, the basis on which continued operability of the above features (automatic isolation signals) is assured via the periodic testing program as is required by the facilities technical specifications.

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.

<u>RESPONSE</u>

a) Verification, by test or inspection, of the operability of redundant safety-related systems is performed prior to the removal of any safety-related system from service. Said testing or operability inspection is likewise performed following the removal of any safety-related system from service at the frequency specified in the Technical Specifications. Our administrative procedure entitled "Shift Organization Requirements" is being revised to formally document this responsibility to the Shift Supervisor. This procedure will be revised by 5/1/79. Further, a new procedure providing formal guidance on removing specific individual pieces of safeguards equipment from service for maintenance and documentation of same is being prepared and will be issued by 7/1/79.

Testing of redundant safety-related equipment prior to removing safety-related equipment from service for testing is redundant. This would require removing all units of a particular type of equipment in a safety-related system for testing prior to testing one of them. When safety-related equipment is tested, operators have manual control of the equipment. Should an emergency need arise for this equipment, it can be placed in service immediately.

b) Following maintenance on safety-related equipment, the equipment is prepared for service through the use of the work permit. The equipment is then tested in accordance with the retest program to ensure its operability. It is always the responsibility of the licensed reactor operators on watch to ensure that this safety-related equipment is properly returned to service within the specified Technical Specification time limit. The administrative procedure entitled "Shift Organization Requirements" is being revised to formally document this responsibility to the Shift Supervisor. This procedure will be effective by 5/1/79.

Following testing on safety-related equipment, all operating surveillance tests include steps to return the equipment to a normal mode of operation. Several refueling surveillance tests require the licensed reactor operator to designate the desired equipment line-up following testing. Since the plant is in a shutdown condition, this equipment may or may not be selected to remain in service. However, prior to returning the plant to service, the check-off list procedures are completed which place the plant in the safe and proper operating condition in accordance with the plant Technical Specifications.

c) The removal of safety-related equipment from service for maintenance or surveillance testing must be approved on an appropriate form or procedure by a licensed ractor operator on watch. Following the required test or maintenance on the equipment, the form or procedure is signed off again by a licensed operator. This procedure ensures that the equipment is operable following return to service.

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.

RESPONSE

Administrative Procedure AP-8 titled "Reporting of Significant Occurrences" has been reviewed. This procedure has been clarified to require notification of the NRC within one (1) hour of the time the reactor is not in a controlled or expected condition of operation. The communication channel will be continuously maintained when established with the NRC. This procedure will become effective by 4/27/79.

The Emergency Plan Procedures Document for Unit No. 3 contains a list of notifications, which includes the NRC, in the event of a site or general emergency. It has been previously demonstrated during official tests of the implementation of this Emergency Plan that the NRC has been notified well within one (1) hour of the initation of the test.

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.

RESPONSE

We have reviewed our operating modes and procedures to deal with hydrogen gas that may be generated and remain inside the primary system or be released to the Vapor Containment Building.

There are several methods for hydrogen removal in the primary system. They are (1) venting the pressurizer vapor space via the pressurizer power operated relief valves to the pressurizer relief tank, (2) transferring water to the volume control tank via the letdown line and (3) venting the primary system in the event of a LOCA. Existing procedures concerning the possible methods of dealing with significant hydrogen generation have been reviewed. These procedures will be updated and become effective by 5/15/79.

In addition a new procedure is being written for removing hydrogen gas in the primary system as a result of TMI and discussion with our NSSS vendor. This procedure will be effective by 5/15/79.

To remove hydrogen gas from the Containment, there are two (2) installed systems. The primary method for hydrogen removal is through the use of two hydrogen recombiners. The recombiners, through oxidation of the hydrogen, form water vapor which is subsequently condensed by the recirculation fan units. The recombiners would be put into service before the lower flammability limit of hydrogen is reached and are capable of maintaining the hydrogen concentration at or below 2% of the containment volume.

The secondary method for hydrogen removal is through the use of the post accident containment venting system which provides a backup system to the hydrogen recombiners. The gases containing hydrogen from containment are vented through charcoal filters at a controlled rate to the vent stack.