TERA

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

December 31, 1979

Docket No. 50-271

Mr. Robert H. Groce
Licensing Engineer
Yankee Atomic Electric Company
20 Turnpike Road
Westboro, Massachusetts 01581

Dear Mr. Groce:

SUBJECT: NRC STAFF EVALUATION OF YANKEE ATOMIC ELECTRIC COMPANY RESPONSES TO IE BULLETIN 79-08 FOR VERMONT YANKEE NUCLEAR POWER STATION

We have completed our review of the information that you provided in your letters dated April 27 and May 4, 1979 in response to IE Bulletin 79-08 for the Vermont Yankee Atomic Nuclear Power Station. We have also completed our review of the supplemental information that you provided in your letter of August 9, 1979.

We have concluded that you have taken the appropriate actions to meet the requirements of each of the eleven action items identified in IE Bulletin 79-08. A copy of our evaluation is enclosed.

As you know, NRC staff review of the Three Mile Island, Unit 2 (TMI-2) accident is continuing and other corrective actions may be required at a later date. For example, the Bulletins and Orders Task Force is conducting a generic review of operating boiling water reactor plants. Specific requirements for your facility that result from this and other TMI-2 investigations will be addressed to you in separate correspondence.

Sincerely,

T. Rooney, for

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Enclosure:
NRC Staff Evaluation

cc w/enclosure:
See next page

1742 093

P 8001110 459

cc:

Ms. J. M. Abbey
Vermont Yankee Nuclear Power
Corporation
77 Grove Street
Rutland, Vermont 05701

Mr. Donald E. Vandenburgh,
Vice President
Vermont Yankee Nuclear Power
Corporation
Turnpike Road, Route 9
Westboro, Massachusetts 01581

John A. Ritsher, Esquire
Ropes & Gray
225 Franklin Street
Boston, Massachusetts 02110

Laurie Burt
Assistant Attorney, General
Environmental Protection Division
Attorney General's Office
One Ashburton Place, 19th Floor
Boston Massachusetts 02108

Richard E. Ayres, Esquire
Natural Resources Defense Counsel
917 15th Street, N. W.
Washington, D. C. 20005

Honorable M. Jerome Diamond
Attorney General
State of Vermont
109 State Street
Pavilion Office Building
Montpelier, Vermont 05602

Anthony Z. Roisman
Natural Resources Defense Council
917 15th Street, N. W.
Washington, D. C. 20005

John R. Stanton, Director
Radiation Control Agency
Hazen Drive
Concord, New Hampshire 03301

John W. Stevens
Conservation Society of
Southern Vermont
P. O. Box 256
Townshend, Vermont 05353

Dr. Mars Longley, Director
Occupational & Radiological Health
10 Baldwin Street
Montiplier, VT 05602

New England Coalition on Nuclear
Pollution
Hill and Dale Farm
West Hill - Faraway Road
Putney, Vermont 05346

Public Service Board
State of Vermont
120 State Street
Montpelier, Vermont 05602

W. F. Conway, Plant Superintendent
Vermont Yankee Nuclear Power
Corporation
P. O. Box 157
Vernon, Vermont 05354

Mr. Charles Sheketoff
Assistant Director
Vermont Public Interest
Research Group, Inc.
26 State Street
Montpelier, Vermont 05602

Brooks Memorial Library
224 Main Street
Brattleboro, Vermont 05301

EVALUATION OF LICENSEE'S RESPONSES

TO

IE BULLETIN 79-08

VERMONT YANKEE NUCLEAR POWER CORPORATION

VERMONT YANKEE NUCLEAR POWER STATION

DOCKET NO. 50-271

1742 095

Introduction

By letter dated April 14, 1979, we transmitted Office of Inspection and Enforcement (IE) Bulletin 79-08 to Vermont Yankee Nuclear Power Corporation (VYNPC or the licensee). IE Bulletin 79-08 specified actions to be taken by the licensee to avoid occurrence of an event similar to that which occurred at Three Mile Island, Unit 2 (TMI-2) on March 28, 1979. By letters dated April 27 and May 4, 1979, VYNPC provided responses to Action Items 1 through 10 of IE Bulletin 79-08 for the Vermont Yankee Nuclear Power Station (VYNPS). By letter dated May 18, 1979, VYNPC provided the response to Action Item 11 of IE Bulletin 79-08.

The NRC staff review of the VYNPC responses led to the issuance of requests for additional information regarding the VYNPC responses to certain action items of IE Bulletin 79-08. These requests were contained in a letter dated July 20, 1979. By letter dated August 9, 1979, VYNPC responded to the staff's requests for additional information.

The VYNPC responses to IE Bulletin 79-08 provided the basis for our evaluation presented below. In addition, the actions taken by the licensee in response to the bulletin requirements and subsequent NRC requests were verified through onsite inspections by IE inspectors.

Evaluation

Each of the 11 action items requested by IE Bulletin 79-08 is repeated below followed by our criteria for evaluating the response, a summary of the licensee's response and our evaluation of the response.

1. Review the description of circumstances described in Enclosure 1 of IE Bulletin 79-05 and the preliminary chronology of the TMI-2 March 28, 1979 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 trains of a safety system 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; and (3) 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 5a of this bulletin); 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.

The licensee's response was evaluated to determine that (1) the scope of review was adequate, (2) operational personnel were properly instructed and (3) personnel participation in the review was documented in plant records.

The licensee's response dated April 27, 1979 described a special lecture series on this subject for all licensed operators and plant management which was prepared, presented and documented. As more information on the TMI-2 accident became available, it was factored into the ongoing operator training program. By a telephone conversation on November 7, 1979, the licensee confirmed that the scope of the review on which the lecture series was based addressed all the issues required by Action Item 1.

We conclude that the licensee's scope of review, instructions to operating personnel and documented participation satisfies the intent of IE Bulletin 79-08, Item 1.

2. Review the containment isolation initiation design and procedures, and prepare and implement all changes necessary to initiate 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.

The licensee's response was evaluated to verify that containment isolation initiation design and procedures had been reviewed to assure that (1) manual or automatic initiation of containment isolation occurs on automatic initiation of safety injection and (2) all lines (including those designed to transfer radioactive gases or liquids) whose isolation does not degrade cooling capability or needed safety features were addressed.

The licensee's April 27, 1979 response noted that a review of the primary containment isolation design and procedures had been completed. This review verified that a safety injection signal will automatically initiate containment isolation of all valves whose isolation does not degrade needed safety features or cooling capability. The licensee also described the various containment isolation features and by letter dated August 9, 1979 provided specific justification based on safety requirements for certain lines not isolating on receipt of a high drywell pressure signal. No changes to design or procedures were reported by the licensee.

We conclude that the licensee's review of containment isolation initiation design and procedures satisfies the intent of IE Bulletin 79-08, Item 2.

3. Describe the actions, both automatic and manual, necessary for proper functioning of the auxiliary heat removal systems (e.g., RCIC) that are used when the main feedwater system is not operable. For any manual action necessary, describe in summary form the procedure by which this action is taken in a timely sense.

The licensee's response was reviewed to assure that (1) it described the automatic and manual actions necessary for the proper functioning of the auxiliary heat removal systems when the main feedwater system is not operable and (2) the procedures for any necessary manual actions were described in summary form.

The licensee, in its response dated April 27, 1979, stated that following a loss of feedwater the reactor will scram automatically on low vessel water level, the main steam line isolation valves will close automatically and the

high pressure coolant injection (HPCI) system and reactor core isolation cooling (RCIC) system will initiate automatically and prevent fuel damage.

In addition, operating procedures are structured to prescribe manual initiation of these systems if sufficient operator response time exists before automatic action occurs. Although manual operator action was not essential, during at least one occasion (a loss of AC power with attendant loss of feedwater), station operators manually initiated the HPCI system in advance of automatic signals.

In summary, the operator actions necessary to manually initiate the HPCI system are (1) start the HPCI system auxiliary oil and vacuum pumps and (2) open the steam admission and water injection valves. The RCIC system can be manually initiated in a similar manner, and either system can be operated through the entire range of reactor pressures until shutdown cooling can be established using the residual heat removal (RHR) system. Station operators also have other alternatives which consist of manually opening one or more main steam relief valves, thereby reducing reactor pressure and permitting makeup from the condensate, low pressure coolant injection, or core spray systems.

We conclude that the licensee's procedural summary of automatic/manual actions necessary for the proper functioning of auxiliary heat removal systems used when the main feedwater system is inoperable satisfies the intent of IE Bulletin 79-08, Item 3.

4. Describe all uses and types of vessel level indication for both automatic and manual initiation of safety systems. Describe other redundant instrumentation which the operator might have to give the same information regarding plant status. Instruct operators to utilize other available information to initiate safety systems.

The licensee's response was evaluated to determine that (1) all uses and types of vessel level indication for both automatic and manual initiation of safety systems were addressed, (2) it addressed other instrumentation available to

the operator to determine changes in reactor coolant inventory and (3) operators were instructed to utilize other available information to initiate safety systems.

The licensee's August 9, 1979 response stated that the range of reactor vessel water level from below the top of the active fuel area up to the top of the vessel is covered by a combination of narrow-range and wide-range instruments. Level is indicated and recorded in the control room via nine different instruments. Of these nine instruments, three provide narrow-range indication and control (0-60 inches) from separate condensing chambers via the reactor feedwater system. This set indicates and records in the control room via two level indicators and one level recorder. The additional narrow-range level instruments are provided with independent condensing chambers which also indicate in the control room. Other safety-related systems are functions served by separate reactor water level instrumentation include the following:

Reactor Core Isolation Cooling System
High Pressure Coolant Injection System
Low Pressure Core Spray System
Residual Heat Removal/Low Pressure Injection Systems
Automatic Depressurization System
Nuclear Containment Isolation System

The primary containment isolation system initiates on low and low-low reactor water level. All other systems initiate on low-low reactor water level. In addition, the RCIC and HPCI systems shut down on high reactor vessel water level. As described previously, the HPCI system restarts automatically upon recession of water level again to the low level setpoint, while the RCIC system must be manually reset or be manually restarted. The remaining systems incorporate seal-in logic with the initial low level trip.

The remaining four level instruments all provide wide-range vessel level indication in the control room. Three share the same condensing chambers as

the feedwater level instruments while the fourth has a completely independent condensing chamber.

Other instrumentation which the operator can use to determine change in reactor coolant inventory include the following:

Drywell Pressure and Temperature
Drywell Radioactivity Levels
Suppression Pool High Temperature and Level
Safety Relief Valve Discharge High Temperature
High Feedwater Flow Rates
High Main Steam Flow
Abnormal Reactor Pressure
High Drywell Equipment and Floor Drain Sump Fill and Pumpout Rates

All types of vessel level indicators were reviewed with the station operators, and the operators were instructed to utilize all other available information to initiate safety systems.

We conclude that the licensee's description of the uses and types of reactor vessel level/inventory instrumentation and instructions to operators regarding the use of this information satisfies the intent of IE Bulletin 79-08, Item 4.

5. Review the actions directed by the operating procedures and training instructions to ensure that:

   a. Operators do not override automatic actions of engineered safety features, unless continued operation of engineered safety features will result in unsafe plant conditions (e.g., vessel integrity).

   b. Operators are provided additional information and instructions to not rely upon vessel level indication alone for manual actions, but to also examine other plant parameter indications in evalating plant conditions.

The licensee's response was evaluated to determine that (1) it addressed the matter of operators improperly overriding the automatic actions of engineered

safety features, (2) it addressed providing operators with additional information and instructions to not rely upon vessel level indication alone for manual actions and (3) that the review included operating procedures and training instructions.

The licensee, in its April 27, 1979 response, stated that training had been conducted with respect to not overriding automatic actions of engineered safety features. In addition, plant operating personnel have been specifically instructed on the potential consequences of overriding safety systems and to make a careful evaluation of all available supporting instrumentation prior to taking such action. In its letter dated August 9, 1979, the licensee stated that reviews and training conducted in April 1979 addressed consideration of all symptoms and indicators which might provide additional information to the operators which would be useful in evaluating abnormal plant conditions. The licensee also stated that all operating procedures and training instructions were included in the April 1979 review.

We conclude that the licensee's review of operating procedures and training instructions satisfies the intent of IE Bulletin 79-08, Item 5.

6. 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 start-up, 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.

The licensee's response was evaluated to assure that (1) safety-related valve positioning requirements were reviewed for correctness, (2) safety-related valves were verified to be in the correct position and (3) positive controls were in existence to maintain proper valve position during normal operation as well as during surveillance testing and maintenance.

The licensee's response dated August 9, 1979 stated that thorough review of all accessible safety-related valves, valve positioning requirements, and actual valve positions was conducted. Comprehensive valve lineup checks are required to be made by station operators following each extended plant shutdown and following major equipment maintenance. Following these, and monthly thereafter, each safety system is performance tested to further demonstrate total system operability. When plant conditions require changing the position of system valves, the new valve positions are recorded in the valve lineup file. In this manner, the file continuously reflects the current status of all station valves. A limited number of valves, specifically, (1) those which are associated with a tagging order, (2) those which are repositioned to perform a surveillance test and (3) those which are identified as routinely operated or as required on the valve lineup sheets are not updated in the valve lineup file because the positions of these valves are either recorded elsewhere or are functionally demonstrated on a continuous basis. Refinements to the valve lineup procedures to address even minor changes to valve positions which become necessary during routine operation and a formal procedure prescribing instrumentation rack valve lineups have been developed.

We conclude that the licensee's review of safety-related valve positioning requirements, valve positions and positive controls to maintain proper valve positions satisfies the intent of IE Bulletin 79-08, Item 6.

7. 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.

The licensee's response was evaluated to determine that (1) it addressed all systems designed to transfer potentially radioactive gases and liquids out of primary containment, (2) inadvertent releases do not occur on resetting engineered safety features instrumentation, (3) it addressed the existence of interlocks, (4) the systems are isolated on the containment isolation signal, (5) the basis for continued operability of the features was addressed and (6) a review of the procedures was performed.

In its April 27, 1979 response, the licensee reported that there is no high radiation signal input into the isolation logic for any of the systems designed to transfer potentially radioactive gases or liquids out of the primary containment. However, appropriate procedural precautions will be made to use existing radiation monitors to ensure that undesired pumping or venting of radioactive liquids and gases will not occur inadvertently as a result of resetting engineered safety features instrumentation.

The following transfer systems receive containment isolation signals:

Equipment and Floor Drain Sumps -
Low reactor water level/hi drywell pressure.

Standby Gas Treatment System -
Low reactor water level/hi drywell pressure/hi-low radiation in reactor building ventilation or refueling floor.

RHR Discharge to Radwaste -
Low reactor water level/hi drywell pressure.

Containment Atmosphere Dilution System -
Low reactor water level/hi drywell pressure/hi-low radiation in reactor building ventilation or refueling floor.

Containment isolation logic is tested at six-month intervals and the sensors that input to the logic are tested monthly to assure operability.

By letter dated August 9, 1979, the licensee added to this listing the main steam system (including drains) and the recirculation sample system. These systems isolate when high radiation indication exists.

Should a containment isolation occur, station operators are required to review the following radiation monitors for evidence of abnormal radiation levels before resetting the isolation signal:

Steam Jet Air Ejector Radiation Monitor
Advanced Off Gas Radiation Monitor
Area Radiation Monitors
Main Steam Radiation Monitors
Radwaste Radiation Monitors
Refuel Floor Radiation Monitors
Fuel Pool Radiation Monitors

The existence of any abnormal radiation condition would disallow resetting of the isolation signal and the appropriate operating procedures have also been revised accordingly.

We conclude that the licensee's review of systems designed to transfer radioactive gases and liquids out of primary containment to assure that undesired pumping, venting, or other release of radioactive liquids and gases will not occur satisfies the intent of IE Bulletin 79-08, Item 7.

8. 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 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.

The licensee's response was evaluated to determine that operability of redundant safety-related systems is verified prior to the removal of any safety-related system from service. Where operability verification appeared only to rely on previous surveillance testing within Technical Specification intervals, we asked that operability be further verified by at least a visual check of the system status to the extent practicable, prior to removing the redundant equipment from service. The response was also evaluated to assure provisions were adequate to verify operability of safety-related systems when they are returned to service following maintenance or testing. We also checked to see that all involved reactor operational personnel in the oncoming shift are explicitly notified during shift turnover about the status of systems removed from or returned to service since their previous shift.

The licensee's response dated April 27, 1979 indicated that the licensee had reviewed maintenance and test procedures and found that the issues of prior operability testing, post-operability testing, and explicit notification of operational personnel have been adequately addressed. Nevertheless, the licensee intensified this area through the following refinements, the completion of which was verified by telephone on November 7, 1979:

(1) The need for explicit notification of control room personnel for all Technical Specification and safety-related inspections, maintenance and surveillance was reemphasized to all station employees at a general staff meeting.

(2) The shift supervisor's log was restructured and broadened to identify on a continuing basis all equipment which is removed from service, when this equipment is returned to service and to provide a more comprehensive record of daily station activities.

(3) The plant equipment control procedure was expanded to include Technical Specification and safety-related equipment. Presently, only the former is recognized. Appropriate departmental procedure changes were also generated to include reference to, or incorporate the interface requirement of, the plant equipment control procedure. By letter dated August 9, 1979, the licensee stated that operability tests are performed immediately prior to removal of any redundant safety system from service irregardless of the Technical Specification surveillance interval. Additionally, these tests consist of total system performance demonstrations rather than only visual inspections.

We conclude that the licensee's review and modification of maintenance, test and administrative procedures to assure the availability of safety-related systems and operational personnel knowledge of system status satisfies the intent of IE Bulletin 79-08, Item 8.

9. 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.

The licensee's response was evaluated to determine that (1) prompt reporting procedures required or were to be modified to require that the NRC is notified within one hour of the time the reactor is not in a controlled or expected condition of operation and (2) procedures required or were to be modified to require the establishment and maintenance of an open continuous communication channel with the NRC following such events.

The licensee, in its April 27, 1979 and August 9, 1979 responses, reported that the appropriate plant procedure was reviewed and revised to assure NRC notification within one hour of the time the reactor is not in a controlled or expected condition or operation. The above procedure details the requirement of establishing and maintaining an open continuous communication channel with the NRC.

We conclude that the licensee's response satisfies the intent of IE Bulletin 79-08, Item 9.

10. 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.

The licensee's response was evaluated to determine if it described the means or systems available to remove hydrogen from the primary system as well as the treatment and control of hydrogen in the containment.

The licensee, in its April 27, 1979 response, stated that a review of operating modes and procedures to deal with hydrogen gas generated during an accident has been initiated. VYNPS has installed capability to control the buildup of hydrogen in both the reactor vessel and the primary containment system.

Hydrogen gas in the reactor vessel can be controlled by venting to the primary containment through either the safety/relief valves (SRV's) or the head vent, both of which can be manually activated from the control room. The SRV's will also automatically open on high pressure or due to an automatic depressurization system (part of ECCS) activation signal.

The licensee has installed a containment atmosphere dilution (CAD) system to prevent the occurrence of combustible mixtures of hydrogen and air within the primary containment following an accident. Proposed Technical Specifications to provide assurance that the CAD system will be operable in the event it is required have been submitted to the NRC. The licensee staff maintains surveillance on the CAD system in accordance with those proposed Technical Specifications and considers the system available if needed.

We conclude that the licensee's response satisfies the intent of IE Bulletin 79-08, Item 10.

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

The licensee's response was evaluated to determine that a review of the Technical Specifications had been made to determine if any changes were required as a result of implementing Items 1 though 10 of IE Bulletin 79-08.

The licensee, in its May 18, 1979 response, advised us that its review has shown that no changes to the Technical Specifications are required. the licensee also advised us that in its continuing review, should modifications to the Technical Specifications be required, they will be proposed in a timely manner. The licensee confirmed by telephone conversation on November 7, 1979 that its review has shown that no additional changes to the Technical Specifications are required.

We conclude that the licensee's response satisfies the intent of IE Bulletin 79-08, Item 11.

Conclusion

Based on our review of the information provided by the licensee to date, we conclude that the licensee has correctly interpreted IE Bulletin 79-08. The actions taken demonstrate the licensee's understanding of the concerns arising from the TMI-2 accident in reviewing their implementation on VYNPS operations, and provide added assurance for the protection of the public health and safety during the operation of VYNPS.

References

1. IE Bulletin 79-05, dated April 1, 1979.

2. IE Bulletin 79-05A, dated April 5, 1979.

3. IE Bulletin 79-08, dated April 14, 1979.

4. VYNPC letter, D. Moody to B. Grier, dated April 27, 1979.

5. VYNPC letter, D. Moody to B. Grier, dated August 7, 1979.

6. NRC staff letter, T. Ippolito to R. H. Grace, dated July 20, 1979.

7. VYNPC letter, R. Grace to B. Grier, dated May 4, 1979.