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VERMONT YANKEE NUCLEAR POWER CORPORATION

FVY 84-25

RD 5, Box 169, Ferry Road, Brattleboro, VT 05301

REPLY TO:
ENGINEERING OFFICE
1671 WORCESTER ROAD
FRAMINGHAM, MASSACHUSETTS 01701
TELEPHONE 617-872-8100

March 23, 1984

U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Attention: Office of Nuclear Reactor Regulation
Mr. Domenic B. Vassallo, Chief
Operating Reactors Branch No. 2
Division of Licensing

References:
(a) License No. DPR-28 (Docket No. 50-271)
(b) Letter, USNRC to All Operating Reactors, Generic Letter 83-28, dated July 8, 1983
(c) Letter, VYNPC to USNRC, FVY 83-117, dated November 7, 1983
(d) "BWR Scram System Reliability Analysis," W.P. Sullivan, et al, September 30, 1976 (transmitted in letter from GE to NRC, "General Electric Company ATWS Reliability Report," September 30, 1976)
(e) Letter, VYNPC to USNRC, Proposed Change No. 79 to Facility Operating License DPR-28, dated March 17, 1980
(f) Letter, USNRC to VYNPC, Amendment No. 58 to Facility Operating License DPR-28, dated November 3, 1980

Dear Sir:

Subject: Generic Letter 83-28, Generic Implications of Salem ATWS Events

By Reference (b), you requested that we address various concerns resulting from the generic implications of the Salem ATWS Events and provide you with our current conformance status, plans and schedules for any needed improvements. By Reference (c) we provided you with our initial response and stated that additional information would be forthcoming. The purpose of this letter is to provide you with the enclosed supplemental information.

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VERMONT YANKEE NUCLEAR POWER CORPORATION

We trust that this information adequately addresses the subject concerns; however, should you have any questions in this matter, please contact us.

Very truly yours,

VERMONT YANKEE NUCLEAR POWER CORPORATION

Warren P. Murphy

Warren P. Murphy
Vice President and
Manager of Operations

WPM/dm

STATE OF VERMONT)
)ss
WINDHAM COUNTY)

Then personally appeared before me, W.P. Murphy, who, being duly sworn, did state that he is Vice President and Manager of Operations of Vermont Yankee Nuclear Power Corporation, that he is duly authorized to execute and file the foregoing document in the name and on the behalf of Vermont Yankee Nuclear Power Corporation and that the statements therein are true to the best of his knowledge and belief.

Diane M. McCue

Diane M. McCue Notary Public
My Commission Expires February 10, 1987

DIANE M. McCUE
NOTARY
PUBLIC
WINDHAM COUNTY, VERMONT

VERMONT YANKEE ADDITIONAL RESPONSES TO REQUIRED ACTIONS BASED ON THE GENERIC IMPLICATIONS OF SALEM ATWS EVENTS

Item 1.1 Post-Trip Review - Program Description and Procedure

As discussed in Reference (c) of the cover letter, a program to address the intent of Items 1.1.1 through 1.1.7 was scheduled to be in place by April 1, 1984. In the interim we have instituted a Standing Order for post-trip review, as described in Reference (c).

At the present time we are finalizing new procedures for post-trip review. We anticipate that these procedures will be approved and implemented by the end of April 1984. A report describing our overall program is being developed and is expected to be finalized in September 1984. It is our intent to submit this report to the NRC in October 1984. Should our schedule for implementing the post-trip review procedures or submitting the program description change, we will notify our Project Manager.

Item 1.2 Post-Trip Review - Data Information Capability

A report describing our existing data and information capability was submitted via Reference (c). As discussed in that report, modification to our capability may result from our programs to address NUREG-0737, Supplement I; however, no other changes are being considered at this time.

Item 2.1 Equipment Classification and Vendor Interface

A. Reactor Trip System Components Identified as Safety-Related

As described in Section 3.1.2.5 of NUREG-1000, the GE Boiling Water Reactor (BWR) trip system design differs from the Pressurized Water Reactor (PWR) design. The GE reactor trip system consists of redundant plant process instrumentation that feed one out of two twice logic that initiates a reactor trip by de-energizing solenoid operated scram pilot valves which vent air from the scram valve diaphragms and insert the control rods. These components are contained within several functional systems at our plant; rather than one system called a Reactor Trip System (RTS). The systems which provide the reactor trip function are as follows:

SYSTEM NAME	DESCRIPTION
(a) Control Rod Drive	Scram valves, scram discharge volume water level sensors, backup scram valves

(b) Reactor Protection	Logic, power supplies, turbine sensors, drywell pressure sensors
(c) Neutron Monitoring	Neutron flux sensors, trips and bypasses (IRM, APRM)
(d) Nuclear Boiler	Reactor high pressure sensors, reactor pressure (600 psi bypass) sensors, main steam line isolation valve sensors
(e) Process Radiation Monitoring	Main steam line radiation sensors

It is not our intent to establish a new system called the Reactor Trip System. We believe that creating a "new" system made up of sub-sets of components from existing systems will result in confusion and necessitate the development/revision of plant procedures, manuals and drawings.

Our review to ensure that the components whose function is required to trip the reactor are identified as safety-related on documents, procedures and information systems used in the plant to control safety-related activities, will be included in our review of all safety-related systems, as discussed in our response to Item 2.2.1.

B. Reactor Trip System - Vendor Interface Program

In response to this concern, Vermont Yankee joined with 55 other utilities and formed an INPO Nuclear Utility Task Action Committee (NUTAC). This committee has developed and approved an industry-wide Vendor Equipment Technical Information Program (VETIP). This program is described in detail in a final report scheduled to be submitted to the NRC in the near future. The program promotes interaction among the major organizations involved in the generation of commercial nuclear power. Under the program, individual utilities exchange safety-related systems and component information with vendors, the NRC, INPO and other utilities. This exchange takes place via written notifications (i.e., Licensee Event Reports, NRC I&E Bulletins and Information Notices, industry newsletters, etc.), as well as industry meetings and day-to- day communication. The goal of these mechanisms is to share equipment technical information so as to improve the safety and reliability of nuclear power plants.

The primary purpose of the VETIP program is to ensure that current information and data will be made available to those personnel responsible for developing and maintaining plant instructions and procedures. It should be noted that these information systems and programs currently exist and are capable of identifying precursors to the industry that could lead

to a Salem-type event. The VETIP program formalizes the exchange of information through an industry controlled and primarily hardware oriented program that does not rely on vendor action (other than the NSSS Supplier) to provide information directly to the utilities. Instead, the VETIP program provides information developed by industry experience through Significant Event Reports (SER's), Significant Operating Experience Reports (SOER's) and other similar reports to the appropriate equipment vendor for comment before it is circulated to the utilities concerned.

Vermont Yankee has an existing vendor equipment information program with General Electric (GE), our NSSS vendor. This program consists of two major categories: (1) information regarding safety-related systems and components; and (2) technical information intended to enhance safety and non-safety-related equipment reliability and improve plant performance. The programs include, but are not limited to:

(a) <u>10 CFR 21 Reporting</u>

The General Electric Company has established a reporting system to handle safety concerns that complies with the requirements of 10 CFR Part 21.

(b) <u>Urgent Communications</u>

In addition to the 10 CFR 21 reports, a procedure for handling urgent communications to BWR owner/operators has been established for use in providing prompt notification of safety concerns. These communications usually take the form of a short letter which provides a brief explanation of the problem and advice or precautionary measures to be observed to avoid potential operational hazards. Due to their urgent nature, these communications are processed to operating plants by the most effective method (i.e., telex, telecopy, cable, special mail handling, etc.) and are normally followed up by a telephone call to a pre-designated utility individual to assure receipt of the information.

(c) <u>Service Information Letters (SIL's)</u>

These information letters usually provide recommendations for equipment modifications, plant design improvements or changes to procedures to improve plant performance.

(d) <u>Service Advice Letters (SAL's)</u>

These letters are issued by the GE Product Departments (other than the San Jose, CA based Nuclear Engineering Product Department) and are used to provide notification of product problems and for service

information on a broad range of consumer and industrial products. Those SAL's that are recognized by the issuing Product Department as applying to devices/equipment used in nuclear power plants are specially identified and are flagged for distribution to all nuclear power plants.

(e) Turbine Information Letters (TIL's)

The letters are issued by GE's Large Steam Turbine Generator Department to provide descriptions of product problems/improvement and to recommend modifications that will mitigate problems and/or improve product performance.

Our involvement in the NUTAC VETIP program will provide a mechanism to assure our receipt of all applicable information from GE. As discussed in our response to Item 2.2.2 in Reference (b), we have a program in place to assure that the above information, as well as any other information received which is pertinent to plant operation, is effectively reviewed, assessed, distributed and acted upon by appropriate personnel.

This program is defined in a plant procedure which was established in response to NUREG-0737 Item I.C.5, Operating Assessment, and incorporates the review of information for both safety-related and non-safety-related equipment and components.

We believe our existing program, as well as plans to incorporate the VETIP program (as discussed in the our response to Item 2.2.2), adequately address your concerns with respect to receipt and disposition of vendor equipment information.

Item 2.2 Equipment Classification and Vendor Interface (Programs for All Safety-Related Equipment)

Item 2.2.1 Equipment Classification

As discussed in Reference (c) of the cover letter, a program exists for determining safety classification in accordance with ANS-22, Draft No. 4, Revision 1, May 1973, as described in the NRC-approved Yankee Atomic Electric Company Operational Quality Assurance Manual (YOQAP-1-A). The general classification of structures, components, and systems is delineated in Appendix C of YOQAP-1-A, with further breakdown provided on system flow diagrams and electrical one-line diagrams.

By Reference (c), we informed you that we were reviewing the adequacy of our existing program and participating with the BWROG in the resolution of certain aspects of this item. We have since determined that, although our existing program meets all applicable criteria, programmatic enhancements should be implemented.

This enhancement will require that we re-review all safety-related systems to ensure that the components (including associated instruments, controls and electrical equipment) necessary for a particular safety-related system to perform its safety function are classified as safety-related on plant documents, procedures and information systems. It is our present intent to structure our re-review such that those systems which comprise the reactor trip function, as identified in our response to Item 2.1, will be addressed first.

We are presently developing an outline for this enhanced program and expect to have the program defined, in terms of scope and manpower needs, by September 1984. At that time we will inform you of our plans and schedule for completing this effort. It should be noted that this program will undoubtedly require an extensive allocation of engineering resources and its development/completion will have to be integrated with ongoing engineering activities.

With respect to the equipment classification program in use at Vermont Yankee for structures, systems and components Important to Safety, we are participating in the Utility Safety Classification Group and are seeking a generic resolution to the Staff's concern in this regard through the efforts of the Group. Our current position remains as stated in our original response to Item 2.2.1.6.

Item 2.2.2 Vendor Equipment Interface Program

As discussed in our response to Item 2.1.(B), we participated in the INPO NUTAC effort to develop the Vendor Equipment Technical Information Program (VETIP). A final report describing this program is scheduled to be submitted to the NRC in the near future. It should be noted that since the program was developed on a generic basis to enhance existing plant vendor information programs, certain provisions of the VETIP may not be applicable to some utilities or may require modification to fit into the plant specific equipment information program. Thus, we expect it will take 90 days from our formal receipt of the final report from INPO to determine the scope of our final program and to establish a schedule to its implementation. At that time we will inform you of our expected implementation date for this program.

Item 3.1 Post-Maintenance Testing (Reactor Trip System Components)

Items 3.1.1 through 3.1.3 were addressed in Reference (c).

Item 3.2 Post-Maintenance Testing (All Other Safety-Related Components)

Items 3.2.1 through 3.2.3 were addressed in our response to Items 3.1.1 through 3.1.3, as discussed in Reference (c).

Item 4.1 Reactor Trip System Reliability (Vendor-Related Modifications)

This item is not applicable to Boiling Water Reactors.

Item 4.2 Reactor Trip System Reliability (Preventive Maintenance and Surveillance Program for Reactor Trip Breakers)

This item is not applicable to Boiling Water Reactors.

Item 4.3 Reactor Trip System Reliability (Automatic Actuation of Shunt Trip Attachment for Westinghouse and B&W Plants

This item is not applicable to Boiling Water Reactors.

Item 4.4 Reactor Trip System Reliability (Improvements in Maintenance and Testing Procedures for B&W Plants)

This item is not applicable to Boiling Water Reactors.

Item 4.5.1 and 4.5.2 Reactor Trip System Reliability (System Functional Testing)

As discussed in Reference (c) of the cover letter, Section 3.1 of our facility Technical Specifications provide functional test surveillance requirements for the Reactor Protection System (RPS). This includes a requirement to perform on-line functional testing of the RPS, including independent testing of the scram pilot valves. Item 4.5.1 of the Generic Letter also recommends on-line functional testing of the backup scram valves at GE plants. The backup scram valves are implied to be "diverse trip features" comparable to the breaker shunt trip features on other plants and, as such, should be functionally tested on-line. However, we believe the inherent difference between the GE RPS design and the Reactor Trip System design of other plants, makes the recommendation for on-line functional testing of the backup scram valves unwarranted.

Specifically, Vermont Yankee has 89 control rods. Each control rod is activated by a pair of independent scram pilot valves. Reference (d) of the cover letter is an analysis performed by GE which concluded that reactor shutdown can be achieved if at least 50% of the control rods in checkerboard pattern or 69% of the control rods in a random pattern are inserted into the core. Clearly, only a fraction of these 89 control rods must successfully function to shutdown the reactor. The probability of independent failure of enough rods to prevent shutdown is negligible.

Two redundant backup scram valves are provided in GE plants to assure that the control rods do actuate should any of the pilot scram valves fail to function. No explicit credit is taken for these valves in plant safety analysis or system reliability analyses, nor are they required by applicable regulatory requirements. Functional testing of these valves during plant operation would require a plant scram, a significant challenge to plant safety systems, and therefore a degradation in plant safety. The backup scram valves are non-safety-related additions that can only enhance the reliability of the safety-related Reactor Protection System.

In contrast, other reactor designs include only two redundant reactor trip breakers, one of which is required to successfully function to scram the reactor Each of these breakers has an undervoltage actuation device and diverse functionally redundant shunt actuation device. The successful function of this system requires the operation of 1 of these 4 actuation devices. In the event of an initiating event (loss of AC power) followed by a single active failure (breaker failure), the shunt devices are rendered useless, therefore successful system operation depends on breaker operation by the remaining action device. Obviously, improper function of the diverse safety-related trip devices on other plants would degrade the reliability of the safety-related Reactor Trip System.

In summary, the functions performed by and the safety-related reliability dependence on shunt actuation devices in one design are considerably different from the reliability enhancement afforded by backup scram valves in the GE design. Similarly, testing requirements on shunt actuation devices should not necessarily be requirements for backup scram valves. We therefore conclude that the need for on-line functional testing of the backup scram valves is not necessary.

It should be noted that we have also installed an Alternate Rod Insertion (ARI) system and a Recirculation Pump Trip (RPT) system in response to previous ATWS concerns. The ARI system provides a means to reduce the probability of a failure of control rods to insert on demand, while the RPT system is provided to mitigate the consequences of an ATWS by reducing core power generation by rapidly reducing core flow. The parameters which initiate the ARI and RPT systems are reactor low-low water level, after a time delay of approximately ten seconds, or high reactor pressure. The use of a ten-second delay on low-low water level trip is desirable to avoid making the consequences of a postulated loss-of-coolant accident more severe.

These systems are described in detail in Section 7.18 of our Final Safety Analysis Report (FSAR). The installation of the RPT system is also reflected in our Proposed Change No. 79 to facility Technical Specifications (Reference (e)), which was subsequently approved by the NRC in Reference (f).

<u>Item 4.5.3 Reactor Trip System Reliability (On-Line Functional Testing Intervals)</u>

As discussed above, our facility Technical Specifications provide functional test/surveillance requirements for the Reactor Protection System. These, as well as all other Technical Specification requirements, are continuously reviewed such that test intervals are consistent with achieving high component and system reliability. If existing intervals are found to be inconsistent with achieving high reliability, we would propose a change to our Technical Specifications to modify them accordingly.

In addition, we are following the efforts of the BWR Owners Group Technical Specification Improvement Committee, which is addressing this particular item as part of their overall effort to review and recommend improvements for various equipment test/surveillance intervals. At the present time, the BWROG committee has not established a firm schedule for completing this effort; however, once a final report is made, it is our intent to review the applicability of their recommendation for this particular item as it relates to our facility and submit a formal request to amend our Technical Specifications, as necessary.