Enclosure 1

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

March 9, 1982

50-271

MEMORANDUM FOR: T. M. Novak, Assistant Director for Operating Reactors, DL

THRU: D. B. Vassallo, Chief
Operating Reactors Branch #2, DL

DBV

FROM: Vernon L. Rooney, Project Manager
Operating Reactors Branch #2, DL

SUBJECT: Vermont Yankee - Env. Qualification and Safety of Continued Operation

On February 12 D. Vassallo and I met with D. Hansen of Vermont Yankee and discussed the safety of continued operation of the Vermont Yankee plant until such time as final resolution of the EQ issue. The discussions at this meeting were documented in a letter from the licensee dated February 26, 1982. I have reviewed the above letter and conclude that continued operation of the Vermont Yankee plant is safe until such time as the equipment of questionable qualification in the licensee's September 4, 1981 submittal is fully qualified.

Vernon. L. Rooney, Project Manager
Operating Reactors Branch #2, DL

Attachment:
Vermont Yankee letter of 2/26/82

cc: M. Williams 528

Dupe OF
8203290123 XA

VERMONT YANKEE NUCLEAR POWER CORPORATION

SEVENTY SEVEN GROVE STREET

RUTLAND, VERMONT 05701

2.C.2.1
FVY 82-22

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

February 26, 1982

United States Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Office of Nuclear Reactor Regulation
Mr. D. G. Eisenhut, Director
Division of Licensing

References: (a) License No. DPR-28 (Docket No. 50-271)
(b) VYNPC Letter to USNRC, dated September 4, 1981 (FVY-81-131)

Subject: Environmental Qualification of Safety-Related Electrical Equipment

Dear Sir:

On February 12, 1982, our Mr. D. Hansen met with Messrs. V. Rooney and D. Vassalo of USNRC and provided information which amplifies our submittal, Reference (b), relative to equipment qualification. The discussion centered on those items identified in Reference (b) by Resolution Notes 3, 4, and 7 with respect to the safety of continued operation until final resolution is accomplished. The details of the discussion are summarized in the Attachment to this letter.

Based on the details provided in the Attachment, we conclude that continued operation is justified in the interim until final resoluion of these items is accomplished. We trust this information is satisfactory, however, if you should have any questions, please contact us.

Very truly yours,

VERMONT YANKEE NUCLEAR POWER CORPORATION

R L Smith

R. L. Smith
Licensing Engineer

RLS/jgh

Attachment

A048
S1/1

CAD-3, 7 — The two MOVs are located in the Reactor Building and are used for post-LOCA hydrogen control. Over the period of a few months after a LOCA, they are operated periodically. The only harsh environment is post-LOCA heatup and radiation. Similar MOVs have been qualified for use in containment. Due to a lack of radiation qualification, the motors are being replaced. If the motor should fail within the few months that venting is required, alternate venting can be achieved through qualified butterfly valves in the PCAC System.

CAD-2, 10 — These SOVs are operated for post-LOCA hydrogen sampling and are operated intermittently for a period of a few months. The only harsh environment is post-LOCA heatup and radiation. There are four sampling paths available using these SOVs. In addition, there is an independent sampling path using qualified valves of a different manufacturer.

HVAC-1, 2 — The RRU ventilation fans are located in a mild environment except for post-LOCA radiation, which could be from 10^5R to 2 x 10^6R, which is not excessive for a motor. It is estimated that the motors will reach their design radiation dose in approximately 900 hours. At that time, in the post-LOCA scenario, only one RHR motor and one RHR service water motor are required. If one of the RRU motors should fail, there is a second RRU located in the same corner room. Furthermore, should the second RRU in each room fail, only the equipment in one corner room (one RHR motor and one RHR service water motor) is necessary at that time in the post-LOCA scenario. This means that the required shutdown cooling function could be provided by alternating operation between the two corner rooms. The equipment in each room could be operated for approximately 24 hours before switching to the alternate room becomes necessary due to heat buildup. The rooms where the pumps are located are below grade level and are vented.

MS-3 — The MOV is the main steam drain valve, and it receives a close signal on PCIS after LOCA. However, this valve and its counterpart outside containment are closed during normal operation, as they are only required to be open until operating temperature is achieved. All components of this MOV have been related to qualified MOVs except for the radiation tolerance of the motor. Since the valve is normally closed, it should not be required to function. If open, it will close within seconds and radiation tolerance is not a significant factor. A third valve in this line is also normally closed.

NBV1-5, 6, 7 — These components (pressure switches and circuit board) are all located in the Reactor Building and are used for post-LOCA safety relief valve position indication. They are subject to an environment of post-LOCA heatup and a low post-LOCA radiation dose of less than 2×10^5R. In the event these components should fail, redundant qualified vessel level and drywell pressure indications are available. If a safety relief valve should be stuck open, the plant operators are sensitive to the situation, are trained to recognize it, and will take appropriate action.

PAM-2, 3 — These components monitor safety valve position and are located in containment. They are currently being qualified. If they should fail, redundant qualified vessel level and drywell pressure indications are available, and as noted above, the operators have been trained to handle this situation.

PAM-4 — As Identified in our NUREG-0737 responses, these radiation monitors were installed to meet a 1/1/82 requirement with complete acknowledgement that qualification testing was in progress. Test data for the monitors has since been received and forwarded to FRC. The cable used in our installation (Rockbestos), however, is still undergoing testing. Although there is no cause to believe the cable will not pass the remaining tests, the following justification is provided for continued operation. These monitors are not required to reach safe shutdown. In addition, redundant qualified vessel level instrumentation and a newly intalled post-accident sampling panel are available to follow the course of an accident.

HPCI-13 — There is an alternate qualified monitor for this transmitter (See PAM-9, 16).

PAM-1, 6, 7, 12, 14 [Elec-3, 30*] — These thermocouples monitor drywell air temperature and torus water and air temperature, and perform no safety function. The plant can be shut down safely without this indication by using qualified vessel level and drywell pressure indications. These thermocouples are very similar to the thermocouples which will replace them. They will perform their indication function before failure will occur due to excessive radiation because their primary function is to indicate temperatures during the intial stages of an accident and is not required after conditions have been stabilized.

*The resolution note for Elec-29 should be Note 1, and for Elec-30 it should be Note 3.

PAM-9, 16 — These torus water level transmitters have already been replaced with qualified transmitters. The note for SER resolution should be changed from Note 3 to Note 1.

PAM-11 — The PAM-11 in the original submittal has already been replaced with qualified transmitters for monitoring drywell pressure. The note for SER resolution should be changed from Note 3 to Note 1.

RCIC-1, 2 — These MOVs are on the RCIC line; one inside containment and one in the steam tunnel, and operate on a low steam pressure signal for LOCA containment isolation and on break detection signal for RCIC line break isolation. The MOV components are qualified except for radiation sensitive components. Since the valves perform their function before there is significant radiation, they will complete their safety function before radiation dose can cause failure.

RRS-1 The recirculation discharge and bypass valves are always open when the reactor is operating and close on a LOCA signal. The MOVs are qualified except for the radiation tolerance of the motor. Since the valves perform their function before there is significant radiation, they will complete their safety function before radiation dose can cause failure.

RWCU-2 These MOVs are on the RWCU line, one inside containment (qualified) and one outside containment (qualified except for radiation dose qualification of the motor). Both valves operate in a very short time after LOCA or RWCU break, so radiation dose is not significant relative to expected motor tolerance. In the event of LOCA, the inboard valve is qualified, so single failure of the outboard valve (in a mild environment) is acceptable. In the event of an RWCU break, the inboard valve is in a mild environment and the outboard valve will not see significant radiation.

RHR-1, 6 These MOVs are normally closed containment isolation valves which are opened only to provide a suction path for shutdown cooling. Opening of these valves is blocked above 140 psig reactor pressure. Radiation sensitive parts in these operators are being replaced. If the valves should fail to open, an alternate flow path is available through qualified valves.

RHR-11 These MOVs are backup valves to allow flooding the reactor with river water, and are located in the RHR corner room, a mild environment except for post-LOCA radiation. The MOVs are qualified except for the motor and other radiation sensitive parts. If these valves should fail, valves are available in the mild environment of the Turbine Building to accomplish the same function by an alternate flow path which is independent of the valves in the RHR corner room. The alternate valves are connected to the turbine building service water header, which is separate from the RHR service water header, and allow water to be pumped to the reactor via the main condenser.

SBGT-1 One of the two redundant standby gas treatment fans is required to provide negative pressure in the Reactor Building in the post-LOCA period for as long as required by radiological conditions. The fan will accumulate negligible radiation dose during the first 8 days. Then the fan is assumed to handle containment hydrogen purge which accumulates a radiation source in the SBGT filter. This accelerates the dose accumulation on the motor and it is estimated that the motor will reach its design dose after 20 days. The alternate motor accumulates radiation dose at approximaely half the rate of the operating motor, due to physical separation, and can be used if the first motor should fail. Due to the length of time that the motors will perform their function prior to failure, sufficient time is available to perform an engineering evaluation of the radiological conditions and determine if further venting is required, and, if required, to allow the connection of an alternate venting system.

CAD-6 These compressor motors provide instrument air for the CAD System and are located in the Reactor Building. The only harsh environment during post-LOCA operation is heatup and radiation, and in the few months they are required, the radiation dose is approximately 10^5R. These conditions are not excessive for the motor and it is our engineering judgement that the rotor will complete its function before failure can occur.

ELEC-25 These microswitches provide position indication for several valves in the Reactor Building in the PCAC System. The valves receive an isolation signal immediately after an accident. Therefore, the position indication switches will perform their function before post-LOCA radiation damage occurs. The valves may have to be reopened to allow containment purging during recovery operations. If the switches have failed, valve position can be monitored by the results of the purging operation, i.e., drywell pressure or stack radiation monitoring.

SBGT-4
ADS-1
MS-1, 2
PCAC-1, 2, 8

These SOVs have been replaced with qualified valves. The note for SER resolution should be changed from Note 3 to Note 1.

RHR-8, 9 These MOVs have been qualified by the vendor, so replacement of the motors will not be required. The note for SER resolution should be changed from Note 4 to Note 1.

Environmental Qualification Review for Justification of Continued Operation of Vermont Yankee Nuclear Power Station

1.0 Introduction

On June 3, 1981 we issued a Safety Evaluation (Reference 1) addressing environmental qualification of safety-related electrical equipment at Vermont Yankee Nuclear Power Station. The safety evaluation identified, in part equipment-deficiencies for which the licensee was to either provide qualification information, or commit to an appropriate corrective action (requalification, replacement, etc.) If the licensee elected to commit to a corrective action, then the licensee was to provide a justification for continued operation until such corrective action could be completed. By letter dated September 4, 1981 (Reference 2) the licensee sumitted it's justification for continued operation. Our contractor for environmental qualification, Franklin Research Center, reviewed the licensee's justification and concluded that the licensee had not provided a technically sound rationale as a basis for continued plant operation (Reference 3). The licensee subsequently presented additional information supporting continued operation at a meeting on February 12, 1982 and in a submittal dated February 26, 1982 (Reference 4).

2.0 Evaluation

There are 43 items for which justification for continued operation is required. We reviewed each of these items to determine whether or not each item could be placed in any one of four previously determined acceptance categories. The four categories are:

Category 1. Redundant equipment is available to substitute for the unqualified equipment

Category 2. Another system is capable of providing the required function of the system with unqualified equipment.

Category 3. The unqualified equipment will have performed its safety function prior to failure.

Category 4. The plant can be safely shut down in the absence of the unqualified equipment.

The results of this review are presented in the following table. Ten of the items had been replaced with environmentally qualified equipment by the time of the February 12, 1982 meeting. The remaining thirty-three items were found to fit into one or more of the acceptance categories.

Assignment of Acceptance Categories

For Equipment Identification See Reference 2.

Category 1. CAD 2,3,7,10; MS-3; PAM-2,3,4; HPCI-18; RHR 1,6,11; NBVI-5,6,7; RWCU-2

Category 2. HVAC-1,2

Category 3. PAN-1,6,7,12,14; ELEC-3,25,30*; RCIC-1,2; RRS-1; SBGT-1; CAD-6

Category 4. RHR-8,9

The following items were replaced with qualified components before the February 12, 1982 meeting:

PAM-9,16,11; SBGT-4; ADS-1; MS-1,2; PCAC-1,2,8.

3.0 Summary

Based upon the evaluation described above, and as documented in Reference 5, we find continued operation to be justified for Vermont Yankee Nuclear Power Station.

References

1. NRC letter dated June 3, 1981. T. A. Ippolito to R. L. Smith, Subject Environmental Qualification of Safety-related Electrical Equipment.

2. VYNPC letter dated September 4, 1981, R. L. Smith to T. A. Ippolito, Subject: Environmental Qualification of Safety-related Electrical Equipment.

3. FRC memorandum dated January 25, 1982, C. J. Crane to R. A. Clark, Subject: FRC review of Licensee's responses to NRC EEQ SER concerning justification for interim operation.

4. VYNPC letter dated February 2, 1982, R. L. Smith to D. G. Eisenhut Subject: Environmental Qualification of Safety-Related Electrical Equipment.

5. NRC letter dated March 9, 1982, D. B. Vassallo to R. L. Smith, Subject: Environmental Qualification of safety-related Electrical Equipment for Nuclear Power Plants.

APPENDIX D - REVIEW OF LICENSEE'S RESPONSE TO NRC EEQ SER CONCERNING JUSTIFICATION FOR INTERIM OPERATION

1. BACKGROUND

The NRC Safety Evaluation Report (SER) concerning equipment environmental qualification (EEQ) states [12]:

> "Subsection 4.2 identified deficiencies that must be resolved to establish the qualification of the equipment; the staff requires that the information lacking in this category be provided within 90 days of receipt of this SER. Within this period, the licensee should either provide documentation of the missing qualification information which demonstrates that such equipment meets the DOR guidelines or NUREG-0588 or commit to a corrective action (requalification, replacement, relocation, and so forth) consistent with the requirements to establish qualification by June 30, 1982. If the latter option is chosen, the licensee must provide justification for operation until such corrective action is complete."

On January 19, 1982, FRC representatives met with NRC Division of Licensing personnel at NRC offices to discuss the potential for FRC to assist the staff in the technical review of licensees' statements regarding justification for interim plant operation submitted in response to outstanding qualification deficiencies in the NRC EEQ SERs. The results of the meeting were as follows: (1) FRC was requested to proceed immediately with the technical review of licensees' justification for interim operation, (2) the format was established, and (3) the criteria for the review were established. These criteria are presented in Section 2 of this appendix.

On January 21, 1982, the NRC provided the following modification to Final Assignment 13 concerning this subject:

"The FRC review will consist of:

- o Review the licensee's justification of interim operation and provide FRC independent analysis which shows whether or not licensee provided technically sound rationale as a basis for justification for continued plant operation.

- On January 27, 1982, FRC shall provide a list of those power reactors that have provided technically sound justification for continued operation. FRC shall also provide a list of those power reactors which have not provided technically sound justification for continued operation. In addition to the lists, FRC may provide any additional information which in FRC's judgment is necessary to support the conclusions regarding justification for continued operation."

On January 25, 1982, the NRC was provided with the completed review of the licensees' statements presented as a basis for justification for interim operation in response to the NRC EEQ SER.* On February 5, 1982, at the NRC's request, the NRC was provided with actual examples of licensees' responses to the NRC EEQ SER that provide adequate rationale as a basis for justification for interim operation.**

2. GENERAL DISCUSSION

In general, licensee-submitted justifications for interim operation are based on systems considerations, equipment operability evaluations, or failure-modes-and-effects analyses.

Systems considerations often involve the availability of backup equipment capable of performing the particular safety function of concern. The backup equipment is either environmentally qualified, unqualified but not exposed to a harsh environment at the same time as the primary equipment, or located so that it is unlikely that both the primary and backup equipment would be simultaneously exposed to a severe environment. In general, these systems discussions should consider (1) the possibility of a single-active failure

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* C. J. Crane
Letter to R. A. Clark, NRC. Subject: Transmittal of FRC Review of Licensees' Responses to NRC EEQ SER Concerning Justification for Interim Operation
FRC, 25-Jan-82

** C. J. Crane
Letter to R. A. Clark, NRC. Subject: Transmittal of Actual Examples of Licensees' Responses to NRC EEQ SER Which Provide Adequate Rationale as a Basis for Justification of Interim Operation
FRC, 5-Feb-82

disabling the backup equipment, (2) any major differences in the characteristics of the primary and backup equipment (unless it is obvious that the equipment is essentially identical), (3) the possibility of electrical failure of the primary equipment causing an adverse effect on other safety-related equipment or power supplies, and (4) in the case of display instrumentation, the possibility of an operator being misled by the failed primary equipment. Where equipment has not been demonstrated to be qualified, some justifications discuss administrative procedures or revised operating procedures in effect. Depending upon the specific equipment involved, each of the above considerations need not be discussed in every instance, but, in general, a complete systems discussion would consider the above points.

Where equipment qualification evaluations were used, licensees generally (1) received additional information from manufacturers, (2) applied engineering judgment, (3) performed material analysis, and/or (4) used partial test data in support of the original qualification documentation. Where these evaluations were performed, the licensees determined that, although full qualification was not documented, there was sufficient evidence to suggest that the equipment would perform its intended safety function, thereby justifying interim operation until qualified equipment is installed.

Some licensees provided detailed failure-modes-and-effects analyses of electrical circuitry to demonstrate that, under all identified failure modes, the safety function of the equipment could still be accomplished.

Other justifications involved a combination of qualification information and systems information. For example, if a licensee has qualification information (such as a generic test report or other partial qualification documentation) that tends to confirm the ability of the equipment to remain operable for a specified period of time, justification for interim operation often was based upon a discussion of the required safety function being performed prior to the potential failure. This type of discussion often applies to equipment which performs a short-term trip or isolation function in the early stages of an accident.

3. PLANT-SPECIFIC REVIEW

As a result of the review, this plant was evaluated and the results documented on the "Summary of Review of Licensee's 90-Day Response" form reproduced below:

"EQUIPMENT ENVIRONMENTAL QUALIFICATION (EEQ)
Review of Licensees' Resolution of Outstanding Issues
From NRC Equipment Environmental Qualification
Safety Evaluation Reports

SUMMARY OF REVIEW
OF LICENSEE 90-DAY RESPONSE

Utility: Vermont Yankee Nuclear Power Corp.
Plant Name: Vermont Yankee
NRC Docket No. 50-271
NRC TAC No. 42480
NRC Contract No. NRC-03-79-118
FRC Project No. C5257
FRC Assignment No. 13
FRC Task No. 496

References:

a. R. L. Smith
Letter to T. A. Ippolito (NRC)
Subject: Environmental Qualification
of Safety-Related Electrical Equipment - Response to Safety
Evaluation Report for Vermont Yankee
Vermont Yankee Nuclear Power, 04-Sep-81
FVY 81-131

b. Office of Nuclear Reactor Regulation
Safety Evaluation Report for Vermont Yankee Environmental
Qualification of Safety-Related
Electrical Equipment
NRC, 03-Jun-81

The Licensee has submitted technical information in Reference a in response to the NRC SER [b] on environmental qualification. FRC has reviewed these documents [a, b].

In general, the Licensee's submittal did not adequately address justification for interim operation for deficient equipment items identified in the SER.

The Licensee's resolution of each equipment item identified in Appendices B and C of the SER was indicated by assigning one or more of the following 'notes for resolutions' to the specific items:

'NOTES FOR RESOLUTIONS

1. Qualification documents associated with this piece of equipment have been evaluated and have been found to meet the intent of the applicable standards and is therefore qualified. (Refer to latest revision of 79-01B worksheet.)

2. Subsequent detailed review of the 90 day submittal resulted in deletion of this equipment from the worksheet list for one or more of the following reasons:

   (1) Equipment does not perform essential safety functions in the harsh environment, and equipment failure in the harsh environment will not impact safety-related functions or mislead an operator.

   (2a) Equipment performs its function before its exposure to the harsh environment, and the adequacy for the time margin provided is adequately justified, and

   (2b) Subsequent failure of the equipment as a result of the harsh environment does not degrade other safety functions or mislead the operator.

   (3) The safety-related function can be accomplished by some other designated equipment that has been adequately qualified and satisfies the single-failure criterion.

   (4) Equipment will not be subjected to a harsh environment as a result of the postulated accident.

3. Due to advances in equipment design, this equipment is slated for replacement during the next available outage consistent with equipment delivery time requirements.

4. Additional documentation is being assembled and reviewed to correct deficiencies. It has been determined that additional documentation is available, can be assembled, and that the review will adequately establish qualification. Should it be established otherwise either on technical or economic grounds, equipment replacement program will be undertaken. NRC will be notified of the change along with the applicable schedule for replacement.

5. Equipment deleted from master list.

6. TMI Items, to be addressed by a supplement.

7. Qualification testing is currently being conducted on this piece of equipment by the manufacturer. Upon completion of testing, reports will be reviewed to provide adequate qualification documentation.'

Equipment items designated with several of these 'notes of resolution' (e.g, Nos. 3, 4, and 7) are not documented as environmentally qualified. Review of Reference 1 does not reveal justifications for interim operation for these items."

4. SUBSEQUENT REVIEW

As a result of FRC's review of the Licensee's 90-day response, described in Section 3 above, a meeting was held between the NRC staff and certain Licensee personnel. Following the meeting, the Licensee submitted Reference 54, in which additional information justifying interim operation was submitted for each equipment item not documented as environmentally qualified.

Evaluation

An evaluation has been conducted of the information provided by the Licensee in Reference 54 regarding justification for interim operation. After reviewing the technical basis of the Licensee's justification for continued operation for each item, it is concluded that the Licensee has provided sufficient technical basis to support justification for interim operation.