DD-97-25

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION

Samuel J. Collins, Director

In the Matter of) Docket No. 50-271
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VERMONT YANKEE NUCLEAR POWER) License No. DPR-28
CORPORATION)
) (10 CFR 2.206)
(Vermont Yankee Nuclear Power)
Station))

PARTIAL DIRECTOR'S DECISION PURSUANT TO 10 CFR 2.206

I. INTRODUCTION

On December 6, 1996, Mr. Jonathan M. Block, submitted a Petition to the Office of the Secretary of the U.S. Nuclear Regulatory Commission (NRC) pursuant to Section 2.206 of Title 10 of the Code of Federal Regulations

(10 CFR 2.206). The Petition was submitted on behalf of the Citizen's Awareness Network, Inc. (CAN or Petitioner), and contained two Memoranda from CAN. The first Memorandum enclosed with the Petition is dated

December 5, 1996. It reviews information presented by the Vermont Yankee Nuclear Power Corporation (Licensee) at a predecisional enforcement conference held on July 23, 1996, involving the minimum flow valves in the residual heat removal (RHR) system at the Vermont Yankee Nuclear Power Station (Vermont Yankee facility). CAN raises a concern that the corrective action taken by the Licensee in opening these valves may have introduced an unreviewed safety question with regard to containment isolation.

The second Memorandum enclosed with the Petition is dated December 6, 1996, and contains a review of certain licensee event reports (LERs) submitted by the Licensee in the latter part of 1996. Various issues are presented, such as fire protection, tornado protection, thermal protection for piping lines, equipment operability, and equipment testing. On the basis of its analysis of the LERs, CAN reaches certain conclusions regarding Licensee performance and actions that should be taken. In the Petition, the Petitioner requested that the NRC evaluate these documents, pursuant to 10 CFR 2.206, to see if enforcement action is warranted based upon the information contained therein.

On February 12, 1997, the NRC informed the Petitioner in an acknowledgement letter that the Petition had been referred to the Office of Nuclear Reactor Regulation for the preparation of a Director's Decision and that action would be taken within a reasonable time regarding the specific concerns raised in the Petition.

II. DISCUSSION

The NRC staff evaluation of these documents follows.

A. The Residual Heat Removal System

The first document enclosed with the Petition is a CAN Memorandum dated December 5, 1996, that reviews information presented by the Licensee at a predecisional enforcement conference held on July 23, 1996, involving the minimum flow valves in the Vermont Yankee RHR system.[1] The Vermont Yankee RHR system consists of two loops. Each

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[1]Several statements in the December 5, 1996, Memorandum are either unclear or incorrect. A single power supply failure does not prevent RHR minimum flow valves in both loops from operating,

loop has two pumps that take suction from the suppression chamber. Each pump has a minimum flow line equipped with a minimum flow valve that returns flow to the suppression chamber. The RHR pumps start automatically to cool the reactor in case of a loss-of-coolant accident (LOCA). The minimum flow valves close to prevent flow from being diverted from the reactor core to the suppression pool when flow is being supplied from the RHR pumps to the reactor core, and open automatically on high pump discharge pressure to protect the RHR pumps if other valves between the RHR pumps discharge and the reactor core are not yet open.

The Licensee discovered a vulnerability to single failure which could prevent the minimum flow valves from opening to protect the RHR pumps during a LOCA. To resolve this concern, the Licensee changed the normal and failed positions of these valves from CLOSED to OPEN. The Petitioner is concerned that the corrective action taken by the Licensee in opening these valves may have introduced an unreviewed safety question with regard to containment isolation.[2] A pipe break outside containment would breach primary containment with an OPEN minimum flow valve.

This issue must be addressed in terms of the Vermont Yankee facility licensing basis. The basic design for early boiling-water reactors, including the Vermont Yankee facility which

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contrary to the statement on page 2 of the Memorandum. Minimum flow valves in both loops will not remain open if a single power supply failure occurs, contrary to the statement on page 3 of the Memorandum. Also, on page 4 of the December 5, 1996, Memorandum, CAN questions the remote manual closure capability of the minimum flow valves. The minimum flow valves have remote manual closure and opening capability, but the pump protection logic will override any remote manual closure or opening signal.

[2] The NRC staff assumes Petitioner's reference to an "unreviewed safety question" is in the context of the NRC's regulation 10 CFR 50.59, "Changes, Tests, and Experiments".

was reviewed and accepted by the NRC, considered the piping of the RHR and Core Spray (CS) Systems to be a closed extension of primary containment. Failure of the passive pressure boundary (piping) of these systems during either the short-term (injection phase) or long-term (recirculation phase) course of a design-basis accident (DBA) was not a design basis assumption. As a result, the RHR and CS suction and minimum flow lines were not provided with containment isolation valves, or if valves were provided in these lines, they were not provided for the purpose of meeting containment isolation requirements and thus were not classified as containment isolation valves. In most if not all cases, the penetrations of concern in the older plants were originally provided with at least one valve capable of performing the containment isolation function, and these valves are periodically tested under inservice testing (IST) program requirements. The Vermont Yankee minimum flow valves can be remotely closed and are periodically tested under the IST program.

For more recent facilities, emergency core cooling system (ECCS) closed systems outside containment are required to have at least one recognized isolation valve at each penetration. This is not the case for the Vermont Yankee facility.

In view of the licensing criteria applicable to the Vermont Yankee facility, maintaining the minimum flow valves of the RHR system in the OPEN position is permitted and acceptable. The Vermont Yankee final safety analysis report (FSAR) does not describe the minimum flow valves as being in the CLOSED position, and placing these valves in the OPEN position is not a change to the facility under the meaning of 10 CFR 50.59 and no unreviewed safety question is presented. For the above reasons, no enforcement action is warranted with regards to this issue.

B. Licensee Event Reports

The second document enclosed with the Petition is a CAN Memorandum dated December

6, 1996, that contains a review of several LERs submitted by the Licensee in the latter part of 1996. Various issues are presented, such as fire protection, tornado protection, thermal protection for piping lines, equipment operability, and equipment testing. On the basis of its analysis of the LERs, CAN reaches certain conclusions regarding Licensee performance and actions that it believes should be taken. First, CAN requests that the NRC and the Licensee review all safety analyses conducted since initial startup of the Vermont Yankee facility with particular attention to their role in providing a complete and up-to-date FSAR. Second, the Licensee needs to correct serious deficiencies in its design change control process and should undertake a historical review of its design control documentation to verify its accuracy. Third, the Licensee should perform a global evaluation to determine how many modifications have been inadequately tested since startup. Fourth, the Licensee needs to initiate a thorough retraining program to review and emphasize the underlying safety purposes of Technical Specifications, the FSAR, design bases and NRC regulations in relation to routine operation of the Vermont Yankee facility, emergency preparedness, and practical implementation of the NRC's "defense in depth" philosophy. Finally, CAN strongly recommends that the Licensee's Vermont Yankee staff receive training on the proper use of the "Single Failure" criterion.

The LERs identified in the CAN Memorandum are briefly discussed below.

1) LER 96-13: "Two fire suppression systems do not meet design requirements due to personnel error on the part of [the] vendor who designed and installed the systems"

CAN asserts that the LER did not address the cause and consequences of the foam suppression system deficiency, which is one of the two fire suppression systems addressed in this LER. CAN is correct in that the Licensee did not determine a precise root cause because such a long time had elapsed since the occurrence (1978). It is not unreasonable for a licensee to be unable to ascertain the exact root cause of a personnel error that took place many years before (18 years in this case). Key points that are considered in reviewing an LER are (1) whether the specific problem is being appropriately addressed, (2) whether the potential for a broader problem exists and (3), if a broader problem exists, whether it is properly addressed. In this case, the Licensee reviewed its current design process and procedures and determined that a similar occurrence would not be expected to occur now, and the Licensee had two teams that were actively reviewing the fire protection design bases and searching for the types of problems reported in the LER. CAN is incorrect in stating that the consequences of the foam system deficiency were not discussed in the LER. The Licensee stated that any fire in the area would be contained and suppressed, preventing its spread to safety-related equipment.

Because the design deficiencies addressed in this LER were licensee-identified and corrected, they were treated as Non-Cited Violations in Inspection Report 50-271/96-05 in accordance with Section VII.B.1 of the NRC's

Enforcement Policy,[3] and the LER was closed in Inspection Report 50-271/96-

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[3]The NRC's policy and procedures for determining the enforcement action that may be warranted for a violation are discussed in NUREG-1600, "General Statement of Policy and Procedures for NRC Enforcement Actions" (Enforcement Policy). Because regulatory requirements have varying degrees of safety, safeguards, or environmental significance, the first step in the enforcement process is to evaluate the significance of the violation and then assign a severity level to the violation. A violation is assigned one of four severity levels. As described in Section IV of the Enforcement Policy, Severity Level I is assigned to violations that are the most safety significant and Severity Level IV is assigned to violations that are the least safety significant. Consistent with the recognition that violations have different degrees of safety significance, the Enforcement Policy recognizes that there are other violations of minor safety or environmental concern that are below the level of significance of Severity Level IV violations. These minor

06. Further enforcement action is not warranted.

(2) LER 96-14: "Failure to provide tornado protection for diesel generator rooms as specified in the Final Safety Analysis Report due to unknown cause"

The FSAR states that large venting areas are provided to vent the diesel generator room in the event of a tornado to provide pressure equalization. The LER notes that the facility as constructed did not include venting.

CAN asserts that "flaws in the FSAR cause serious, rippling effects throughout VY's [Vermont Yankee facility's] safety systems" and that the Licensee "must include assessments of the impact of the deficient conditions upon all affected programs."

The Licensee took immediate action to insure emergency diesel generator (EDG) operability in the absence of the pressure relief panels. The Licensee took immediate compensatory measures which included blocking open the EDG room doors and posting fire and security watches. The Licensee took additional compensatory actions for the restoration of operability of the diesel and day tank enclosures during cold weather months when the EDG doors had to be shut. An NRC inspector verified that the recommended compensatory measures were properly implemented.

The discrepancy between the actual plant design and the FSAR is a de facto change to the facility as described in the safety analysis report, and thus required an evaluation to meet the requirements of 10 CFR 50.59. The failure to perform such a 10 CFR 50.59 evaluation was

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violations are not normally the subject of formal enforcement action and are not usually described in inspection reports. To the extent that such violations are described, they are usually described as "Non-Cited Violations."

categorized as a Severity Level IV violation, and was dispositioned in Inspection Report 96-11 as a Non-Cited Violation in accordance with Section VII.B.1 of the Enforcement Policy.

Other plants have been found to have FSARs which do not properly describe the facilities. Consequently, for this reason and as a result of lessons learned from events at Millstone Nuclear Power Station and Maine Yankee Atomic Power Station, on October 9, 1996, the NRC requested information from all power reactor licensees, to verify, among other things, that the plant FSARs properly describe the facilities, and that the systems, structures, and components are consistent with the design basis. In conjunction with this request for information, and in order to encourage licensees to identify discrepancies, the Commission approved a modification to the NRC Enforcement Policy that allows the NRC staff to exercise enforcement discretion for a period of 2 years for violations related to FSAR discrepancies identified by licensees. The policy revision was published in the Federal Register on October 18, 1996 (61 FR 54461).

In the Licensee's response to this request for information dated February 14, 1997, the Licensee committed to complete its FSAR verification program in 1998.

CAN raises a concern about a potential error in the Licensee's statement in this LER of no prior occurrences, based on a James A. Fitzpatrick Nuclear Power Plant report of a similar problem. Licensees are only required to report prior similar occurrences at their facility, and not at any other facility. Therefore, the Licensee was accurately reporting that a similar event had not previously occurred at Vermont Yankee Nuclear Power Station. This LER is closed. Further enforcement action is not warranted. The Licensee has issued a supplement to this LER to document the long term corrective actions to vent the EDG room in the event of a tornado to provide pressure equalization. This LER supplement remains open pending NRC inspection of the Licensee's modifications to the EDG room to provide the required pressure equalization.

(3) LER 96-15: "Original B31.1 ANSI Code Section that Required Overpressurization Relief for Isolated Piping Sections was not Considered during [the] Original Design"

Certain piping sections, which would be isolated after a LOCA, were found to lack overpressure protection contrary to code requirements. The water in this piping could expand because of the high temperatures accompanying a LOCA and exceed the design pressure rating of the piping. CAN asserts that the Licensee failed to take advantage of earlier opportunities to identify this design error when making modifications to the six systems discussed in the LER. CAN is correct in that the LER represented the first discovery of this problem, although modifications had been made to the affected systems earlier. This potential overpressurization problem has been identified at other plants, as evidenced by the issuance of NRC Information Notice (IN) 96-49 on August 20, 1996, and NRC Generic Letter (GL) 96-06 on September 30, 1996. The Licensee did maintain an awareness of events in this area and identified this issue at its site before the generic communications referred to above were issued. The NRC staff encourages licensee initiatives to identify and correct safety problems that may be generic to the industry in advance of generic NRC staff communications to the industry. The Licensee's corrective actions included a design change which provided the required overpressure protection for the affected lines. The change was completed in the 1996 refueling outage.

This LER remains open. Responses from power reactor licensees to GL 96-06 were received by the NRC staff in February 1997 and are undergoing review to assure that the overpressure protection issue is being adequately addressed and resolved. Following this generic review, a determination will be made of whether enforcement action is warranted for specific plants. Information regarding the completion of this activity and any enforcement action taken will be publicly available in the plant specific Inspection

Reports. This LER will be further discussed in a Final Director's Decision when the LER is closed.

(4) LER 96-18: "Inadequate Installation and Inspection of Fire Protection Wrap Results in Plant Operation Outside of Its Design Basis, A Single Fire Would Impact Multiple Trains of Safety-Related Equipment"

CAN asserts that this deficiency had significant adverse safety implications. The reported deficiency consisted of a small gap in the fire barrier installed on a cable tray support. The cable tray contained wiring to support operation of the ECCS. The NRC staff does not consider CAN's claim, that a fire could have rendered both divisions of the ECCS inoperable, credible. The Licensee's evaluation found that existing fire protection analyses were very conservative, and that, with the combustible loading and fire detection and suppression equipment in the area, no credible fire threat could challenge the functionality of the "as found," wrapped cable. The Licensee has acted appropriately to correct the fire barrier deficiency and to prevent similar problems in the future. With the combustible loading, fire detection, and suppression equipment in the area, the NRC staff conceptually agrees with the Licensee's conclusion that no credible fire threat could challenge the functionality of the "as found" wrapped cable. Inspection activities were performed the week of August 18, 1997 to verify the Licensee's conclusion.

This LER remains open. Results of the inspection and any enforcement action as a result of this inspection activity will be made publicly available through plant specific Inspection Reports. This LER will be further discussed in a Final Director's Decision when the LER is closed.

(5) LER 96-19: "Half scram and group III containment isolation caused by loose Reactor Protection System breaker termination"

The NRC staff agrees with CAN that this event presented no significant risk to public health and safety. This LER is closed. No violation was involved, therefore the NRC staff concludes that enforcement action is not warranted.

(6) LER 96-20: "Inadequate vender [sic] design activity and Licensee design verification result in inability to demonstrate Fire Suppression System Operability"

This LER involved the inability of the carbon dioxide fire suppression system to fully extinguish a deep-seated fire, as required. The Licensee stated in the LER that this event had no safety significance. The NRC staff considered this LER to have little apparent actual or potential safety significance. This conclusion was based on the Licensee's analysis that although the carbon dioxide suppression systems might not fully extinguish a deep-seated fire, the suppression and detection systems would function. Fire detection would alert the fire brigade, and because the carbon dioxide fire suppression system had reduced the fire, the fire brigade could extinguish the fire more easily. The NRC staff closed this LER in Inspection Report 96-11. Pending inspector review of the Licensee's corrective actions, the unresolved item initiated for this issue in Inspection Report 96-08 (URI 96-08-01) was left open. As documented in Inspection Report 97-05, unresolved item 96-08-01 was closed and a Non-Cited Violation was issued, consistent with Section VII.B.1 of the NRC Enforcement Policy. Further enforcement action is not warranted.

CAN asserts that this LER reveals a serious deficiency in the Licensee's design change control process, and that the Licensee should determine how many other modifications have been inadequately tested since startup. The NRC staff agrees that this event demonstrated a weakness in the Licensee's modification and testing programs associated with fire protection. As noted under the discussion regarding LER 96-13, the Licensee has initiated reviews of the fire protection design bases to search for these types of problems, and believes that the current design process and procedures are adequate to prevent similar problems. As discussed earlier, by letter dated October 9, 1996, the NRC staff requested information from all licensees, to verify, among other things, the adequacy of the design change control process and to determine the rationale for concluding that design-basis requirements are properly translated into operating, maintenance, and testing procedures. The Licensee responded by letter dated February 14, 1997.

(7) LER 96-21: "Inadequate procedural controls of MOV Limit Switch Settings result in a potential common cause failure mode with the capacity to affect multiple safety significant components"

This LER involved two limit switches on shutdown cooling suction motor-operated valve (MOV) to the "D" RHR pump. The switches measure valve travel towards the open position. One open limit switch permits the pump motor to start after the valve position is sufficiently open, and the other limit switch stops valve travel so that the motor doesn't drive the valve too far and damage the valve. The Licensee identified that a modification to the valve's motor operator resulted in the improper setting of these two limit switches.

Inspector follow-up, as documented in Inspection Report 97-05, led to the conclusion that this error was of low safety significance. The failed start of the "D" RHR pump because of this limit switch error on the shutdown cooling suction valve affected only the shutdown cooling mode of operation of the RHR system. The failure did not impact the other modes of RHR system operation and the safety design bases functions of the RHR system. Further, prompt Licensee action was taken to check the other recently modified MOVs. Their limit switches were found to be properly set and therefore their safety functions were unaffected. This licensee-identified and corrected violation resulted in the issuance of a Non-Cited Violation, consistent with Section VII.B.1 of the NRC Enforcement Policy. This LER is closed. Further enforcement action is not warranted.

(8) LER 96-22: "Combination of poor man-machine interface, an inadequate procedure, inadequate Operating Experience Review results in a common cause failure mechanism, and an Emergency Diesel Generator to exceed Tech Spec [sic] outage time"

The output breaker for one emergency diesel generator (EDG) was found to be incapable of closing because of a missing cotter pin which was necessary for a mechanical linkage. As a result of the absence of this cotter pin, the breaker closing springs failed to recharge, rendering the breaker incapable of being closed from the control room. The only indication that the closing springs had failed to recharge was a mechanical flag indicator located behind the breaker cubicle door. No licensee procedures required verification of the closing spring status. The closing springs were apparently in an uncharged condition for over three weeks without discovery. Because the periodic surveillance interval for the breaker is greater than the EDG limiting condition for operation (LCO), the Licensee unknowingly operated in violation of its Technical Specification (TS) governing diesel generator operability. After reviewing the Licensee's root cause analysis of this event, the NRC staff determined that the missing cotter pin would not reasonably have been expected to be detected by the Licensee's existing quality assurance program or through other related control measures.[4] The Licensee identified the EDG inoperability, investigated to determine when the problem arose, and reported that the LCO time was exceeded. The Licensee responded to the inoperable equipment when the inoperability was discovered. The Licensee did not intentionally exceed an LCO. Rather, the Licensee discovered an equipment problem caused by a malfunction beyond its control which meant that, in hindsight, an LCO had been exceeded. The Licensee is designing a modification for this and other circuit breakers of similar design to allow monitoring of the charging status of the closing springs without having to open the circuit breaker cubicle door.

Because the EDG inoperability was not avoidable by reasonable Licensee quality assurance measures or management controls, the NRC did not issue a Notice of Violation for this issue. This is consistent with Section VI.A of the Enforcement Policy. This LER is closed. The NRC staff concludes that further enforcement action is not warranted.

(9) LER 96-23: "Inadequate Surveillance Procedure results in failure to meet Technical Specification requirements for Radiation Monitor Functional Testing"

[4] CAN asserts that the Licensee misconstrues the purposes of TS Limiting Conditions for Operation (LCOs) as part of a "chronic pattern of misunderstanding" of TS, FSAR design bases, and NRC regulations. For the reasons described herein, LER 96-22 does not provide a basis for this assertion.

The reactor building and refueling floor radiation monitor test procedure did not verify the high alarm contact actuation as required by TS. The NRC staff agrees with CAN that this event presented no significant risk to public health and safety. Considering that the monitors were verified to be fully functional, and were in the condition required by Plant Technical Specifications, this specific event appears to have been limited to an inadequate testing methodology. The Licensee's corrective actions included revising the deficient surveillance test procedure to properly test the high alarm output contacts.

However, the LER remains open as the NRC staff has not completed its inspection activities related to this LER. The NRC staff will look historically to see if this is an isolated case as part of the enforcement consideration. On January 10, 1996 the NRC issued Generic Letter (GL) 96-01 , "Testing of Safety-Related Logic Circuits," that requested, among other things, that all power reactor licensees review their surveillance test procedures to ensure that all portions of the logic circuitry are being tested. The Licensee's response to GL 96-01, due to be sent to the NRC in September 1997, will be evaluated with respect to the Licensee's long-term corrective action for logic testing procedures, because any associated corrective action could be considered in determining whether enforcement action is warranted. Information regarding any enforcement action taken will be available publicly in plant-specific Inspection Reports. This LER will be further discussed in a Final Director's Decision when the LER is closed.

(10) LER 96-25: "Inadequate testing leads to misadjustment of isolation valve mechanical stop and failure to meet Technical Specification leak rate limits for containment purge isolation valve"

This LER involved a containment isolation valve which leaked in excess of TS requirements. The amount of valve leakage was influenced by the direction in which the valve was leak tested and the adjustment of a mechanical stop. CAN's concern appears to be that the Licensee failed to apply the single-failure criterion in assessing the significance of the failure in its LER. Section 50.73(b)(3) requires that an LER contain an assessment of the safety consequences and implications of the event, including the availability of other systems or components that would have performed the safety function of the failed system or component. In this case, the requirement is that the assessment include the availability of a redundant component (valve) that would have performed the safety function (torus isolation). Petitioner's issue is thus whether the LER should have, in addition, assessed the potential radiological consequences had a design-basis accident (DBA) occurred with failure of the redundant isolation valve. Compliance with Section 50.73(b)(3) does not require that the assessment consider an additional single failure beyond the failure which forms the basis for the assessment. On the basis of required reporting, LER 96-25 was not deficient in omitting discussion of the potential consequences of failure of the redundant valve. Inspection Report 50-271/96-11 dispositioned this Severity Level IV TS violation as a Non-Cited Violation in accordance with the criteria for enforcement discretion in Section VII.B.1 of the Enforcement Policy. Although the event was considered to be of more than minor safety significance, the outboard valves had successfully passed all previous tests,

and thus the demonstrated containment integrity was always maintained for the two affected penetrations. This LER is closed. No further enforcement action is warranted.

C. Summary

In summary, with respect to CAN's concern that an unreviewed safety question with respect to containment isolation may have been introduced by Licensee actions in opening the RHR minimum flow lines, the NRC staff determined that no unreviewed safety question was introduced and, therefore, no enforcement action is warranted. With respect to CAN's concerns related to the LERs, the NRC staff finds that the Enforcement Policy has been applied consistently for the LERs that have been closed and further enforcement action is not warranted.

For those LERs which remain open the Inspection/Enforcement process will continue until the staff has completed its investigation and consideration of the issues involved. LER closure and enforcement action, as appropriate, will be documented publicly as is NRC staff practice, and will documented in a Final Director's Decision.

With regard to CAN's overall conclusions based on its analysis of the above LERs, the NRC staff has reached the following conclusions:

With respect to CAN's conclusion that the NRC and the Licensee should review all safety analyses conducted since startup of the Vermont Yankee facility with particular attention to their role in providing a complete and up-to-date FSAR, the NRC staff has taken actions as noted in the discussion above related to LER 96-14 with respect to identifying and correcting FSAR inaccuracies. This action was taken in a request on October 9, 1996, to all licensees, including Vermont Yankee, to provide the requested information. In addition, the NRC staff has implemented a series of engineering design inspections, including an inspection to verify portions of the Licensee's design control process and maintenance of the Licensees's FSAR commitments. The results of the NRC design inspection conducted at Vermont Yankee were reported in Inspection Report 97-201 dated August 27, 1997.

With respect to CAN's conclusion that the Licensee needs to correct serious deficiencies in its design change control process and should undertake a historical review of its design control documentation to verify its accuracy, the NRC staff has taken action as noted in the discussion related to LER 96-20 with respect to identifying and correcting design change control process deficiencies. In the October 9, 1996 letter to all licensees, including Vermont Yankee, the NRC staff requested information to verify, among other things, the adequacy of the design change control process and to determine the rationale for concluding that design-basis requirements are properly translated into operating, maintenance, and testing procedures. As also noted in the discussion related to LER 96-20, the Licensee has undertaken a review of the fire protection design bases to search for the type of problems involved in LER 96-20, and believes that the current modification programs are adequate to prevent similar problems.

With respect to CAN's conclusion that the Licensee should perform a global evaluation to determine how many modifications have been inadequately tested since startup, as noted in the discussion related to LER 96-20, the Licensee has been required to provide verification of the design change control process, including among other things the rationale for concluding that design basis requirements are translated into testing procedures.

With respect to CAN's conclusion that the Licensee needs to initiate a thorough retraining program to review and emphasize the underlying safety purposes of TSs, the FSAR, design bases and NRC regulations in relation to routine operation of the Vermont Yankee facility, emergency preparedness, and practical implementation of the NRC's "defense in depth" philosophy, the NRC staff disagrees. In the discussion related to LER 96-22, the NRC staff addresses CAN's assertion that the Licensee misconstrues the purposes of TS LCO as part of a "chronic pattern of misunderstanding" of TS, FSAR design bases and NRC regulations. The NRC staff finds no basis to require such a retraining program.

Finally, CAN strongly recommends that the Licensee's Vermont Yankee staff receive training on the proper use of the "Single Failure Criterion." In the discussion related to LER 96-25, the NRC staff addresses what seems to be the basis for CAN's recommendation: i.e. the perception that the Licensee failed to properly apply the Single Failure Criterion in assessing the significance of a leaking isolation valve in LER 96-25. Compliance with Section 50.73 does not require that the assessment consider an additional single failure. The enforcement conference related to the minimum flow valves concerned a problem in implementation of the Single Failure Criterion; not a misunderstanding of the requirements of the Single Failure Criterion. Because the Licensee did not err in the instance described in LER 96-25 and the Petition provides no other instances in which problems were caused by a misunderstanding of the Single Failure Criterion, the NRC staff finds no basis for requiring additional training.

III. CONCLUSION

The NRC staff has reviewed the information submitted by the Petitioner. The Petitioner's request is granted in that the NRC staff has evaluated the majority of issues and LERs raised in the Memoranda provided by the Petitioner to see if enforcement action is warranted based on the information contained therein. The NRC staff has discussed each Memorandum above and described any related enforcement action taken for those issues and LERs which are closed. The NRC will continue the same process and consideration for the LERs that remain open and documentation of any inspection and/or enforcement action will be consistent with agency practices and will also be the subject of a Final Director's Decision.

As provided in 10 CFR 2.206(c), a copy of this Decision will be filed with the Secretary of the Commission for the Commission's review. This Decision will become the final action of the Commission 25 days after issuance, unless the Commission, on its own motion, institutes review of the Decision in that time.

Dated at Rockville, Maryland, this 8th day of October 1997.
FOR THE NUCLEAR REGULATORY COMMISSION

Original signed by

Samuel J. Collins, Director
Office of Nuclear Reactor Regulation