U.S. NUCLEAR REGULATORY COMMISSION
REGION I

Report No. 50-271/90-15

Docket No. 50-271 License No. DPR-28

Licensee: Vermont Yankee Nuclear Power Corporation
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Ferry Road
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Facility: Vermont Yankee Nuclear Power Station

Location: Vernon, Vermont

Dates: October 10 - November 26, 1990

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Approved by: John F. Rogge 12/6/90
John F. Rogge, Chief, Reactor Projects Section 3A Date

Inspection Summary: Inspection on October 10 - November 26, 1990

Areas Inspected: Resident safety inspection of the following areas: plant operations, radiological controls, maintenance and surveillance, security, engineering and technical support, and safety assessment and quality verification.

Results: Inspection results and conclusions are summarized in the attached Executive Summary. One non-cited violation was identified in the area of engineering and technical support involving thirty-one Rosemount transmitters that were determined not to be environmentally qualified for high-energy line break accidents (See Section 6.3). One unresolved item was identified in the area of engineering and technical support concerning the adequacy of the mechanism for ensuring timely incorporation of non-routine procedural changes pertaining to FSAR requirements into all applicable plant procedures (See Section 6.4).

33

EXECUTIVE SUMMARY

Vermont Yankee Inspection Report 50-271/90-15

October 10 - November 26, 1990

Plant Operations

During plant start-up activities following the refueling outage and during periodic control room panel walkdowns, operators consistently demonstrated a high level-of-knowledge regarding ongoing plant evolutions and equipment status. However, on two occasions the inspector found valves apparently mispositioned during control room panel walkdowns.

A reactor trip during performance of a weekly turbine emergency governor trip test was handled well by operators and followup maintenance actions were timely and effective. The licensee demonstrated excellent interdepartmental coordination during a loss of Advanced Off-Gas (AOG) auxiliary equipment control power. During troubleshooting of the loss of AOG control power, an uncontrolled release of radioactivity occurred. A conservative analysis showed the release fell significantly below regulatory reporting requirements. Routine inspections of the reactor building identified some minor safety concerns involving lighting, a fire extinguisher, a radiological posting, and a roof access door.

Radiological Controls

A new initiative to enhance the radiological protection (RP) program is discussed. Routine inspector tours of the radiological controlled area and a reactor building RP walkdown with the on-shift RP assistant did not identify significant radiological safety concerns.

Maintenance and Surveillance

The licensee response to feedwater check valve flaws, potential inadequacies of Teflon seals in the containment personnel airlock, and the main steam relief valve accumulator leakage test failure are discussed. An excess flow check valve functional test resulted in three Group I isolations. The inspector concluded that this event was reportable under 10 CFR 50.73. The inspector observed subsequent excess flow check valve testing and determined corrective actions were sufficient to prevent recurrence.

Security

The licensee identified two events involving suitability of personnel for site access. The events were conservatively reported to the NRC and the inspector determined that management corrective actions will likely reduce recurrence of these events.

Executive Summary

Engineering and Technical Support

Linear flaws in several control rod drive (CRD) cap screws exceeded ASME Code Section XI requirements. A conservative bounding evaluation provided adequate justification for continued operations. Core operating limits and core component qualification for Cycle 15 are discussed. An unresolved item concerning qualification of Rosemount transmitters using Teflon thread sealant is closed. During investigation of this unresolved item, the licensee determined 31 Rosemount transmitters were not environmentally qualified (NCV 90-15-001). LER 90-10 Supplement 1 provided additional details on the failure to meet Technical Specifications for diesel generator testing. However, the inspector's concern for ensuring timely incorporation of non-routine changes pertaining to FSAR requirements to all applicable plant procedures remains unresolved (UNR 50-271/90-15-002).

Safety Assessment and Quality Verification

Outage management a noteworthy strength. The outage organization was very effective in ensuring timely and appropriate resolution of safety concerns.

TABLE OF CONTENTS

Table of Contents

ATTACHMENTS

DETAILS

1. Summary of Operations

Vermont Yankee Nuclear Power Station (VY) entered the report period shutdown, with unit personnel conducting plant restoration and start-up preparations. On October 14, with turbine generator restoration and startup surveillance activities complete, operators commenced a reactor start-up. The turbine was synchronized to the power grid, removed for overspeed testing and subsequently synchronized to the power grid on October 16. Power ascension progressed and on October 22, the reactor attained 100 percent of rated thermal power. Normal reactor power operations continued until November 4 when the reactor was automatically shutdown due to an equipment malfunction experienced during performance of a weekly turbine generator emergency governor trip test. After corrective maintenance repaired the equipment failure, unit operators completed a reactor start-up. On November 7, the reactor was operating at 100% power and normal power operations continued throughout the remainder of the report period.

2. Plant Operations (71707, 93702)

2.1 Inspection Activities

The inspectors verified that the facility was operated safely and in conformance with regulatory requirements. Management control was evaluated by direct observation of activities, tours of the facility, interviews and discussions with personnel, and independent verification. The inspectors performed 214 hours of normal and backshift inspection including deep backshift weekend and holiday inspection on October 8, 13, and 14, and November 4, 12, 22, and 23. Operators and shift supervisors were alert, attentive and responded appropriately to annunciators and plant conditions.

2.2 Inspection Findings and Significant Plant Events

A. Reactor Startup Following Refueling Outage

On October 14, the inspector observed reactor start-up from the control room. The start-up culminated completion of an approximately 6 week refueling/maintenance outage. Rod withdrawal was interrupted because control rod 34-15 could not be withdrawn beyond notch position 46. The full-out position for Vermont Yankee control rods is position 48. After troubleshooting identified no instrumentation problems, the licensee concluded that the rod was mechanically restricted from reaching the full-out position. Movement of the rod between the full-in position and the position 46 was not restricted and the rod could perform its intended safety function.

After consulting with the Reactor and Computer Engineering supervisor, start-up rod withdrawal resumed. Control rod 34-15 remained at position 46.

During the reactor start-up, the inspector reviewed the Switching and Tagging log, the Maintenance Request log, control room panel system lineups, the Operating Log, and the Shift Turnover log. The inspector conducted a thorough walkdown of control room equipment panels and questioned operators about plant equipment status and start-up prerequisites.

Operators were knowledgeable and demonstrated a questioning attitude. Shift personnel and plant management were actively involved in start-up activities and the resolution of the problem with control rod 34-15 received an appropriate level of review. The reactor start-up was conducted professionally and actions taken by plant personnel were deliberate, timely, and demonstrated a strong safety perspective.

B. Control Room Panel Walkdown

The inspector conducted frequent control room observations and walkdowns on control room equipment operating and status panels. The inspector routinely reviewed the Switching and Tagging Log, the Maintenance Request Log, the Shift Turnover Log, the Operations Department Night Orders Notebook, the Operating Log and preliminary Potential Reportable Occurrence reports.

Control room operators consistently demonstrated detailed level-of-knowledge regarding ongoing plant evolutions and equipment status. In general, procedures for evolutions and surveillances were frequently referred to by operators during execution of the procedure.

On October 10, the inspector noted that the recirculation pump discharge bypass valves were not open during Residual Heat Removal (RHR) shutdown cooling operations. Procedure OP 2124 "Residual Heat Removal System," requires that approximately 10 minutes after the start of the RHR pump, the reactor recirculation pump in the loop that shutdown cooling is established shall be shutdown and the recirculation pump discharge valve RV-53 A(B) closed. The recirculation pump discharge bypass valve RV-54 A(B), according to the procedure, should remain open to provide loop cooling. In addition, when the bulk reactor coolant temperature is less than or equal to 190 degrees F, the other reactor recirculation pump is required to be shutdown, the pump discharge valve closed and the discharge bypass valve should be opened. This was the second time the inspector noted that recirculation pump discharge bypass valves were shut during RHR shutdown cooling operations.

The recirculation pump discharge bypass valve provides a path for loop cooling and prevents thermal stratification in the recirculation system loop. This is particularly important during significant cooldown operations. In these particular instances the RHR shutdown cooling system had been utilized routinely to maintain bulk reactor coolant temperature during the refueling shutdown, RHR shutdown cooling had been shutdown for a short period of time, and a small cooldown gradient was established. The inspector concluded that the safety significance of the apparently mispositioned valves was minor.

The inspector concluded that the cause of these events was lack of attention-to-detail by the control room operators. The inspector also noted that these events occurred near the end of the refueling outage, when operators had become familiar with operating the RHR system in the shutdown cooling mode. The inspector discussed these events with the Operations Supervisor and determined that some ongoing system and valve pre-startup testing may have contributed to the as-found recirculation system status. The inspector concluded that additional procedural clarity is needed to provide operators specific guidelines for the normal position of the recirculation discharge bypass valve during shutdown cooling operations.

C. <u>Reactor Trip During Weekly Turbine Emergency Governor Trip Test</u>

On November 4, the reactor tripped during performance of the weekly turbine emergency governor trip test contained in operating procedure OP 4160, Rev. 17, "Turbine Surveillances." The emergency governor trip test demonstrates that the emergency governor overspeed (110 percent) protection is functional. The trip test is accomplished by pulling the emergency governor trip test switch handle to the "Lockout" position and then rotating the switch clockwise, through the reset position, to the trip position. The lockout position hydraulically prevents actuation of the emergency governor trip. Indicating lights, above the switch handle, inform the operator if the governor is tripped or reset.

While performing this surveillance on November 4, the emergency governor trip test switch was pulled to the lockout position and rotated to the trip position. The trip light did not energize and the reset light remained illuminated. The computer printout did not indicate the turbine trip-emergency trip valve in the trip condition. Operators returned the switch to the vertical position, noted no change in the status of the reset and trip lights, and performed the emergency governor trip test again. No change in reset or trip indicating lights was noted. The red reset light remained energized and the green trip light remained de-energized throughout the test. The emergency governor trip/test switch handle was returned to the vertical position. After verifying procedural requirements for returning the switch to normal were met, the operators pushed the switch and released the lockout. A turbine control valve fast closure scram signal was received as a result of the turbine trip. The reactor was operating at approximately 93.4 percent of rated thermal power prior to the reactor trip. All reactor plant systems functioned as expected during the plant transient.

A corrective maintenance request (CMR 90-3274) was generated to investigate the cause and, if necessary, repair the emergency trip assembly. A collar and spring assembly on the linkage between the emergency governor trip piston and the position indicating limit switch had become dislocated within the turbine front standard. This resulted in a failure of the limit switch to follow the trip piston movement which resulted in failure of the trip light to illuminate. Two other collars were found loose and were tightened. Post maintenance testing of the emergency governor at the front standard and in the control room was completed satisfactorily.

Corrective troubleshooting provided a detailed explanation for failure of the emergency governor trip light to accurately reflect the condition of trip piston when in the trip position. However, the cause of the failure of the emergency governor to reset could not be positively determined. Mechanical binding or switch sensitivity near the reset position may have contributed to this failure. Post maintenance testing rigorously addressed operator manipulation of the switch, and was particularly sensitive to the amount of time the switch remained in the trip or reset position.

The inspector concluded that control room operators responded appropriately to the plant transient and that maintenance personnel actions to correct the failure were effective. Post maintenance testing provided adequate assurances that maintenance activities had thoroughly addressed and corrected the root cause of the failure.

D. Loss of Advanced Off-Gas Auxiliary Equipment

On October 19, Advanced Off-Gas (AOG) control power fuse FU-1-80 on control room panel (CRP) 9-50 blew, resulting in the loss of some AOG auxiliary equipment including the condensate AOG booster pump, AOG drain tank pumps, and AOG ventilation. When replaced, the fuse repeatedly blew. Control room operators and instrument and control (I&C) personnel traced the cause of the blown fuse to the actuating level switch for the AOG drain pit sump pump.

The AOG system delays the release of gaseous radioactive effluents to the environment so that the total radiation exposure to persons outside the controlled area is as low as practical. During this incident, the AOG system continued to adequately delay the releases of gaseous radioactive effluents to the environment.

Three entries into the AOG drain pit room were made to perform investigation and the subsequent correction of the cause of the blown control power fuse. The initial entry survey results indicated radiation and contamination levels below those required for issuance of a radiation work permit (RWP). During the second entry into the drain pit, personnel took appropriate radiological precautions consistent with the initial survey results. While second entry personnel were still in the AOG drain pit, the control power fuse blew and the AOG condensate booster pump, AOG drain pit sump pump and AOG ventilation stopped. The radiation protection (RP) technician providing work coverage noted climbing dose rates and ordered the two personnel to evacuate the area. The total time between AOG control power fuse failure and personnel egress was estimated at two minutes.

An RWP was prepared for the third entry into the AOG drain pit. Personnel radiological precautions for entry were defined and three air sampling techniques were used to provide a representative composite of the actual airborne exposure environment in the AOG drain pit. The AOG drain pit had become pressurized and upon opening the AOG drain pit door, some of the atmosphere within the drain pit room exhausted to the atmosphere. The I&C personnel disconnected the level switch for the AOG drain pit sump pump. The level switch was determined to be the cause of the repeated blowing of the control power fuse. The individuals immediately vacated the drain pit room. The estimated time that the individuals were in the drain pit room was two minutes. The AOG drain pit room door was opened only for personnel ingress and egress.

The three personnel directly involved in the incident were contaminated, primarily as a result of noble gas build-up in the AOG drain pit. The personnel were appropriately released from the RP checkpoint and directed to obtain a whole body count.

VY concluded that an uncontrolled release of radioactivity from the AOG drain pit room occurred when the AOG drain pit room door was opened for the third entry. Based on air sample results, a calculated atmospheric dispersion factor, a conservative room volume estimate, and the assumption that the entire volume of room was exhausted to the atmosphere, Vermont Yankee concluded that the release to the environment was a factor of 500 below the 10 CFR 20 limit for radioactivity releases when averaged over an hour.

The inspector concluded that the control room operators and I&C personnel adeptly handled the out-of-normal AOG operation and responded appropriately to mitigate the effect on plant operations. The analysis of the uncontrolled release of radioactivity to the environment was technically sound and exercised conservative judgement.

The inspector identified a weakness in the area of personal/industrial safety. While radiological concerns were reasonably addressed, the inspector concluded more emphasis should be placed on evaluating non-radiological safety hazards. Specifically for this event, more careful consideration should have been placed on personal hazards associated with operations in a confined space with a potentially depleted oxygen atmosphere.

E. Reactor Building Inspections

The inspector conducted routine tours of the reactor building and radwaste areas. On several occasions the inspector accompanied the reactor building auxiliary operator on his rounds. In general, the inspector determined that the reactor building auxiliary operators were thorough, exhibited appropriate radiological work practices, and demonstrated a questioning attitude.

On November 15, the inspector noted that lights were out in the northeast side of the lower torus area. These non-functioning lights were located over a radiologically roped-off area. The inspector concluded that this situation presented a personnel industrial safety and potentially a radiologically safety concern. On a subsequent tour, the inspector noted that the area was properly illuminated.

On November 19, the inspector noted that a fire extinguisher in the high pressure coolant injection (HPCI) corner room required recharging. The fire extinguisher was replaced and a subsequent Vermont Yankee inspection identified two other reactor building fire extinguishers that required replacing. The inspector concluded that the licensee response to this inspection finding was consistent with their strong fire protection safety culture.

Also on November 19, the inspector noted that a radiological posting at an entrance to a contaminated area in the lower torus area was not properly positioned. The barrier rope which held the sign was laying on the ground next to the step-off pad, apparently left there after personnel exited the contaminated area. This has been a recurring problem and the licensee is exploring long-term corrective actions. On subsequent tours, the inspector noted that the area had been decontaminated and the boundaries removed.

On November 26, the reactor building auxiliary operator discovered the roof access door in the radwaste building open. According to the auxiliary operator log sheets, the door is required to be shut and locked. A label on the door states to contact Radiological Protection (RP) prior to using the door. The auxiliary operator immediately contacted the control room, informed the senior operator, and ascertained whether work was in-progress on the radwaste building roof. The auxiliary operator shut the door, verified that the door was locked, and was instructed by the control room to contact RP at the conclusion of his rounds.

The inspector concluded that these events were of minor safety significance and immediate corrective actions by the licensee were effective.

3. Radiological Controls (71707)

3.1 Inspection Activities

Compliance with the radiological protection program was verified on a periodic basis.

3.2 Inspection Findings and Review of Events

A. Efforts to Enhance Radiological Protection Program

On November 13, the licensee formed a group to work full time on an integrated plan to enhance radiological protection (RP) practices. The group, assembled from a disciplinary cross-section of Vermont Yankee employees, is chaired by the Radiation Protection Supervisor.

The group trained and prepared for approximately two weeks and is expected to travel to several other nuclear facilities to observe RP practices. Based on these observations, the group will develop an enhanced radiological protection program and establish performance indicators to monitor program success. During the inspection period, the licensee also distributed a RP/contamination survey developed to help assess Vermont Yankee personnel attitudes toward controlling radioactive contamination. Responses to the survey may help reshape the licensee's policy and reduce personal contamination incidents.

The inspector concluded that these efforts to enhance and improve the RP program and RP practices are laudatory.

B. Routine Inspection Findings

The inspector conducted frequent tours of the Radiological Control Area (RCA) inspecting many Radiological Work Permit (RWP) areas. During these tours, the inspector assessed the effectiveness of the radiological housekeeping program, reviewed radiological posting requirements, and observed radiological work practices.

The inspector concluded that the radiological housekeeping is adequate. Recent management attention in this area was evident. Some minor discrepancies with radiological postings were noted and immediately corrected. The inspector found workers adhering to established radiological work practices.

4. Maintenance and Surveillance (62703, 92700)

4.1 Maintenance Inspection Activity

The inspectors observed selected maintenance activities on safety related equipment to ascertain that these activities were conducted in accordance with approved procedures, Technical Specifications, and appropriate industry codes and standards.

4.2 Maintenance Observations

A. Feedwater Check Valve Flaws

During the 1990 refueling outage, Vermont Yankee replaced the two inboard feedwater lift check valves (valves 28A and 28B) with swing check valves. These swing valves are the same design as the existing first outside containment valve in each feedwater line (valves 27A and 96A). Vermont Yankee committed to replacing or repairing the 28B valve when, during the 1989 refueling outage, visual cracking observed in the stellite wear pads in the piston guide portion of the valve exceeded ASME Code Section XI limits. In 1989, all four feedwater check valves of similar lift check design were ultrasonically inspected to determine flaw depths (there are three feedwater check valves in series in each of the two feedwater lines). NRC assessment and analysis of the Vermont Yankee activities associated with the flaw in valve 28B is contained in NRC inspection report 50-271/89-02.

Vermont Yankee completed in-service testing and inspection of the two remaining lift check valves (valves 27B and 96B) using ultrasonic transducers with improved capabilities for examining cast carbon steel.

These inspections identified flaws in both check valves. In valve 96B, the maximum flaw depth was 0.15 inches, which is within the acceptance criteria of ASME Code Section XI. In valve 27B, the maximum flaw depth was 0.50 inches. The wall thickness at the location of the 0.50 inch flaw is approximately 2.6 inches. This flaw is contained within the width of the stellite wear pad, is smaller than the flaw previously evaluated for the 28B valve, and is bounded by previous engineering analysis.

Vermont Yankee letter to the NRC dated October 4, (BVY 90-097), discussed the flaw associated with valve 27B and requested NRC approval for an additional cycle of operation with the flaw on valve 27B. In that letter, Vermont Yankee committed to repair or replace valve 27B during the next refueling outage. In addition, Vermont Yankee committed to install leak monitoring tape on valve 27B. This system is capable of detecting very small amounts of leakage (0.1 gpm or greater) and is provided with an alarm in the main control room. This system is in addition to the normal leak detection system that monitors the steam tunnel area.

The NRC responded to Vermont Yankee's request for approval to restart in a letter dated October 10. The NRC acknowledged the flaw in valve 27B exceeded ASME Code Section XI acceptance criteria and concluded that the engineering analysis for valve 28B bounds the flaw for valve 27B. The analyses for valve 28B was found acceptable in an NRC Safety Evaluation dated April 19. Although valve 96B contained a flaw within the ASME Code Section XI acceptance limits, the NRC staff believed that the valve should be reinspected during the next refueling outage to verify the fracture mechanics analysis. The NRC staff concluded that feedwater check valve 27B will be capable of functioning for one additional fuel cycle.

On October 14, prior to startup from the refueling outage, the inspector verified the leak monitoring tape for valve 27B was installed.

The inspector concluded that the Vermont Yankee Engineering Support Department (ESD), Yankee Atomic-Bolton, and plant management interacted in a timely and effective manner to provide a resolution to this issue. The commitments to replace or repair valve 27B and to install an enhanced leak detection system for valve 27B demonstrate a deliberate conservative safety perspective.

B. Teflon Seals in the Containment Personnel Airlock (LER 90-14)

On October 10, VY determined that the containment personnel airlock was originally fitted with Teflon material. The Teflon material is located in airlock pressure equalizing valves and in shafts which penetrate the bulkhead, forming a portion of the primary containment leakage path. VY received this information from the airlock vendor, Chicago Bridge and Iron Services, Inc. (CB&I).

The personnel access airlock has two gasketed doors in series designed to maintain integrity under drywell design pressure. The doors are mechanically interlocked to ensure that at least one door is locked at all times when primary containment is required. The locking mechanisms are designed so that a tight fit seal will be maintained when the doors are subjected to internal pressure. The seals are subject to 10 CFR 50 Appendix J Type B leak rate test.

The Teflon that existed in the airlock seals had characteristic material damage limits in the range of 10E4 to 10E6 RADS. The VY design basis accident radiation dose can exceed (by one order of magnitude) that needed for Teflon material damage. Thus the Teflon material could fail under accident conditions and compromise primary containment integrity.

On October 10, maintenance personnel replaced the Teflon material shaft seals with graphite and on October 13, a temporary modification was completed which replaced the equalizing valves with valves that do not contain Teflon. Subsequently, a local leak rate test was completed successfully.

The inspector concluded that response to the vendor notification was appropriate, corrective actions adequate, and that the event was properly reported to the NRC.

4.3 Surveillance Inspection Activity (61726, 90712, 92700)

The inspectors performed detailed procedure reviews, witnessed in-progress surveillance testing, and reviewed completed surveillance packages. The inspectors verified that the surveillance tests were performed in accordance with Technical Specifications, approved procedures, and NRC regulations.

The surveillance testing activities inspected were effective with respect to meeting the safety objectives of the surveillance testing program.

4.4 Surveillance Observations

A. Excess Flow Check Valve Functional Test

On October 10, while performing operational procedure (OP) 4378, Rev.17, "Excess Flow Check Valve Functional," three separate Group I isolations actuated as a result of testing flow transmitters FT-6-51A, FT-6-51B, and FT-6-51C. These flow transmitters (4 total; FT-6-51D is the fourth flow transmitter which transmits a signal representing main steam line flow to the feedwater control system) normally transmit a signal representing main steam line flow to the feedwater control system. Four differential pressure transmitters, located downstream of each flow transmitter, tap off a common sensing line. The pressure transient resulting from testing the flow transmitter was sensed as a high steam flow condition, and actuated a Group I isolation. Three separate Group I isolations resulted from performance of this surveillance.

The initial Group I isolation occurrence was attributable to a weakness in the testing procedure. The inspector reviewed the subsequent procedural change and determined that the changes were effective to prevent recurrence of this event. The inspector accompanied instrument and control (I&C) technicians during performance of the excess flow check valve functional test for pressure transmitter (PT) 2-2-24A, differential pressure transmitter (DPT) 2-111A (LO), and flow transmitter FT-6-51D. One instrument, DPT 2-111A (LO), initially failed the functional test. After mechanical agitation of the associated excess flow check valve, the functional test for DPT 2-111A (LO) was performed successfully.

The licensee evaluated this event (Potential Reportable Occurrence (PRO) 90-55) and determined that it was not reportable. The licensee concluded that the Primary Containment Isolation System (PCIS) was not capable of performing its intended function and was considered inoperable. The inspector reviewed the licensee reportability determination and concluded that although primary containment was not required and not maintained, the PCIS actuated. This actuation was not part of a preplanned sequence nor a procedurally anticipated event and therefore should be reported under 10 CFR 50.73. The licensee understood the inspector's conclusion and agreed to formally report the event to the NRC.

The inspector concluded that the response to the event was appropriate. Procedural revisions were timely and effective and the observed excess flow check valve functional tests were performed by knowledgeable technicians. The inspector determined that operators and technicians involved in the excess flow check valve functional tests which resulted in Group I isolations could have exhibited a more questioning attitude and thus precluded actuation of the second and third Group I isolations.

B. Failure of 10 CFR 50 Appendix J, Leak Rate Testing (LER 90-12)

During the 1990 refueling outage, two containment isolation valves (FDW-96A, PCAC-6B) were found to have seat leakage above that permitted by Technical Specification (TS) 3.7.A.4. The measured leakage for each valve exceeded test apparatus capacity. Allowable single valve leakage is limited to 0.522 pounds-mass per hour.

On September 3, as a result of the leakage through check valve FDW-96A, the sum total leakage for Type B (penetrations) and Type C (valves) exceed that allowed by 10 CFR 50 Appendix J. Appendix J limits the total Type B and Type C leakage to 0.60 La.

The failure of feedwater check valve FDW-96A (an Anchor Darling check valve) was due to erosion of the elastomeric seat when the valve was subject to low flow conditions. These low flow conditions exist during Feedwater system startup and during outages when only Reactor Water Cleanup system and Control Rod Drive system return flow is present in the feedwater system. Consequently, the elastomeric seats were removed from all four Anchor Darling feedwater check valves. The resulting Stellite to Stellite seating surface is not as susceptible to wear due to low flow conditions.

As a result of the leak rate test failure of FDW-96A, VY evaluated the safety impact of removal of these resilient, elastomeric seats. On November 2, in a letter to the NRC (BVY 90-108), VY notified the NRC of a change regarding one aspect of the valve replacement effort implemented under 10 CFR 50.59 that affects the bases of an NRC Safety Evaluation Report issued in support of license Amendment No. 122. As a result of recent engineering evaluation, VY concluded that inclusion of the resilient seat does not improve, and may, as a result of certain low flow operational conditions, adversely affect the leak tightness of these valves when compared to hard seating capabilities.

The root cause of the failure of PCAC-6B was the failure of the elastomeric seat. Excessive flexing of the seat resulted in localized ripping. The compression gasket that compresses the seat material was found to be misaligned.

VY completed maintenance on valve FDW-96A and valve PCAC-6B. The valves were repaired and retested to ensure leakage was within allowable limits prior to startup.

The inspector concluded that the licensee response to this event was appropriate, that corrective actions were adequate and that the event was properly reported to the NRC.

C. Main Steam Safety Relief Valve Accumulator Leakage Test Failure (LER 90-13)

On September 5, the "C" main steam relief valve accumulator assembly was found to have check valve seat leakage that exceeded the acceptance criteria specified in the surveillance procedure. The other three main steam relief valve accumulator assemblies passed the leak tests. The seat leakage would result in pressure bleed down from the accumulator during a loss of nitrogen supply. In this event, the accumulator could not provide assurance that the relief valve would be held open.

The accumulator assembly exceeded the acceptance criteria due to a check valve not properly seating. The check valve was disassembled, internals cleaned, and a new resilient seat installed. On September 20, the assembly was successfully retested. The root cause for the leakage was the build-up of dirt/corrosion products on the check valve seat. The dirt/corrosion product was from the containment atmosphere system carbon steel piping and is believed to have built-up when the system was supplied with air. The system is currently supplied with clean, dry nitrogen.

Similar events were reported to the NRC in 1986 (LER 86-09) and 1987 (LER 87-09). During the 1990 refueling outage new stainless steel piping, with two in-series particulate filters, was installed. The carbon steel accumulators were replaced with stainless steel accumulators sized at twice the capacity. This provided an additional safety margin and should insure that new corrosion products are not generated downstream of the filters. VY committed to replace the relief valve filters each refueling outage.

The inspector considered back-up alternatives for a loss of nitrogen supply, the design capacity of each relief valve (each relief valve provides one third of required relief capacity), availability of the containment air system, and the operability of the three remaining safety relief valves. The inspector determined that adequate assurance, provided by a defense-in-depth approach, existed to ensure the depressurization function provided by these safety relief valves would have been performed successfully in spite of this leakage problem.

5. Security (71707, 90712, 92700)

5.1 Observations of Physical Security

Compliance with the security program was verified on a periodic basis, including the adequacy of staffing, entry control, alarm stations, and physical boundaries.

5.2 Access Control

Two events involving suitability of personnel for site access were reported to the NRC. On October 29 and on November 14, VY determined that individuals badged for site access were untrustworthy. The decision to grant site access was based, in part, on erroneous information supplied by each individual during access control processing. In both cases, FBI fingerprint check results conflicted with information furnished by the individuals. One of these individuals was granted site access under a business agent "Good Guy" letter. Subsequent interviews with these individuals revealed that they had intentionally misrepresented themselves on access control processing documentation. While the areas of misrepresentation would have likely precluded site access, VY's final declaration regarding unsuitability for access was based on demonstrated untrustworthiness. These individuals were employed by VY contractors during the 1990 refueling outage and had left the site before to receipt of the fingerprint check results.

Vermont Yankee recently experienced several events regarding access control problems (NRC inspection reports 50-271/90-10 and 90-11). VY recognized the need for further evaluation and initiated revisions to their procedure for contractor screening requirements for unescorted access (VYP 325). This revision is specifically designed to make VY requirements for contractor unescorted access consistent with Nuclear Management Resources Council (NUMARC) guidelines.

The inspector concluded that these events were conservatively reported and that management actions are a positive step to reduce recurrence of similar events.

6. Engineering and Technical Support (71707, 90712)

6.1 Control Rod Drive Cap Screws

The Vermont Yankee reactor has 89 control rod drives (CRD). Each drive is bolted to a housing at the bottom head of the reactor vessel with eight cap screws.

In May 1988, General Electric Company (GE) issued a notice (RICSIL No. 019) to owners of GE Boiling Water Reactors informing them that circumferential crack indications and pitting corrosion were found in the area directly below the cap screw head. The indications were detected during the ASME Section XI

Code required visual in-service examination of the CRD cap screws. The condition was attributed to corrosion of magnesium sulfide inclusions in the cap screw material. In March 1989, GE issued a follow-up Service Information Letter (SIL No. 483). GE recommended that suspect cap screws, as determined by the ASME visual inspection, be magnetic particle or liquid penetrant tested.

During the 1990 refueling outage, 12 control rod drives, containing a total of 96 cap screws, were replaced. A sample of thirty-two cap screws were removed from service and replaced with new cap screws. The remaining 64 cap screws were visually inspected and 13 additional cap screws were discarded and replaced with new cap screws because of mechanical damage, such as burred threads,head damage, etc. No cracking was visually detected in any of the 64 cap screws.

The 32 cap screws removed from service were submitted for magnetic particle examination. A highly sensitive fluorescent inspection technique was used and 17 cap screws were found to have linear flaw indications. The maximum flaw depth was 0.077 inches, with the majority of the indications between 0.036 inches and 0.046 inches deep.

The cap screw linear flaw issue was reviewed at the Plant Operations Review Committee (PORC) meetings on October 10 and October 12. Following the PORC meeting on October 10, VY management conservatively decided to seek NRC guidance in resolving this issue and permission for subsequent restart. The NRC concurred with VY's conclusion that the condition of the bolts, based on the observed flaws and bounded by a conservative evaluation, presented no significant or imminent safety concern.

In a VY letter to the NRC dated October 19, VY formally described a conservative bounding evaluation for the observed flaws and requested relief from ASME Code Section XI Code requirements. Specifically, VY requested NRC approval for operation with components that contained relevant indications but are acceptable by analysis. VY also requested relief from the sample expansion requirements of Section XI. VY committed to replace all cap screws during the scheduled 1992 refueling outage. The NRC found this acceptable.

The inspector concluded that the conservative approach, ensuring that the cap screw issue received appropriate levels of review, was noteworthy. Because the visual inspection did not detect flaws, the inspector questioned the adequacy of the ASME Code Section XI required visual inspection to provide adequate assurance that these types of flaws will be detected. The highly sensitive fluorescent inspection technique used to detect the linear flaw indications was not required by ASME Code. The inspector reviewed the bounding flaw evaluation and determined that the engineering assumptions were conservative.

6.2 Cycle 15 Core Operating Limits and Core Component Qualification

The inspector reviewed Vermont Yankee Cycle 15 Core Operating Limits Report and the Vermont Yankee Cycle 15 Core Performance Analysis Report. The Core Operating Limits Report was submitted to the NRC in accordance with Technical Specification Section 6.7.A.4 under cover letter dated October 11 (BVY 90-096). The Core Operating Limits Report provides cycle-specific thermal limits including limits for the maximum average planar linear heat generation rate, maximum linear heat generation rate, and the minimum critical power ratio. The inspector was specifically interested in the analytical treatment of the four qualification fuel bundles manufactured by Advanced Nuclear Fuels.

During the 1990 refueling, VY discharged 128 irradiated fuel bundles and inserted 128 new fuel bundles. The resultant core consisted of 128 new fuel bundles and 240 irradiated fuel bundles (Cycle 13 and Cycle 14). One fuel bundle was reconstituted with two new fuel pins.

Four qualification fuel bundles manufactured by ANF were placed into the core at positions 05-20, 39-20, 05-26, and 39-26. These locations are expected to be nonlimiting with respect to thermal limits throughout the entire cycle.

The ANF fuel bundles differ from General Electric (GE) bundles in the following ways: 1) the average bundle enrichment is lower; 2) the fuel pins are smaller in diameter and consequently there are more fuel pins per bundle; and 3) a large square inner water channel is used rather than a large round water rod. The ANF bundles are designed to match the GE bundles neutronically and thermal-hydraulically. With some conservative adjustments, the ANF bundles will be monitored as a GE bundle (BP8DW311 bundle).

VY also replaced eight standard control blades with eight GE Marathon control blades. The Marathon control blades are longer life control blades which utilize a combination of boron carbide (B4C) and hafnium as neutron-absorbing materials. NRC approved the use of both materials in VY control blades and the change was incorporated into License Amendment No. 123. The control blades have been designed to be a direct replacement for any of the current GE control blade assemblies. During core component qualification, the Marathon control blades are located in nonlimiting locations with respect to shutdown margin.

The inspector concluded the treatment and operating limitation analysis of the ANF fuel bundles and the GE Marathon control blades was conservative.

6.3 <u>(Closed) UNR 90-03-002: Use of Teflon Tape for Rosemount Transmitter Thread Sealant</u>

In NRC Inspection Report 50-271/90-03, the inspector identified a concern regarding the consistent and the appropriate use of Teflon tape as a thread sealant for environmentally qualified (EQ) Rosemount transmitters. During the 1990 refueling outage, Vermont Yankee conducted a walkdown of all environmentally qualified Rosemount transmitters. These transmitters are qualified to provide additional operational assurances during high energy line break (HELB) conditions.

Forty-two Rosemount transmitters were inspected. Vermont Yankee QC inspectors determined that 31 Rosemount transmitters were not in compliance with EQ requirements because the electrical conduit connections were not sealed with a thread sealant. Vermont Yankee determined that this equipment condition was not reportable (Potential Reportable Occurrence 90-59) because the improved sealing characteristics of the modified conduit hubs would be sufficient to restrict the penetration of moisture into the electronic enclosure of the transmitter housing. Non-conformance Report (NCR) 90-18 was issued to document and correct this condition.

NCR 90-18 addressed the inspector's additional concern for breakdown of Teflon material in a high radiation environment. In a HELB scenario, radiation levels are below the Teflon breakdown threshold (1.7E4 RADS). During the loss of coolant accident scenario, radiation levels would exceed the Teflon breakdown threshold; however, because these transmitters are located outside the primary containment, moisture intrusion would not be a concern.

The inspector verified, on a sampling basis, that Teflon tape was installed on the environmentally qualified Rosemount transmitters. Vermont Yankee identified that 31 Rosemount transmitters were not environmentally qualified in violation of 10 CFR 50 49 and took prompt and appropriate corrective actions. This event was of minor safety significance and the violation is not being cited because the criteria specified in Section V.A. of the Enforcement Policy were satisfied (NCV 50-271/90-15-001).

The inspector determined that long-term corrective actions identified in the NCR 90-18 (i.e. to review the as-qualified configurations of equipment to ensure other requirements, added after equipment installation, are tracked and in-place) are appropriate.

6.4 LER 90-10 Supplement 1, Failure to meet Technical Specifications for Diesel Generator Testing

LER 90-10 was discussed in NRC Inspection Report (IR) 50-271/90-10. In IR 90-10, the inspector concluded that information garnered after submission of LER 90-10 required further licensee and NRC analysis. The inspector also addressed several areas of concern regarding this event. Specifically, the inspector was concerned about the effectiveness of the interface between engineering support activities and plant operation activities, the timeliness of incorporating changes pertaining to FSAR requirements into all applicable plant procedures, the accuracy of vendor supplied information, and the adequacy of technical review.

Supplement I to LER 90-10 clarified information presented in the original LER submission. The supplement adequately described licensee actions with regard to testing the diesel generators beyond the continuous rating of the diesel generators. The inspector concluded that causal analysis of the event was complete and proposed corrective actions appropriate.

Some concerns expressed by the inspector were not thoroughly addressed by the LER supplement. The effectiveness of the interface between engineering support activities and plant operation activities will continue to be evaluated by the NRC. The inspector was concerned about the mechanism for ensuring timely incorporation of non-routine procedural changes pertaining to FSAR requirements into all applicable plant procedures. LER 90-10 is closed, however, resolution of the later concern remains unresolved (UNR 50-271/90-15-002).

7. Safety Assessment and Quality Verification (40500, 71707)

7.1 Outage Organization and Management

The overall responsibility for the day-to-day conduct of the 1990 refueling and maintenance outage rested with the Outage Manger. The Outage Manager reported directly to the Plant Manager. The Outage Manager position was filled by the Operations Superintendent, but rotated among senior plant managers to provide continuous outage coverage.

An Operations Planning Group (OPG) was established, in part, to track items requiring resolution in order to achieve a particular milestone and ensure that resolution responsibilities are assigned and successfully completed. Additional OPG responsibilities included review and implementation of tagging requests, coordination of system retests, tracking of system work progress and scheduling requests.

The inspector concluded that outage planning and management were extremely effective. The completion of the outage only days behind schedule was noteworthy when considering the scope of work accomplished and the ambitious 6 week schedule. The outage organization demonstrated versatility and flexibility by efficiently resolving complex personnel and equipment issues. The outage organization emphasized safety and immediately took steps to correct or address deteriorating safety conditions. Senior management adeptly delegated authority and attentively addressed individual concerns. Frontline supervisors provided valuable input into outage related issues and enjoyed wide decision-making responsibility.

8. Licensee Event Reports (LER), Periodic and Special Reports, Unresolved Item Followup (90712, 90713, 92700, 92701)

8.1 LERs

The inspector reviewed the licensee event reports listed below and determined that, with respect to the general aspects of the events: (1) the report was submitted in a timely manner, (2) the description of the event was accurate, (3) a root cause analysis was performed, (4) safety implications were considered, and (5) corrective actions implemented or planned were sufficient to preclude recurrence of a similar event.

A. LER 90-10, Supplement 1, "Failure to Meet Technical Specifications for Diesel Generator Testing" (See Section 6.4).

B. LER 90-14, "Potential Loss of Primary Containment Due to Teflon Seals in the Containment Personnel Airlock" (See Section 4.2.B).

C. LER 90-12, Supplement 1, "1990 10 CFR 50 Appendix J Type B and C Failure Due to Seal Leakage" (See Section 4.4.B).

D. LER 90-13, "Relief Valve Accumulator Failed Due to Check Valve Leakage" (See Section 4.2.C).

8.2 Periodic and Special Reports

The plant submitted the following periodic and special reports which were reviewed for accuracy and the adequacy of the evaluation:

-- Monthly Statistical Reports for plant operations dated October 10 and November 10, 1990.

- -- Vermont Yankee Cycle 15 Core Performance Analysis Report, YAEC-1749, August 1990 (See Section 6.2).

- -- Vermont Yankee Cycle 15 Core Operating Limits Report, Revision 0, September 1990 (See Section 6.2).

- -- Vermont Yankee Supplemental Report, Effluent and Waste Disposal Semi-annual Report for the third and fourth quarters 1989, including annual radiological impact on man for 1989; June 27, 1990.

- -- Vermont Yankee Nuclear Power Station Effluent and Waste Disposal Semi-annual Report for the First and Second Quarters 1990; August 29, 1990.

- -- Vermont Yankee Annual Radiological Environmental Surveillance Report for calendar year 1989.

8.3 Unresolved Item Followup

Unresolved items are items about which more information is required to ascertain whether they are acceptable, violations or deviations. Unresolved items discussed in this inspection report are tabulated below for cross references purposes:

- -- (Closed) NCV 90-15-001, Section 6.3
- -- (Open) UNR 90-15-002, Section 6.4
- -- (Closed) UNR 90-03-002, Section 6.3

9. Management Meetings (30702, 30703)

9.1 Preliminary Inspection Findings

At periodic intervals during this inspection, meetings were held with senior plant management to discuss preliminary inspection findings. A summary of findings for the report period was also discussed at the conclusion of the inspection and prior to report issuance. No proprietary information was identified as being included in the report.

9.2 Region Based Inspection Findings

Two Region based inspections were conducted during this inspection period. Inspection findings were discussed with senior plant management at the conclusion of the inspection. These inspection activities are described below:

Date	Subject	Rpt #	Inspector
10/29-11/02	Emergency Operating Procedures	90-16	T. Fish
11/05-11/09	Inservice Inspection Program	90-17	R. McBrearty

9.3 Licensee Requested Management Meetings

A. Vermont Yankee Corporation President Drop-In Meeting with Region I Regional Administrator on October 11

On October 11, Mr. J. G. Weigand, Vermont Yankee Corporation President and Chief Executive Officer, met with Mr. T. Martin, Region I Regional Administrator; Mr. W. Kane, Region I Deputy Regional Administrator; and Mr. E. McCabe, Acting Branch Chief, Reactor Projects Section 3 at the NRC Region I Regional Office in King of Prussia, PA. The meeting was held at the request of Mr. Weigand to discuss the refueling outage progress, recent allegations, and NRC inspection activities. In a letter to Mr. Weigand dated November 20, the NRC documented, for record purposes, major topics of discussion during this meeting.

B. Regional Management Meeting on November 19

On November 19, Vermont Yankee Senior Management met with selected Region I management and NRR personnel to discuss proposed methods for determining equipment and system operability, handling of allegations, and criteria for initiating 10 CFR 50.59 reviews. The issues of operability determinations and 10 CFR 50.59 reviews will be discussed in the NRC response to the Vermont Yankee response to the Notice of Violation contained in NRC Inspection Report 50-271/90-09. The presentation regarding the handling of allegations was for information and contained similar information as discussed in the meeting in section A above. A list of meeting attendees and hard copies of overhead slides used in the Vermont Yankee presentation are contained in Attachments A and B to this inspection report.

ATTACHMENT A

List of Attendees

Regional Management Meeting November 19, 1990

NRC Attendees

C. Hehl, Director, Division of Reactor Projects (DRP)
J. Johnson, Chief, Projects Branch 3, DRP
J. Rogge, Chief, Reactor Projects Section 3A, DRP
H. Eichenholz, Senior Resident Inspector
J. Durr, Chief, Engineering Branch, Division of Reactor Safety (DRS)
E. Gray, Chief, Materials Program Section, DRS
S. Chaudhary, Senior Reactor Engineer, DRS
M. Fairtile, Project Manager, Project Directorate (PD) I-3, Office of Nuclear Reactor Regulation (NRR)
D. Wigginton, Project Manager, PD IV-1, NRR
W. Hodges, Director, DRS
P. Drysdale, Senior Reactor Engineer, DRS

Licensee Attendees

W. Murphy, Senior Vice President, Operations
D. Reid, Plant Manager
R. Sojka, Operations Support Manager
M. Palionis, Senior Operations Engineer
J. Herron, Operations Supervisor
R. Pagodin, Technical Services Superintendent

Other Attendees

W. Sherman, State Nuclear Engineer, State of Vermont

ATTACHMENT B

Vermont Yankee Presentation Slides

November 19, Meeting

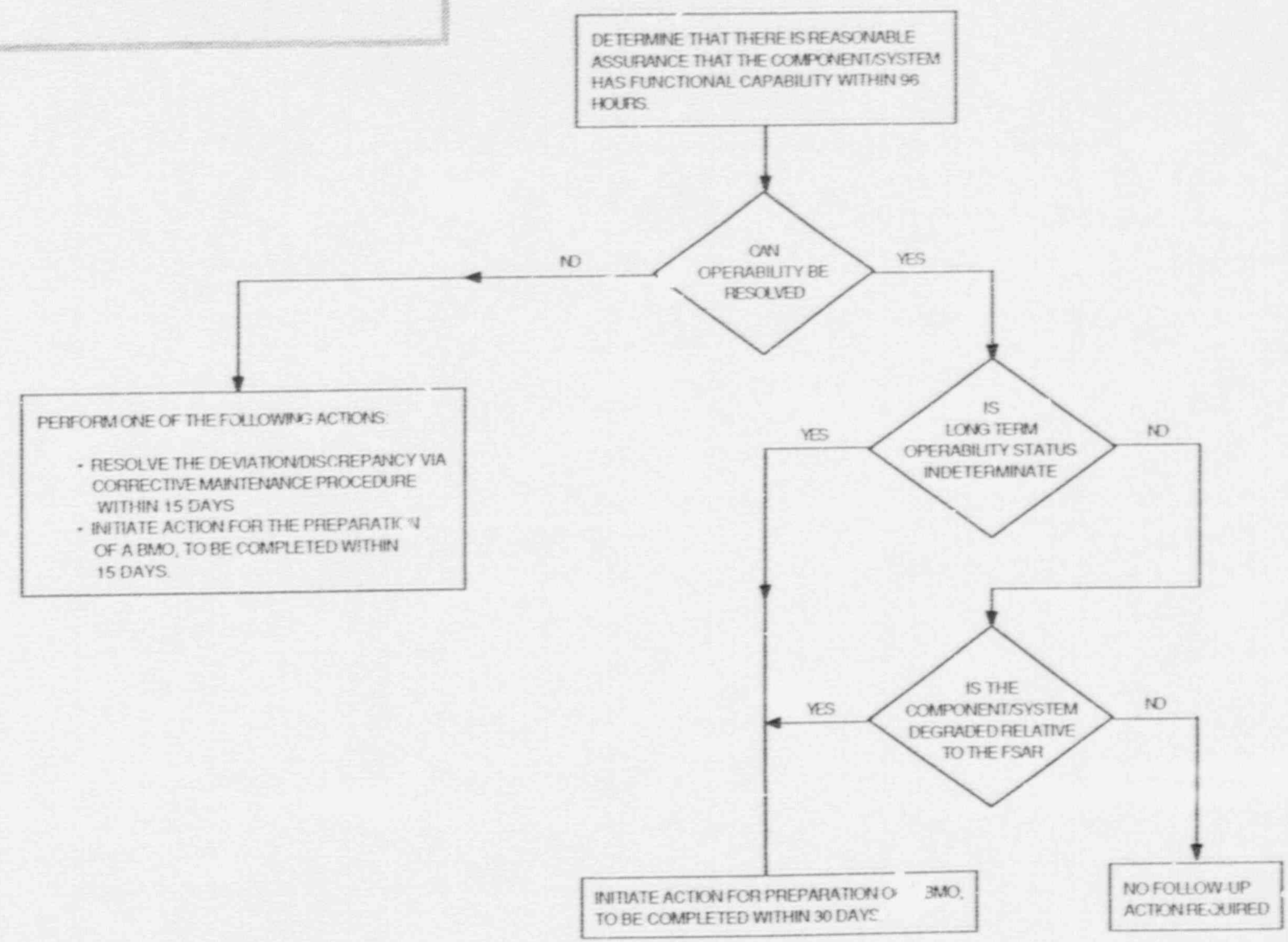
PURPOSE:

PROVIDE A SYSTEMATIC, COMPREHENSIVE APPROACH TO ADDRESS DISCREPANCIES / DEVIATIONS.

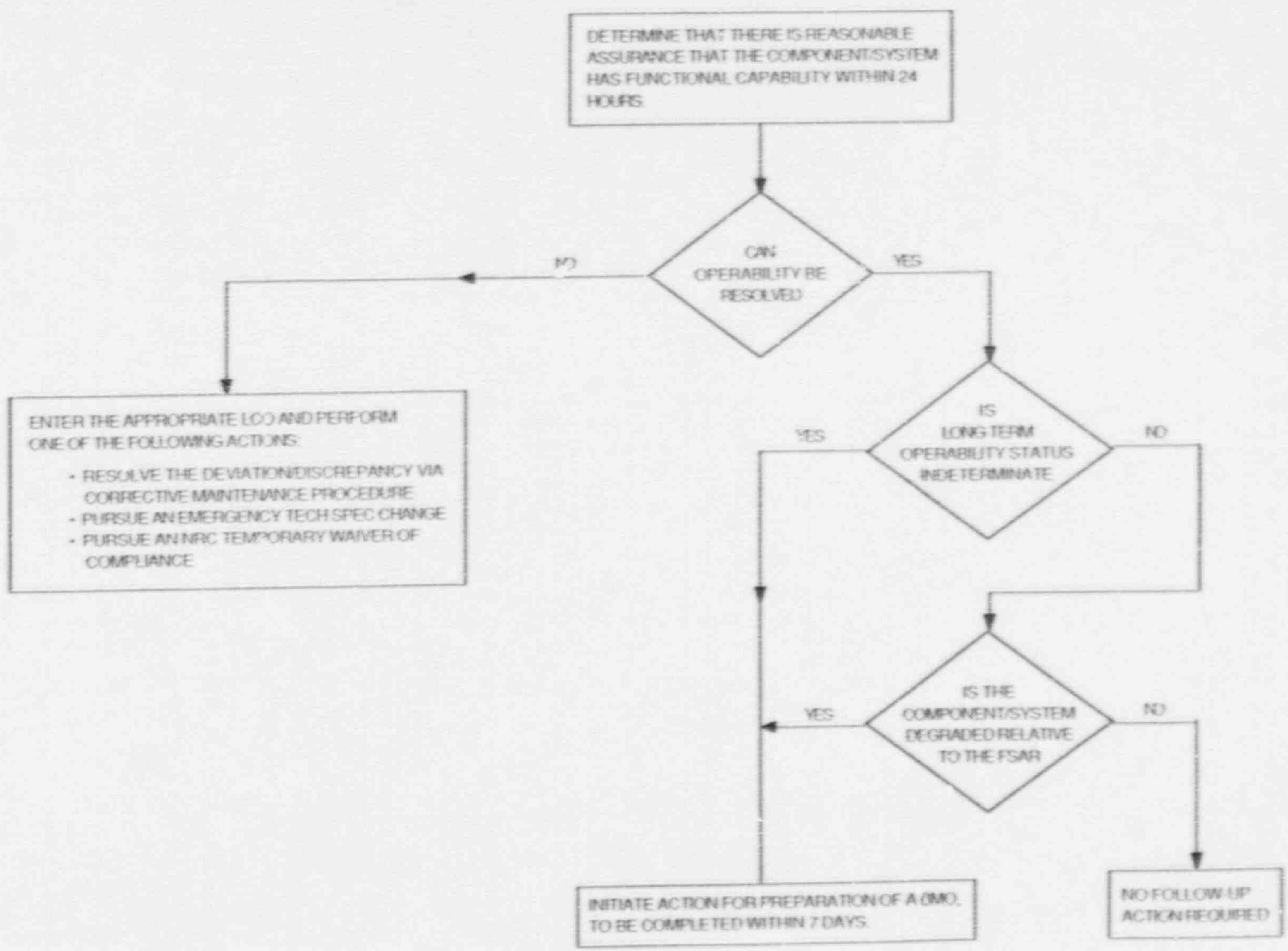
<u>Includes:</u>

- assessment of safety significance
- prioritization criteria
- managed approach to resolving discrepancies / deviations

OPERABILITY DETERMINATION FOR NON-TECH SPEC EQUIPMENT



OPERABILITY DETERMINATION
FOR
TECH SPEC EQUIPMENT



BASIS FOR MAINTAINING OPERABILITY

(BMO)

IF	THE BMO CANNOT BE COMPLETED WITHIN THE ALLOWABLE TIME PERIOD OR IT CANNOT BE DETERMINED THAT THERE IS REASONABLE ASSURANCE THAT THE HEALTH AND SAFETY OF THE PUBLIC IS PROVIDED FOR	THEN	ONE OR MORE OF THE FOLLOWING ALTERNATIVES SHALL BE PURSUED: - ENTER THE APPROPRIATE LCO IF A TECH SPEC COMPONENT/SYSTEM IS INVOLVED - PLACE THE PLANT IN A CONDITION SUCH THAT THE COMPONENT/SYSTEM IN QUESTION IS NO LONGER RELIED UPON FOR CONTINUED SAFE OPERATION

CONSIDERATION SHOULD BE GIVEN TO THE FOLLOWING DURING PREPARATION OF THE BMO:

1. FSAR DESIGN BASIS
2. FSAR DESIGN FEATURES
3. CODES & STANDARDS
4. PROBABILITIES
5. TECH SPECS
6. ADMIN CONTROLS
7. COMPENSATORY MEASURES
8. AVAILABLE MARGINS/CONSERVATISMS
9. TIME PERIOD ALLOWED
10. CORRECTIVE ACTIONS REQUIRED AND SCHEDULE FOR COMPLETION

IF	THE CONDITION ANALYZED IN THE BMO - IS TO BE MADE PERMANENT, OR - REPRESENTS A CHRONIC CONDITION, OR - WILL REMAIN IN EFFECT GREATER THAN 6 MONTHS	THEN	PREPARE A 10 CFR 50.59 SAFETY EVALUATION

EVALUATION OF SAFETY-RELATED MR'S

January 1989 to July 1989

Total MRs in sample:	2530
Total Safety-Related MRs in sample:	441

Safety-related / Non-Tech Spec:	80		
Operable, but... (> 30 days):	0	*	
Operable, but... (< 30 days):	10		
Inoperable (> 15 days):	1	*	
Inoperable (< 15 days):	4		
Safety-related / Tech Spec:	361		
Operable, but... (> 96 hours):	28	*	++
Operable, but... (< 96 hours):	44		

* requires BMO

++ average time between initiation of MR and release of equipment by SS was 36 days