U.S. NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION

Division of Reactor Inspection and Safeguards

Report No.: 88-200

Licensee: Vermont Yankee Nuclear Power Corporation
RD 5 Box 169
Ferry Road
Brattleboro, Vermont 05301

Docket No.: 50-271

Facility Name: Vermont Yankee Nuclear Power Station

Inspection Conducted: June 1-10, 1988

Inspector: A D Smith for 7/21/88
*J. E. Dyer, Team Leader, NRR/DRIS — Date Signed

A D Smith for 7/21/88
J. Macdonald, Resident Inspector, RI — Date Signed

Consultants: *K. M. Spencer (INEL-EGG)
*W. E. Gilmore (INEL-EGG)
*D. Waters (Beckman-Prisuta)
*P. R. Farron (NEC)

Accompanying Personnel: *G. Grant, SRI, RI

Reviewed by: 7/21/88
L. J. Norrholm, Chief, Team Inspection Appraisal and Development Section #1, DRIS, NRR — Date Signed

Approved by: 7/29/88
*C. J. Haughney, Chief Special Inspection Branch, DRIS, NRR — Date Signed

*Attended Exit Meeting on June 10, 1988

Scope:

An NRC inspection team performed a special, announced inspection to review the licensee's program for implementing Emergency Operating Procedures (EOPs) as required by NUREG 0737 Supplement 1. The inspection effort was performed in accordance with Temporary Instruction TI 2515/92 and evaluated the overall EOP development process, validated specific portions of the EOPs, reviewed EOP training and assessed the human factor engineering of system design and EOPs.

Results:

The inspection team concluded that the station operators could successfully accomplish the Vermont Yankee EOPs to manage the different postulated casualties within the plant. The team noted strengths with the material condition and cleanliness of the station, operator knowledge of EOPs, and supplemental activities undertaken by the licensee for accident management beyond the Emergency Procedure Guidelines (EPG) guidance. The deficiencies noted during the inspection included inadequate notification of the NRC for deviations from the NRC approved EPGs, improper implementation of the EOP verification and validation process, and incorrect development of the Procedures Generation Package (PGP) Writer's Guide.

Table of Contents

		PAGE
1.	INSPECTION OBJECTIVES	1
2.	BACKGROUND	2
3.	DETAILED INSPECTION FINDINGS	3
3.1	Emergency Operating Procedure (EOP) Program Evaluation	3
3.1.1	Procedure Generation Package (PGP) Review	3
3.1.2	Licensee Validation and Verification of EOPs	3
3.1.3	Response to IE Information Notices	4
3.1.4	Supplemental Safety Activities	5
3.2	Comparison of EOPs to Emergency Procedure Guidelines (EPGs)	5
3.2.1	Unjustified Deviations from the EPGs	5
3.2.2	Justified Deviations from the EPGs	6
3.2.3	Accuracy of Deviation Document	7
3.3	Simulator Validation of EOPs	8
3.3.1	Simulator Scenario No. 1	8
3.3.2	Simulator Scenario No. 2	9
3.3.3	Simulator Scenario No. 3	9
3.3.4	Validation of EOP Entry Conditions	10
3.4	Walkthrough Validation of EOPs	11
3.4.1	Technical Adequacy of Procedures	11
3.4.2	Availability of Special Tools and Equipment	12
3.4.3	Station Material Condition	13
3.4.4	Containment Venting Pathways	14
3.5	Verification of EOPs	14
3.5.1	Adequacy of PGP Writer's Guide	14
3.5.2	Implementation of PGP Writer's Guide	15
3.5.3	Verification of Hardware-Procedure Interface	16
3.5.4	Verification of Calculations and Setpoints	16
3.6	Operator EOP Training	17
3.6.1	Site Specific Simulator	17
3.6.2	Operator Requalification Training for EOPs	17
3.6.3	Operator Training on EOP Changes	18
4.	MANAGEMENT EXIT MEETING	20
Appendix A - Personnel Contacted		A-1
Appendix B - Documents Reviewed		B-1

1 INSPECTION OBJECTIVES

The inspection team reviewed the licensee's Emergency Operating Procedures (EOPs), operator training and plant systems to accomplish the following objectives in accordance with NRC Temporary Instruction (TI) 2515/92:

(1) Determine whether the EOPs conformed to the vendor generic guidelines and were technically correct for the Vermont Yankee Power Plant.

(2) Assess whether the EOPs can be carried out in the plant under the expected environmental conditions with the most limiting operating crew complement.

(3) Evaluate whether the plant staff has been adequately trained to perform the EOP functions in the time available.

2 BACKGROUND

Following the Three Mile Island (TMI) accident, the Office of Nuclear Reactor Regulation developed the "TMI: Action Plan," (NUREG-0660 and NUREG-0737) which required licensees of operating plants to reanalyze transients and accidents and to upgrade Emergency Operating Procedures (EOPs) (Item I.C.1). The plan also required the NRC staff to develop a long-term plan that integrated and expanded efforts in the writing, reviewing, and monitoring of plant procedures (Item I.C.9). NUREG-0899, "Guidelines for the Preparation of Emergency Operating Procedure," represents the NRC staff's long-term program for upgrading EOPs, and described the use of a Procedures Generation Package (PGP) to prepare EOPs.

The licensees formed four vendor owners' groups corresponding to the four major reactor vendor types in the United States: Westinghouse, General Electric, Babcock & Wilcox, and Combustion Engineering. Working with the vendor company and the NRC, these owners' groups developed generic procedures that set forth the desired accident mitigation strategy. For General Electric plants, the generic guidelines are referred to as Emergency Procedure Guidelines (EPGs) which were to be used by licensees in developing their PGPs. The NRC has issued generic safety evaluation reports (SERs) for approval of Revisions 2 and 3 of the EPGs. Revision 4 of the EPGs is currently under review by the NRC. Generic Letter 82-33, "Supplement 1 to NUREG-0737 - Requirements for Emergency Response Capability" required each licensee to submit to the NRC a PGP which included:

(1) Plant Specific Technical Guidelines (PSTGs) with justification for safety significant differences from the EPG.

(2) A Plant Specific Writer's Guide (PSWG).

(3) A description of the program to be used for the verification and validation of EOPs.

(4) A description of the training program for the upgraded EOPs.

Submittal of the PGP was made a requirement by Confirmatory Order to the Vermont Nuclear Power Corporation dated June 12, 1984. Plant specific EOPs were to have been developed that would provide the operator with directions to mitigate the consequences of a broad range of accidents and multiple equipment failures.

A representative sample of EOPs from each of the four vendor types have been selected for review by NRC teams from Regions I, II, III and IV. Emergency Operating Procedures for 13 additional plants with GE BWR Mark I containments were selected for review by teams from the NRC Office of Nuclear Reactor Regulation (NRR). This inspection at the Vermont Yankee Nuclear Power Corporation was one of the supplemental reviews conducted by NRR.

3 DETAILED INSPECTION FINDINGS

3.1 Emergency Operating Procedure (EOP) Program Evaluation

The inspection team reviewed the licensee's program for upgrading EOPs in accordance with Section 7 of NUREG 0737 (Supplement 1), "Upgrade Emergency Operating Procedures," to determine whether the intent of NUREG requirements had been accomplished and that proper documentation had been submitted to the NRC for review. The program documentation reviewed is listed in Appendix B to this report. The team concluded that the licensee had expended a significant amount of resources to improve the EOP program as evidenced by the four revisions to the Procedure Generation Package (PGP), five revisions to the EOPs, responses to IE Information Notices and supplemental studies performed for containment safety. However, the team was concerned that the licensee had also made significant deviations from the EPGs and that these deviations had not been adequately documented and submitted for NRC review.

3.1.1 Procedure Generation Package (PGP) Review

The licensee has committed to implement Revision 3 of the EPGs, but has never submitted the Plant Specific Technical Guidelines (PSTGs) for NRC review as required by NUREG 0737 (Supplement 1). In response to an NRC letter, dated September 9, 1984, requesting additional information including the PSTGs, the licensee stated that the information comprising the PSTGs was available at the station for NRC review.

The inspection team found no other evidence that the PSTGs had been reviewed and approved by the NRC and concluded that the current PGP submittal did not satisfy the requirements of NUREG 0737 (Supplement 1). The team was concerned about the lack of the PSTG submittal because the licensee had taken significant exceptions to the EPGs as discussed in Section 3.2 of this report, which appeared to place the licensee outside the scope of the generic safety evaluation report issued for Revision 3 of the EPGs.

3.1.2 Licensee Validation and Verification of EOPs

The licensee has issued as many as five revisions to selected EOPs in a continued effort to improve procedure quality. The inspection team reviewed the documentation associated with the validation and verification of these revisions and made the following observations:

(1) The initial verification of Revision 1 of the EOPs was completed on August 31, 1984 by an independent contractor. Procedure OE 3105, "Secondary Containment Control," was not part of this verification package. The verification report identified that several EOP supporting documents were missing from the verification package and prioritized the procedure deficiencies found for resolution either before or after EOP implementation. The licensee dispositioned the findings and revised the EOPs as necessary.

(2) The initial validation of Revision 1 of the EOPs was completed on February 22, 1985 with the assistance of an independent contractor and the Dresden simulator. Procedure OE 3105 was not part of this validation package. All EOP decision steps not validated on the simulator were addressed by a

table top review. The validation was documented in accordance with the PGP and appeared to be comprehensive.

(3) The licensee initially verified and validated procedure OE 3105, Revision 0, before issuance in April 1986 using PGP guidance. The documentation for the verification appeared to be limited to a signature on the procedure cover page. The details of the validation were not included in the EOP package so the team could not determine whether all decision paths had been validated.

(4) The documentation for verification and validation of subsequent EOP revisions did not appear to be adequate as was previously identified by the licensee in the independent audit (see Section 3.1.3).

3.1.3 Response to IE Information Notices

In August 1986, the NRC issued IE Information Notice 86-64, "Deficiencies in Upgrade Programs for Plant Emergency Operating Procedures," to alert licensees to problems found during NRC reviews of PGPs and EOPs. The licensee's internal response to IE Information Notice 86-64 reviewed the existing development program and concluded that the existing PGP and EOPs were satisfactorily implemented. In response to Information Notice 86-64 (Supplement 1), issued on April 20, 1987 to describe further EOP and PGP problems, the licensee performed an independent audit to determine whether problems existed with the Vermont Yankee EOP development program. The audit followed the guidance provided in NRC Temporary Instruction (TI) 2515/79, "Inspection of Emergency Operating Procedures," and revealed the following problems:

(1) Not all deviations from the EPGs were identified in the controlled deviations document.

(2) Calculations performed to support the EOPs were not properly controlled.

(3) The PGP Writers Guide did not address all the issues specified in NUREG 0899.

(4) The PGP Writers Guide was not properly implemented for the EOPs.

(5) There was no mechanism for ensuring EOP revisions were properly verified and validated.

The licensee had developed a corrective action program for the deficiencies identified in the audit with all items scheduled for completion by November 1988. A followup audit was also scheduled after November 1988 to verify that the corrective actions were adequate. The team commended licensee management for conducting the independent audit and concluded that the proposed corrective actions were appropriate for the audit findings. However, a concern was raised about the potential uncertainty of the schedule. The licensee was planning to implement Revision 4 of the EPGs upon NRC approval by a generic safety evaluation report (SER). The licensee believed the issuance of this SER to be imminent and was delaying their next revision to the EOPs to incorporate both the audit findings and the EPG, Revision 4 changes. The team was concerned that the corrective actions for the independent audit should not be delayed if the NRC SER for Revision 4 of the EPGs were not promptly issued.

3.1.4 Supplemental Safety Activities

The licensee has performed two studies which supported their decisions for the primary containment control EOPs. The first was the "Vermont Yankee Containment Safety Study", dated August 1986, which analyzed the expected performance of the Vermont Yankee Nuclear Power Plant containment under postulated event scenarios. This study identified 37 items which would improve safety at the Vermont Yankee plant in the event of station blackout or other severe accident scenarios. During the inspection, the team verified the adequacy of seven of these items which enhanced the EOPs beyond the EPG guidance. Additionally, the licensee had developed "Vermont Yankee Containment Venting Guidelines," which provided a listing of 32 containment venting paths, the advantages and disadvantages of each path and expected availability under the various scenarios. These guidelines were available in the Technical Support Center for licensee management to consult and provide recommendations to the Shift Supervisor for event management. The inspection team concluded that these studies contributed significantly to the safety of the plant.

3.2 Comparison of EOPs to Emergency Procedure Guidelines (EPGs)

The inspection team compared the EOPs with Revision 3 of the EPG and identified deviations from the owners' group guidance. These deviations were compared with the licensee's controlled deviation document to review the adequacy of the justifications. The team concluded that deviations from the EPGs in the EOPs had not always been properly identified and justified. The licensee had previously identified a similar concern during their independent audit and had scheduled corrective action commensurate with the next revision to the EOPs.

3.2.1 Unjustified Deviations from the EPGs

The inspection team identified the following deviations from the EPGs in the EOPs which were not identified in the deviation document:

(1) The licensee decided to reformat the EOPs so that the various EPG decision paths for Reactor Pressure Vessel (RPV) Control were separated into three procedures: OE 3100, "Reactor Scram Control," Revision 4, OE 3101, "Reactivity Control," Revision 4, and OE 3102, "RPV Level Control," Revision 5. These EOPs were independent of each other and the decision paths were not entered simultaneously as described in the RPV Control EPG. This arrangement resulted in a situation where the step to confirm emergency core cooling system (ECCS) initiation, emergency diesel generator starting and containment isolation signals was only implemented for Procedure OE 3102 and not the other EOPs.

(2) Procedure OE 3101, steps LC-3 and LC-4 appeared to direct entry into emergency depressurization in a sequence different from that specified in the RPV Control EPG, Contingency 6, "RPV Flooding." The team was concerned that this resequencing might delay isolation and controlled depressurization of the RPV.

(3) Procedure OE 3101 required continued EOP actions for stuck rods even when reactor power was less than the average power range monitor (APRM) down scale trip setpoint (2%). The reactor power control path of the RPV

Control EPG allowed the operator to exit the EOP to the scram recovery procedure if power was less than the APRM downscale setpoint. Although this deviation appeared to be conservative, the team was concerned that remaining in the EOP unnecessarily may distract the operators from other more significant concerns.

(4) Procedure OE 3105, "Secondary Containment Control," Revision 2, specified an EOP entry condition as a reactor building exhaust radiation level reading of 14mr/hr, the Technical Specification limit. This level appeared contrary to the Secondary Containment Control EPG, which specified an entry condition as the value initiating an isolation of the reactor building ventilation system. The isolation setpoint was 12mr/hr at the Vermont Yankee Nuclear Power Plant. The team was concerned that because the ventilation system isolated before reaching the entry condition radiation level, the EOP would not always be entered when intended.

(5) Procedure OE 3102 did not require isolation of the high pressure core injection (HPCI) and reactor core isolation cooling (RCIC) systems when flooding the RPV with rods remaining out of the core and a feedwater pump available. The team was concerned that flooding the RPV with HPCI and RCIC systems, which feed to the top of the core, could dilute the boron concentration on the top of the core from the standby liquid control system. The feedwater system should be used to allow flow from outside the shroud up into the core through the plenum to allow mixing with the concentrated boric acid in the core.

3.2.2 Justified Deviations From the EPGs

The licensee did not implement several significant actions in the EOPs that were prescribed in Revision 3 to the EPGs. These exceptions were identified in the controlled deviation document, "Vermont Yankee Nuclear Power Corporation Technical Justifications for Deviations, Additions and Alterations from the BWR Owners' Group Technical Guidelines," with the licensee's reasons for deviating from the owners group guidance. Because the licensee had not submitted the PSTGs as part of the PGP, the NRC had not reviewed the following significant deviations and justifications:

(1) There was no EOP for implementing the Radioactivity Release Control EPG. The licensee's justification was that the EPG guidance was an unnecessary duplication of the actions of other EOPs for indications other than high radioactivity release rate. The only event which would cause a high release rate was a high energy line break, which by itself would initiate emergency depressurization either by the size of the break or other EOPs for RPV level, primary containment or secondary containment control.

(2) Procedure OE 3105 allowed the plant to remain operating with a stuck-open safety relief valve (SRV) until torus water temperature approached 110°F. The Torus Temperature Control EPG required scramming the reactor if the SRV could not be closed within two minutes. The justification provided by the licensee was to offer alternate actions to initiate torus cooling and, if the torus water temperature could be maintained below 110°F, conduct an orderly shutdown instead of scramming the reactor. Licensee studies supported the conclusion that torus cooling could support a single stuck-open SRV so that the scram would only be required if there were problems with torus cooling.

(3) Procedure OE 3105 did not require emergency depressurization for high radiation levels in the secondary containment. The licensee's justification for this deviation from the Secondary Containment Control EPG explained that the basis for emergency depressurization would be to reduce the release rate of a primary system discharging into the secondary containment. If this situation were to occur, there would be other indications such as high temperature or levels in the secondary containment which would already lead to emergency depressurization.

(4) Procedure OE 3105 initiated emergency depressurization only after the maximum safe operating temperature was exceeded in two areas, as defined by a limiting condition matrix, instead of a single area as specified in the Secondary Containment Control EPG. The limiting conditions matrix was a tabulation of area pairs affecting both redundant trains of a safety function. The licensee's justification was that emergency depressurization should only occur when the preservation of reactivity control, ECCS initiation and cooling, RPV level and pressure control, decay heat removal, or post-accident monitoring functions were in jeopardy.

(5) The licensee decided not to implement the RPV Control EPG, Contingency 7, "RPV Power/Level Control." The justification cited the concerns raised about the contingency in the NRC generic safety evaluation for Revision 2 of the EPGs and problems with Vermont Yankee operators accepting the concept to lower water level during an event when rods remained out of the core and the core was generating power. Subsequently, the licensee performed further analyses of RPV power/level control actions and gained operator acceptance of the concept. The licensee now planned to implement RPV power/level control actions during the next major revision to the EOPs.

(6) Procedure OE 3100 allowed the mode switch to be placed in either the "shutdown" or "refuel" position while the RPV Control EPG specified only the "shutdown" position. Placing the switch in "shutdown" provided a backup scram signal in the event rods remained out of the core. During the simulator validation, the team observed that the operators used the "refuel" position exclusively to obtain a rod permissive light for rod bottom indication. The team noted that Procedure 3100 did require the initiation of a manual scram signal if all rods were not fully inserted which energized the same scram relay as the "shutdown" position, but this justification was not identified in the deviation document.

3.2.3 Accuracy of Deviation Document

During the comparison of the EOPs with the EPGs, the team became concerned about the accuracy of the controlled deviation document, "Vermont Yankee Nuclear Power Corporation Technical Justification for Deviation, Addition, and Alteration from the BWR Owners' Group Technical Guidelines." There were several instances where identified deviations did not agree with the text of the EOPs. Further review by the team and interviews with licensee personnel responsible for maintenance of the deviation document revealed that changes were incorporated on a chronological basis when the procedures were revised and that a validation of the complete deviation document was not always performed. The licensee had identified this problem during the independent audit and had scheduled a complete review of the deviation document as part of the next major revision to the EOPs.

3.3 Simulator Validation of EOPs

The inspection team validated portions of the EOPs using the licensee's site specific simulator. The licensee provided qualified station and simulator operators to support the validation. Three event scenarios were performed on a crew of four operators and a fourth scenario was conducted with the assistance of the simulator operators to test EOP entry conditions. These scenarios were designed to test the maximum number of EOP decision paths during the available simulator time and were not suitable for testing licensed operator performance. Event sequences were accelerated by the use of malfunctions beyond the design bases of the plant. In two scenarios, limitations in the controlling software caused the simulator to malfunction, ending the scenario before the decision paths were fully tested. During this validation, the team identified deficiencies with the EOPs that could cause operators problems during an actual casualty.

3.3.1 Simulator Scenario No. 1

From starting conditions of 100% reactor power and no equipment out of service, the malfunctions of a turbine trip, loss of turbine bypass function and failure to scram (ATWS) were initiated. This scenario was designed to validate the following EOP flowpaths under dynamic conditions:

- OE 3101 Rod Insertion, Level Control and Boron Injection
- OE 3102 RPV Emergency Depressurization with Rods Out
- OE 3103 Drywell High Pressure Control
- OE 3104 Torus High Water Temperature Control

The inspection team made the following observations during the conduct of the validation and after debriefing plant staff following the simulation:

(1) The rod insertion, level control and boron injection EOP flowpaths were properly executed by the operating crew.

(2) The crew delayed RPV emergency depressurization because they were not aware that the heat capacity temperature limit (HCTL) curve for the torus was exceeded by plant conditions.

(3) An operator initially misread torus water level as being above the nozzles and torus sprays were not promptly initiated to reduce torus and drywell pressure. This mistake was corrected a short time later by another operator and sprays were used to control pressure.

(4) The net positive suction head (NPSH) curve was violated for the residual heat removal (RHR) pumps taking suction on the torus. During the simulation debrief, concerns were raised by the operators about the location of the RHR pump NPSH graph on the flowchart with respect to the sequence of steps being followed during the scenario. The only NPSH graph was located near the torus pressure control decision path, although the temperature control flow path was a significant variable in the curve and the graph was referenced on two occasions in the decision path.

3.3.2 Simulator Scenario No. 2

From starting conditions of 100% reactor power with no equipment out of service, the malfunction of an unisolable major steam leak in the HPCI pump room was initiated. When the reactor was scrammed, five rods failed to insert. This scenario was designed to validate the following EOP flowpaths under dynamic conditions:

- OE 3100 Manual Scram
- OE 3101 Rod Insertion
- OE 3102 RPV Emergency Depressurization
- OE 3104 Torus Water Temperature Control
- OE 3105 Secondary Containment High Temperature Control

During the scenario the simulator malfunctioned causing the feedwater system to operate sporadically, complicating the validation. The inspection team made the following observations during the validation and after debriefing the plant staff following the simulation:

(1) The operating crew properly entered Procedure OE 3105 and manually scrammed the reactor.

(2) An operator failed to bypass the scram discharge volume high level scram input and reset the scram. This inaction prevented the use of the control rod drive system to drive the rods into the core and caused the scenario to deviate from the desired path of depressurization with all rods inserted. The crew's actions for the alternate reactivity control decision path in Procedure OE 3101 were correct.

(3) The crew unnecessarily flooded the reactor up to the safety relief valves instead of controlling level in a prescribed band. Interviews with the operators involved revealed that Procedure OE 3102, Step LC/B-2, was cumbersome to use. The intent of the procedure was to initiate flooding and then control within a prescribed band. Instead, because the action block contained terminology to flood until a contingency was met, the operator continued flooding above the desired level. This procedure structure problem was previously identified during the independent audit and was an example where the licensee had not followed the PGP Writer's Guide.

3.3.3 Simulator Scenario No. 3

From starting condition of 100% reactor power with no equipment out-of-service, the malfunction of an unisolable steam leak in the tunnel was initiated. Additionally, when the operators manually scrammed the reactor, all rods remained out and the RHR isolation valves failed to open when operators attempted to spray the drywell. This scenario was designed to validate the following EOP flowpaths under dynamic conditions:

- OE 3101 Rod Insertion, Boron Injection, Level Control and Pressure Control
- OE 3102 Emergency Depressurization and Level Restoration at Power
- OE 3103 Drywell Pressure Control Containment Venting

- OE 3104 Torus High Temperature Control

The major emphasis of this scenario was to determine whether adequate guidance was provided to make containment venting decisions. The simulator inadvertently aborted the scenario before this objective could be reached, but the team made the following observations during the simulation:

(1) The crew properly implemented the actions in Procedure OE 3101 for reactivity control with stuck rods and failure of the standby liquid control system.

(2) The decision to depressurize the RPV was delayed and the torus HCTL curve was exceeded for a few minutes before emergency depressurization. The actions taken by the crew during depressurization appeared adequate.

(3) The corrective actions for drywell pressure control described in Procedure OE 3101 were not taken until drywell pressure was 17 psig. This delay prevented the use of the standby gas treatment system to reduce drywell pressure and appeared to occur when the Shift Supervisor became overloaded while managing the RPV pressure and level, reactivity, and torus temperature control problems. The Shift Supervisor delegated the conduct of Procedure OE 3103 for drywell pressure control to the Senior Control Room Operator (SCRO). The SCRO verified the entry conditions, but then became distracted and did not take any actions to mitigate the high drywell pressure. The Assistant Control Room Operator (ACRO) subsequently advised the Shift Supervisor that drywell pressure was 17 psig and the corrective actions were initiated. Just as these actions were initiated, the simulator aborted the scenario and the validation was stopped.

The inspection team identified two causes for this observed problem. The first was that the crew was short-handed for the validation; missing an on-shift engineer who monitors plant response to the casualty. The team concluded that the shift engineer could have detected the lack of drywell actions much sooner in the scenario. The second cause was a lack of training on management of complex casualties as a team on the simulator. The licensee had promulgated guidance for assigning responsibilities during EOP performance in an Operations Department Memo dated April 15, 1985. This memo provided general areas of responsibility for control of the casualty, but left specific assignments to the Shift Supervisor. The inspection team concluded that training on complex casualties was weak as discussed in Section 3.6.2 of this report and that the crew performance as a team reflected this lack of team training.

3.3.4 Validation of EOP Entry Conditions

The inspection team used the simulator to validate the availability of the indication referenced by the EOPs under a loss-of-offsite power condition. All instrumentation required for EOP entry was available under these conditions, but indication for one entry condition to Procedure OE 3105 for Secondary Containment Control was not available. The entry condition of a continuously running sump pump was not available since it was supplied from a nonvital bus. There was backup indication of a high room level alarm. The use of a continuous sump pump running as an entry condition was a deviation from the EPGs which identified a high sump level alarm as the entry condition,

but the consequences of a loss-of-offsite power were not discussed in the justification.

3.4 Walkthrough Validation of EOPs

The inspection team conducted walkthroughs of all the appendices to the EOPs and the operating procedures referenced in the EOPs to accomplish activities outside the control room. The specific appendices and procedures validated by the walkdowns are listed in Appendix B of this report. The walkthroughs were conducted with the operators expected to perform the procedures in an emergency and validated the following concerns:

- o adequacy of procedure guidance
- o ability of operators to perform the procedures
- o availability of special tools and equipment in the plant
- o material condition of the systems and equipment being operated by the procedures.

Additionally, the team walked down the various pathways for venting the containment as described in the "Vermont Yankee Containment Venting Guidelines." The inspection team determined that the guidance provided for EOP activities outside the control room was adequate and was impressed with the operators' knowledge of the EOPs, equipment material condition, and plant cleanliness.

3.4.1 Technical Adequacy of Procedures

The overall technical adequacy provided by the appendices and operating procedures referenced in the EOPs appeared to be adequate. With the information in the procedures, plant operators responsible for performing the activities were able to walkthrough the tasks for the inspection team. The team did identify the following deficiencies with the procedures during the walkthroughs:

(1) Procedure OE 3101, Appendix C, "Boron Injection Using Control Rod Drive System from Standby Liquid Control Tank," Revision 4, directed the operator to connect 200 feet of hose from the standby liquid control tank to the control rod drive (CRD) pump inlet, but did not provide guidance for venting the hose. The team was concerned that air in the line could bind or damage the CRD pump. The licensee tested the hoses and concluded that the self-venting features of the hose were inadequate for the intended use and committed to revise Appendix C to include venting guidance.

(2) Procedure OE 3104, Appendix D, "Torus Makeup From Core Spray System," Revision 4, referenced valves CST-2, CS-8A and CS-8B as being locked in position, but during the walkdown the team found these manual valves to be unlocked. The operators stated that the valves had been locked in the past, but when the locks were intentionally removed, the procedure was not updated. The licensee committed to revise Appendix D to reflect the unlocked positioning.

(3) Procedure OE 3101, Step RC/R-29, directed that the CRD hydraulic control unit (HCU) withdraw line be vented, but did not reference a procedure by which this was to be accomplished. The operators interviewed were aware that this was to be accomplished by Procedure OP 2111, "Control Rod Drive System," Revision 16. Procedure OP 2111 did not provide guidance for obtaining keys for access to plant areas where the special tools and fittings for this evolution were kept and did not caution the operator that under EOP conditions, the discharge being vented could be a dual phase mixture of steam and liquid that could cause injury. During the inspection, the licensee committed to revise the procedure to provide better guidance for the HCU venting evolution.

(4) Procedure OE 3101, Appendix D, "Boron Injection Using Reactor Water Cleanup System" Revision 4, required coordinated action between the control room and the reactor water cleanup system local control panel in the reactor building. The delegation of responsibilities between the two locations was not delineated in the procedure and in one instance an operator incorrectly stated that a specific valve control operation would be performed from the control room when it should have been performed from the local control panel. The licensee committed to revise the procedure to clarify the responsibility for performing the actions between the two locations.

(5) Procedure OE 3105, Caution 24, stated that isolation interlocks may need to be bypassed under certain plant conditions in order to operate the reactor building ventilation system in accordance with Step SC/T-5. Neither the step nor the caution statement referenced Appendix A, "Bypassing Reactor Building HVAC Trips," Revision 2, which described how to defeat the interlocks.

(6) Procedure OE 3102, Step LC/D-13, referenced Caution 22 which stated, "Defeating isolation interlocks may be required to accomplish this step." The step directed rapid depressurization of the RPV using the turbine bypass, mainsteam line drain, HPCI, RCIC and RPV head vent systems. No EOP appendix existed to direct how the interlocks were to be defeated.

3.4.2 Availability of Special Tools and Equipment

The availability of tools and equipment in the plant appeared to be adequate to accomplish the activities required by the appendices and procedures referenced in the EOPs. The licensee had prestaged special EOP toolboxes in the various areas of the plant and supporting equipment and chemicals were designated for EOP use only and set aside from other equipment and supplies. The inspection team inventoried the EOP tool boxes and identified shortages of electrical tape, electrical jumpers and knives in some of the tool boxes. In all cases the operators could identify where these replacement supplies and alternate tools were available, but the team was concerned that this shortage would cause unnecessary delays in accomplishing the procedure task. The licensee agreed with the team and restocked the EOP tool boxes with the identified shortages during the inspection.

3.4.3 Station Material Condition

The inspection team reviewed the material condition of the station during the plant walkdowns to ensure that necessary equipment and components were accessible and functional. The overall material condition of the plant appeared to be excellent and all material deficiencies identified could be compensated for by the operators. The team made the following observations:

(1) Plant cleanliness was exceptional. During plant operations, the torus areas and ECCS rooms were accessible to personnel without the use of anti-contamination clothing. Also, the team did not observe any interference in the reactor building due to scaffolding or maintenance activities. The team concluded that plant cleanliness was a strength when considering the ease of personnel movement for accomplishing EOP activities outside the control room.

(2) During the plant tours it was observed that there was no battery operated emergency lighting in the torus catwalk area and very limited lighting in the area underneath the torus. There were several locations in these areas where adequate lighting was needed for both personnel safety reasons and equipment operation following certain accidents. The licensee indicated that the battery operated lights were placed in locations where equipment operation would be required following a station blackout or following a fire which necessitated shutdown from outside the control room. Otherwise, lighting was provided in these areas from vital busses. The operators assisting the team in the walkdowns indicated a concern for adequate lighting and had compensated for it by carrying flashlights when in the areas.

(3) The team was impressed that all major valves and components were identified or tagged in some manner. However, the team observed that minor tagging deficiencies existed, such as use of temporary labels on the RWCU local panel in the Reactor Building, and the use of magic markers for identifying valves. None of these instances hampered the operator's knowledge of the valves and their function, indicating their capability to compensate for the lack of ideal tags. The licensee informed the team that a program existed to replace damaged or missing tags whenever valve lineups were performed. This program should correct the deficiencies observed.

(4) The team noted that noise levels in the below grade areas of the Reactor Building were significant, and would be very high following an accident when emergency equipment (HPCI, RHR) would be operating. The only communications equipment available was the Gaitronics public address system and, in some locations, sound powered phones. Plant operators indicated that they anticipated communications problems in the post-accident environment, and also indicated that portable radio communications might not be possible in these areas. The inspection team concluded that the operators could adequately compensate for the anticipated problems, but that doing so would further complicate accident management.

(5) Procedure OE 3103, Appendix A, "Primary Containment Spray Using the Fire System to RHR System," Revision 5, directed operation of RHR system manual cross-connect valve, RHR-20. Operators interviewed expressed concerns as

to whether this valve would operate since it had not been recently cycled. The licensee committed to look into the operating history of valve RHR-20 and take appropriate actions to ensure it properly operated. The operators identified an alternate cross-connect path through valves RHR-56 and RHR-75 which would bypass RHR-20 if problems were to occur.

3.4.4 Containment Venting Pathways

The inspection team performed walkdowns of the various pathways outlined in the "Vermont Yankee Containment Venting Guidelines" to assess the material condition and accessibility of components considered by the guidelines. No deficiencies were noted with the material condition of any of the components identified in the venting lineups. Additionally, the operators accompanying the team on the walkdown were quite familiar with the potential use of the paths even-though those decisions were intended to be made by the Technical Support Center personnel and Shift Supervisors.

3.5 Verification of EOPs

The inspection team performed an independent verification of the PGP Writer's Guide development and implementation, EOP-hardware interface and EOP calculations to determine whether the licensee had properly accomplished the verification process. Based on the findings, the team concluded that the verification of the current revision of the EOPs was not adequate. The licensee had previously identified similar deficiencies in their independent audit and had scheduled corrective actions to resolve the verification issues before issuance of the next EOP revision.

3.5.1 Adequacy of PGP Writer's Guide

The inspection team reviewed the licensee's EOP Writer's Guide as described in Section 4.0 of the PGP (Revision 01) to determine whether the methods and guidance of NUREG-0899 had been properly incorporated. The team concluded that the licensee had not properly incorporated the NUREG-0899 requirements for the following areas:

AREA	NUREG-0899 SECTION
Formulas and Calculations	5.6.9
Verification Steps	5.7.2
WARNING, CAUTION, and NOTE Statements	5.5.3
Placekeeping Aids	5.5.4
Sequencing	5.7.1
Consistency Between Staffing and Procedures	5.8.1
Division of Responsibility	5.8.2
Correcting Discrepancies	3.3.5.2

The licensee was aware that their writer's guide was not in accordance with NUREG-0899 based on a finding identified during the independent audit of the EOP program. Corrective action was scheduled to revise the PGP Writer's Guide to be in agreement with NUREG-0899 in July 1988.

3.5.2 Implementation of PGP Writer's Guide

The inspection team performed an independent verification of EOP flowcharts and appendices to determine whether the writer's guide was properly implemented. The following deficiencies were noted:

(1) Procedure OE 3103, Graph DW/P-2, did not conform with the guidance of the PGP Writer's Guide, Section 2.10 "Component Identification." The PGP Writer's Guide directed that values used in the flow charts should conform to plant instrument readings. The graph was labeled "Containment Water Level (ft)" on the x-axis, while the control room instrument was identified as "Torus Water Level."

(2) Procedure OE 3103, Step DW/T-14 did not conform with the guidance of the PGP Writer's Guide, Section 2.2, "Complexity." The PGP Writer's Guide directed that the number of actions called out in an action block be limited to one. Step DW/T-14 appeared to provide direction for several actions concerning spraying the drywell and directions for actions to be taken when drywell sprays were no longer required.

(3) There was no abbreviation list associated with the EOPs reviewed as directed by the PGP Writer's Guide, Section 2.4.b, "Abbreviations."

(4) The Appendices did not conform with the format and style outlined in the PGP Writer's Guide, Section 3.0, "Appendices Guidelines."

(5) Procedure OE 3012, Step LC-11, directed RPV emergency depressurization in accordance with the steps provided on another sheet of the EOP. Step LC-12 was located immediately below step LC-11 and directed that injection from sources outside the primary containment be secured, except for boron injection and CRD, when RPV pressure is below 280 psig. The team concluded that placement of step LC-12 increased the potential for this integral step to be overlooked and was an example of improper step sequencing.

(6) Administrative control for the job performance aids posted in the control room appeared to be weak. A master list of the aids was kept in the "Operator Aid Status Book," but there was no control over the aids posted in the control room for operator use.

(7) The temperature in the drywell was referred to inconsistently in Procedure OE 3013 as Drywell Atmospheric Temperature (entry condition), Drywell Temperature (Step DW/T-2), and Drywell Average Air Temperature (Step DW/T-9).

The team concluded, based on the above findings and the format deficiencies identified with other procedures in the validation process, that the licensee had not properly verified the EOPs in accordance with the PGP Writer's Guide. The licensee was already aware of this general concern from the independent audit conducted of their EOPs. The corrective actions for these findings was scheduled for implementation on the next revision in August 1988.

3.5.3 Verification of Hardware - Procedure Interface

The licensee accomplished the verification that correspondence existed between the EOPs and plant hardware as part of their detailed control room design review (DCRDR). The inspection team performed an independent review of the EOPs and the control room and identified several human engineering deficiencies. In each case, however, the team's deficiencies were already identified in the licensee's DCRDR document and a corrective action was scheduled to resolve each deficiency. The team concluded that the licensee's hardware - procedure interface verification was adequate.

3.5.4 Verification of Calculation and Setpoints

The team selected seven calculations and bases for figures and setpoints used in the EOPs. These calculations are identified in Appendix B to this inspection report. The following deficiencies were noted:

(1) The torus volume used for Torus Water Level Indication Curve T/L-1 was inconsistent with the value used in the calculation of Pressure Suppression Pressure Curve DW/P-2. The results of this difference were minor and did not affect the curves, but the licensee could not confirm the correct value.

(2) The calculation procedure used for determining the Pressure Suppression Pressure Curve DW/P-2 was from Revision 2 (draft) of the EPGs. The licensee could not confirm whether this calculation was correct and applicable to the Revision 3 EPGs they had implemented.

(3) The calculation package for the Pressure Suppression Pressure Curve DW/P-2 had several annotations in the margins indicating that the various input values to the calculation were incorrect. The licensee could not confirm what the correct inputs should be or whether the errors were safety significant.

(4) The calculation package for the Drywell Spray Initiation Pressure Limit Curve DW/P-1 could not be reconciled with the design inputs. The total mass flow rate to the drywell and wet well calculated in the package could not be reproduced using the licensee's input data. In another area of the calculation package, the total mass flow rate to the wet well was identified on the input data sheet as 350 gpm, but the value used for computer input was 332.5 gpm. For both of these problems, the resulting error to the curves appeared to be minor, but indicated a lack of proper control over calculations.

The team concluded that, although none of the deficiencies were safety significant, the inconsistencies were indicative of a poor program for independent verification of calculations. The licensee had already identified this same problem during the independent audit of the EOP program. Procedure AP 0017, "Calculations and Analyses," Revision 0, was issued on June 6, 1988 to provide control for calculations, and verification of all EOP calculations and bases was to be completed in September 1988 as part of the audit corrective actions.

3.6 Operator EOP Training

The inspection team reviewed the use of the simulator for EOP training, adequacy of requalification training and methods for training on EOP changes. The licensee had conducted a significant amount of training using the Dresden simulator and classroom sessions during the initial implementation of the EOPs in 1985. Since then the requalification training program has incorporated EOP review as a recurring topic.

3.6.1 Site Specific Simulator

The inspection reviewed the use of the site specific simulator for training and validation of EOPs. The Vermont Yankee site specific simulator had a minimum number of outstanding modifications and adequately reflected the design of the plant. The software support package for the simulator computer was not designed for EOP training and would not provide indication for significant accident conditions in the plant. During the three scenario runs by the inspection team the simulator twice malfunctioned and aborted the validation prematurely. Despite this shortcoming, the team concluded that the simulator appeared to be a significant asset for EOP training, offering a dynamic environment for operators to work through the EOPs and make decisions on a real-time basis.

3.6.2 Operator Requalification Training for EOPs

The inspection team reviewed the requalification training conducted since April 1986. A list of the documents reviewed is provided in Appendix B to this report. The requalification training sessions consisted of a classroom presentation of the EOP to be demonstrated, a step-by-step walkdown of the EOP and a real-time scenario run on the simulator. There have been 16 EOP scenarios performed by each crew since April 1986 which involved the following casualties:

- Anticipated Transit Without Scram (ATWS)
- Stuck-Open Relief Valve
- Steam Leak in the Drywell
- Loss of Feedwater
- Steam Leak in the Secondary Containment
- Loss of High Pressure Emergency Core Cooling System (ECCS)

Occasionally, two of the above casualties would be combined to test multiple EOPs at the same time, but most of the time the simulated casualty scenarios were not complicated and the operators were able to exit the EOPs after one or two decisions. The team could not find any scenarios where a low pressure ECCS was lost. The inspection team traced through the 16 scenarios and identified the following EOP steps which were not covered by requalification training:

(1) Procedure OE 3102, Steps LC-10 through LC-18, covering RPV level control-steam cooling activities.

(2) Procedure OE 3102, Steps LC/F-8 through LC/F-16 covering RPV level control with no level indication and all rods fully inserted.

(3) Procedure OE 3103, Steps DW/T-9 through DW/T-13, concerning drywell temperature control when temperature is above 260°F.

(4) Procedure OE 3103, Steps DW/P-11 through DW/P-22 concerning drywell pressure control below the primary containment pressure limit.

(5) Procedure OE 3104, Steps T/L-4 through T/L-14 concerning torus low water level control.

(6) Procedure OE 3104, Steps T/L-16 through T/L-30 concerning torus water level above the torus load level limit curve.

(7) Procedure OE 3104 Steps T/T-10 through T/T-15 concerning torus water temperature above the torus heat capacity limits.

(8) Procedure OE 3105 Steps SC/R-3 through SC/R-6 concerning secondary containment high radiation level control.

(9) Procedure OE 3105 Steps SC/L-3 through SC/L-3 through SL/L-10 concerning secondary containment high water level control.

The inspection team concluded that operator training on the more complicated EOP scenarios identified above could be enhanced significantly by further use of the site specific simulator. The steps identified in Items (4) and (7) above, were exercised during the validation part of this inspection (Section 3.3) and the team noted that the operators were not as familiar with the actions taken under these conditions as during the initial steps of EOP performance. The licensee stated that their intent was to concentrate simulator training sessions on scenarios that were the most realistic. The more remote scenarios were covered by classroom training and would be factored into future requalification training based on licensee management assessment of operator training needs. The licensee has scheduled a thorough period of EOP training in the future before a major revision to the EOPs is implemented and agreed to factor the findings of this inspection into future training plans. The inspection team agreed with the licensee's proposed approach to future EOP training.

3.6.3 Operator Training on EOP Changes

Since the initial training on Revision 1 of the EOPs in 1985, the licensee has made several revisions to the original EOPs and issued Procedure OE 3105 for secondary containment control. The inspection team reviewed the training conducted for these changes and made the following observations:

(1) Simulator training for Procedure OE 3105 was conducted in early 1986. The team could not determine the extent of the training, but simulator scenarios were used for at least some of the decision paths and walkthroughs of the remaining paths occurred. The team considered this adequate training.

(2) The majority of the EOP revisions were classified as minor changes not requiring a dedicated training session for operators. For these minor changes, the training consisted of a written summary of the change provided in an operations department memorandum that was required reading for all crews before the EOP change was implemented. The team reviewed the nature of selected changes and agreed that the operations memorandum review was adequate. The team also reviewed the completed sign-off sheets for the training memoranda and confirmed that all operators had reviewed the memo before procedure change implementation.

(3) There appeared to be some problems with the training conducted for implementation of Revision 3 to Procedure OE 3103 for drywell temperature and pressure control. The revision incorporated steps for an alternate method of spraying the drywell in two situations and the torus in one situation using the fire system and implemented Appendix A to OE 3103 to describe how to connect the fire system to the containment spray system. The licensee could not find the documentation for the training on Appendix A to Procedure OE 3103, but operators interviewed remembered the lectures and plant walkthroughs of the procedure. Additionally, the operations memorandum that described the change incorrectly stated that new steps were added to spray the drywell using the fire system in three situations within the EOP, when actually the procedure was revised to spray the drywell in two situations and the torus in one situation.

(4) The operations memorandum that described the minor change made by Revision 2 to Procedure OE 3101 appeared to be incorrect. The memo included a change to delete the words "if not shutdown before" to Step RC/B-2 which directed the initiation of the standby liquid control system. However, the procedure was not changed and the contingency clause still existed in Revision 4 of the EOP.

4. MANAGEMENT EXIT MEETING

An exit meeting was conducted on June 10, 1988 at the Vermont Yankee Nuclear Power Plant. The licensee representatives at the exit meeting are identified in Appendix A of this report. Mr. C. J. Haughney, Chief, Special Inspection Branch, NRR, represented NRC Management at this meeting. The scope of the inspection was discussed and the team leader presented the findings and answered licensee questions. The licensee was informed that some observations could become potential enforcement findings and these items would be followed up by NRC Region I.

APPENDIX A

PERSONNEL CONTACTED

S. Aprea	Operations
L. Cantrell	Operations
A. Chesley	Training Simulator Supervisor
L. Doane	Operations
R. Faupel	Operations
*R. Grippardi	QA Supervisor
*J. Herron	Technical Program Manager
M. Horn	Operations
*G. Johnson	Operations Supervisor
M. Kriden	Training
*D. LeBarge	Operations
G. LeClaire	Assistant Operations Supervisor
I. Marsden	Technical Support Supervisor
B. Metcalf	Operations
*R. Pagodin	Technical Services Superintendent
*M. Palionis	Operations
W. Paul	Operations
*J. Peletier	Plant Manager
W. Pittman	Operations
R. Slater	Operations
R. Slaunwhite	Training
*R. Sojka	ERFIS Project Manager
*R. Spinney	Training Manager
D. Tuttle	Training
*M. Varno	Project Engineer
C. Wamser	Operations
*R. Wanzek	Operations Superintendent

*Attended the Exit Meeting on June 10, 1988

APPENDIX B

DOCUMENTS REVIEWED

1. Emergency Operating Procedures (EOPs), Appendices and Operating Procedures

° OE 3100	Scram Procedure	Revision 4
Appendix A	Scram Conditions	
Appendix B	Primary Containment Isolation Groups	
° OE 3101	Reactivity Control Procedure	Revision 4
Appendix A	Deenergization of Scram Solenoids	
Appendix B	Individual Control Rod Scrams	
Appendix C	Boron Injection Using CRD System From SLC Tank	
Appendix D	Boron Injection Using RWCU	
Appendix E	Manual Isolation and Venting of the Scram Air Header	
Appendix G	Bypassing of Group I Isolation Signals	
Appendix H	Local Firing of Squibb Valve	
° OE 3102	Reactor Pressure Vessel (RPV) Level Control Procedure	Revision 5
Appendix A	Alternate Injection Using RHRSW System	
Appendix B	Alternate Injection Using Fire System to RHR	
Appendix C	Alternate Injection From Condensate Transfer System	
° OE 3103	Drywell Temperature and Pressure Control Procedure	Revision 5
Appendix A	Primary Containment Spray Using Fire System to RHR	
° OE 3104	Torus Temperature and Pressure Control Procedure	Revision 4
Appendix A	Torus Makeup from HPCI	
Appendix B	Torus Makeup from RCIC	
Appendix C	Torus Makeup from RHR System	
Appendix D	Torus Makeup from Core Spray System	
Appendix E	Torus Makeup from RHRSW System	
Appendix F	Torus Level Reduction Using HPCI	
Appendix G	Torus Level Reduction Using RCIC	
° OE 3105	Secondary Containment Control Procedure	Revision 2
Appendix A	Bypassing RX Bldg HVAC Trips	

° OP 2125	Containment Atmosphere Dilution System	Revision 11
° OP 2111	Control Rod Drive System	Revision 16

2. Detailed Control Room Design Review Documentation

   ° Detailed Control Room Design Report - Summary Support Supplement (w/enclosures) dated June 25, 1987.

   ° Vermont Yankee Detailed Control Room Design Review, Final Report (Volumes I and II).

   ° Vermont Yankee DCRDR Expanded Task Analysis (OPVY 283/36).

   ° Vermont Yankee Findings File Report dated March 7, 1986.

3. EOP Validation and Verification Documentation

   ° Operating Emergency Procedures Simulator Validation Project - Methodology and Results (GP-R-212134), dated February 22, 1985.

   ° Review of Operating Emergency Procedures for Vermont Yankee Nuclear Power Station, dated August 1984.

   ° Validation of EOP Appendices (Memo) dated March 15, 1985.

   ° Validation Report for OE 3105, Revision 0, April 1986.

4. Training Documentation

° LOR - OE 3105	Secondary Containment Control
° LOR - OEPRT	Integrated Operational Emergency Procedure Training (Simulator)
° LOR - 2.3	Operational Emergency Procedure Refresher Training (Simulator) for Reactor Operator Licensed Personnel
° LOR - 3.3	LOR Training Cycle Three Simulator
° LOR - 4.4	Emergency Operating Procedure Refresher
° LOR - 87.2	Automatic Depressurization System
° LOR - TDM 87.234	On-Shift Training, Outage Mods
° LOT - 09-001	Reactor Scram Response
° LOT - 09-008	Drywell Pressure and Temperature Control
° LOT - 09-009	RPV Level Control
° LOT - 09-010	Torus Temperature and Level Control
° LOT - 09-011	Reactivity Control
° LOT - 09-012	Secondary Containment Control
° LOR - 3.3/2.3/1.4a	Annual/Bi-Annual Reactivity Manipulations
° LOR - 2.36/1.46	NRC Exam Style Scenario

5. Calculations and Setpoint Bases Documentation

   ° Maximum Safe Operating Value for Area Water Levels (OE 3105)
   ° Torus Water Level Indication Correlation, Curve T/L-1 (OE 3104)
   ° Pressure Suppression Limit Curve DW/P-2 (OE 3103)

- Sizing Calculation for Hose Connecting SLC Tank to CRD Pump Suction (OE 3101 Appendix C)
- Drywell Spray Initiation Pressure Limit Curve DW/P-1 (OE 3103)
- Instrument Inaccuracies in Control Room Indications (Caution 31)
- Bases for 280°F Drywell Temperature as Action Level (OE 3103, Step DW/T-9)
- Bases for 14mr/hr Reactor Building Ventilation Exhaust High Radiation Level Entry Condition to Secondary Containment Control (OE 3105)
- Procedure AP 0017, "Calculations and Analysis," Revision 0

6. Miscellaneous Documentation

- Vermont Yankee Containment Venting Guidelines
- Audit Report of Emergency Operating Procedures for Vermont Yankee Nuclear Power Corporation, dated December 10, 1987 (w/response and commitments)
- Vermont Yankee Nuclear Power Corporation Procedure Generation Package (Revision 1), FVY 87-106, dated November 17, 1987
- Vermont Yankee Internal Responses to IE Information Notice 86-64 (Supplement 1)
- Vermont Yankee Nuclear Power Corporation Technical Justifications for Deviations, Additions, Alterations from BWR Owners' Group Technical Guidelines, Revision 1

Distribution: (w/encl)
Docket File 50-271
DRIS R/F
RSIB R/F
PDR
LPDR
NSIC
TMurley, NRR
JSniezek, NRR
BGrimes, NRR
CRoe, NRR
JZwolinski, NRR
CJHaughney, NRR
JEKonklin, NRR
JEDyer, NRR
BBoger, NRR
RWessman, NRR
VRooney, NRR
JCraig, NRR
WRegan, NRR
JPersinski, NRR
WKane, NRR
JWiggins, RI
DHaverkamp, RI
DFlorek, RI
GGrant, SRI Vermont Yankee
JMcDonald, RI Vermont Yankee
MPerkins, RI (DRMA)
RBores, RI
ACRS (3)
OGC (3)
1S Distribution
Regional Administrators
Regional Division Directors
Team Members

DF01
1/1

OFC	RSIB:DRIS:NRR	RSIB:DRIS:NRR	RSIB:DRIS:NRR	AD:DRIS:NRR	DRP I/II:NRR
NAME	JEDyer:jj	LNorrholm	CJHaughney	BEGrimes for	SVarga
DATE	07/29/88	07/29/88	07/29/88	07/29/88	08/10/88

mkn