U.S. NUCLEAR REGULATORY COMMISSION
REGION I

REQUALIFICATION PROGRAM EVALUATION

REPORT NO.:	91-02 (OL)
FACILITY DOCKET NO.:	50-271
FACILITY LICENSE NO.:	DPR-28
LICENSEE:	Vermont Yankee Nuclear Power Corporation RD 5, Box 169 Ferry Road Brattleboro, Vermont 05301
FACILITY:	Vermont Yankee Nuclear Power Station
EXAMINATION DATES:	February 25 - 28, March 9, 18 - 19, 1991
NRC EXAMINERS:	D. Florek, Senior Operations Engineer C. Tyner, INEL T. Morgan, INEL B. Wetzel, OLB, NRR

CHIEF EXAMINER: RJ Conte for
T. Walker, Senior Operations Engineer — 4/12/91 Date

REVIEWED BY: Richard J Conte
Richard J. Conte, Chief, BWR Section
Operations Branch, DRS — 4/12/91 Date

APPROVED BY: for CW Hehl
Lee H. Bettenhausen, Chief
Operations Branch, DRS — 4/19/91 Date

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PDR ADOCK 05000271
V PDR

EXECUTIVE SUMMARY

Written and operating examinations were administered to four Reactor Operators (ROs) and eight Senior Reactor Operators (SROs). These operators were divided into three crews, two operating crews and one staff crew. The examinations were graded concurrently by the NRC and the facility training staff. As graded by the facility and the NRC, all four ROs and six of the eight SROs examined passed all portions of the examination. Two SROs did not perform satisfactorily on the dynamic simulator evaluation as graded by the NRC and the facility. Both SROs passed the remaining two portions of the examination. One of the three crews that were evaluated performed satisfactorily on the simulator portion of the examination as graded by the NRC and the facility. One of the three crews performed unsatisfactorily on the simulator portion of the examination as graded by the NRC and the facility. The remaining crew was evaluated as unsatisfactory by the NRC and satisfactory by the facility on the simulator portion of the examination.

The licensee's licensed operator training program is considered to be unsatisfactory based on the criteria established in section ES-601 of NUREG-1021, Revision 6 (Unresolved Item 271/91-02-01 - Section 3.1). More than one third of the crews demonstrated unsatisfactory performance on the dynamic simulator portion of the examination as evaluated by the NRC. All of the crews demonstrated weaknesses in the competency of communications and crew interactions. This resulted in failure of two of the crews. The most significant weakness was related to the duties and responsibilities of the crew members, specifically the Shift Supervisor (SS), the Senior Control Room Operator (SCRO), and the Shift Engineer (SE) during abnormal and emergency events.

Other programmatic weaknesses were identified with the rotation practices for ROs during dynamic simulator evaluations, the evaluation techniques used by the facility during dynamic simulator evaluations, and the quality of the examination materials developed by the licensee.

The licensee provided their basis for continued operation in a letter dated March 8, 1991. The NRC acceptance of the basis for continued operation and actions to be taken as a result of the unsatisfactory requalification program determination was the subject of Confirmatory Action Letter (CAL) I-91-007, dated March 11, 1991. The NRC also conducted operational evaluations of three operating crews to assess the licensee's immediate corrective actions and the acceptability of the basis for continued operation. All three crews that were evaluated demonstrated satisfactory performance. These results indicated that the immediate corrective actions of redefining the roles and responsibilities of the crew members and providing teamwork and communications training were effective.

A non-cited violation of 10 CFR 50.74 and 55.25 for failure to notify the NRC of a condition which required a conditional license for a licensed operator was identified. The apparent cause of the failure to notify the NRC was a lack of responsibility for medical notifications within the licensee's organization. The violation is not being cited because it was identified by the licensee and has minimal safety significance (271/91-02-02 - Section 4.0).

The Vermont Yankee Plant Specific Technical Guidelines (PSTG) contains many additions, deletions, and deviations from revision 4 of the BWR Owners Group Emergency Procedure Guidelines (EPGs). These differences do not appear to be adequately justified. Many of the justifications appear to use event based reasons to justify departure from the accident mitigation strategy of the EPGs for symptom based emergency operating procedures. It is not apparent that Vermont Yankee has unique features and/or analysis to support the many additions, deletions and deviations in the PSTG. This is considered to be an unresolved item (271/91-02-03 - Section 5.0).

DETAILS

1.0 Introduction

The NRC administered requalification examinations to 12 licensed operators (4 ROs and 8 SROs). Two operating crews and one staff crew were evaluated. The examiners used the process and criteria described in NUREG 1021, "Operator Licensing Examiner Standards," Rev. 6. The Job Performance Measure (JPM) portion of the examination was administered using the Alternative B methodology described in ES-603, "Requalification Walk-through Examination," of NUREG 1021.

An entrance meeting was held with the licensee on February 11, 1991, at the beginning of the examination preparation week. The personnel contacted during the examination and operational evaluations are listed in Attachment 1. The members of the combined NRC/facility examination team and the facility evaluators are also identified in Attachment 1.

2.0 Requalification Examination Results

2.1 Individual Examination Results

The following is a summary of the individual examination results:

NRC Grading	RO Pass/Fail	SRO Pass/Fail	TOTAL Pass/Fail
Written	4 / 0	8 / 0	12 / 0
Simulator	4 / 0	6 / 2	10 / 2
Walk-through	4 / 0	8 / 0	12 / 0
Overall	4 / 0	6 / 2	10 / 2

Facility Grading	RO Pass/Fail	SRO Pass/Fail	TOTAL Pass/Fail
Written	4 / 0	8 / 0	12 / 0
Simulator	4 / 0	6 / 2	10 / 2
Walk-through	4 / 0	8 / 0	12 / 0
Overall	4 / 0	6 / 2	10 / 2

2.2 Generic Weaknesses

The following is a summary of generic weaknesses noted from the results of the individual requalification examinations. This information is being provided to aid the licensee in upgrading the requalification training program. No licensee response is required beyond that identified in Confirmatory Action Letter (CAL) I-91-007.

Written Examinations

- Ability to determine the actions required to vent the torus via the Standby Gas Treatment System with an isolation signal present.

- Ability to determine the cause of a Reactor Water Cleanup (RWCU) Pump trip.

Walk-through Examinations

- Ability to terminate and prevent injection from the High Pressure Cooling Injection (HPCI) system.

- Knowledge of signals that cause a Group III isolation.

- Failure to thoroughly review the prerequisites, precautions, and limitations of the procedure prior to performing a task.

Dynamic Simulator Examinations

- Ability to exchange complete and relevant information in a clear, accurate, and attentive manner.

- Direction and coordination of activities among crew members.

- Ability of the crew to maintain oversight of plant conditions and make team decisions in an effective manner.

3.0 Requalification Program Evaluation Results

3.1 Examiner Standards Evaluation Criteria

The facility program for licensed operator requalification training is considered UNSATISFACTORY in accordance with the criteria established in ES-601, paragraphs C.2.b.(1)(a-c) and C.2.b.(2)(a-f). The unsatisfactory requalification program is considered an unresolved item (271/91-02-01).

The facility grading was as conservative as the NRC grading on 100% of the pass/fail decision for individuals satisfying the criterion of C.2.b.(1)(a). However, the facility's basis for failing the two individuals that failed the dynamic simulator portion of the

examination was based on competency failures, rather than failure of Individual Simulator Critical Tasks (ISCTs). The NRC's basis for failure of the two individuals was failure of ISCTs.

Eighty-three and three-tenths percent (83.3%) of the operators passed the examination satisfying the criterion of C.2.b.(1)(b).

Two of the three crews evaluated were determined to be unsatisfactory on the dynamic simulator portion of the examination as evaluated by the NRC; therefore, the criterion of C.2.b.(1)(c) was not satisfied resulting in an unsatisfactory program determination.

The facility evaluators did not concur with the NRC on one of the unsatisfactory crew evaluations, therefore criterion C.2.b.(2)(a) is applicable.

The facility trains operators in all positions permitted by their licenses, but does not ensure that all ROs are evaluated in all positions during the dynamic simulator portion of the examination. The facility policy is to allow the ROs to fill the position that they normally fill in the crew during the simulator portion of the examination. Rotation into the other RO position is only required if needed to expose the operator to an ISCT. During the NRC administered requalification examinations, all ROs filled their normal positions during the simulator examinations and the licensee verified that the knowledges and abilities required to fill the other position were evaluated in other portions of the examination. This verification was not done until the rotation practice was questioned by the NRC examiners. For the purposes of the program evaluation, criterion C.2.b.(2)(b) was not considered applicable, but the rotation practices for ROs in the simulator is considered a program weakness.

No facility evaluators were determined to be unsatisfactory; therefore, criterion C.2.b.(2)(c) is not applicable. However, weaknesses were identified in the facility's evaluation techniques for the dynamic simulator portion of the examinations. Because of the limited sample of observed deficiencies, no individual evaluators were determined to be unsatisfactory, but the quality of the licensee's evaluation on the simulator portion of the examination is considered to be a program weakness.

The facility has administrative controls to preclude an RO or SRO who does not possess an active license from performing licensed duties without satisfying the requirements of 10 CFR 55.53 to restore the license to active status. There were no indications of deficiencies in this area; therefore, criterion C.2.b.(2)(d) is not applicable.

The examination materials for all portions of the examinations required modifications prior to administration to meet the NUREG-1021 standards. The problems with the materials did not appear to be caused by a lack of quality control and no changes were required after the examinations were administered; therefore, criterion C.2.b.(2)(e) was not considered applicable. However, the quality of the examination materials is considered a program weakness.

The facility's failure rate for individuals was identical to the NRC's failure rate; therefore, criterion C.2.b.(2)(f) is not applicable.

3.2 Requalification Examination Sample Plan

The sample plan submitted by the facility did not meet the standards of NUREG 1021 for use in preparation of the examinations. The sample plan did not identify the subjects to be evaluated, the preferred testing media for evaluating each subject area, or learning objectives intended to be evaluated. Proposed examinations were submitted that had been developed to meet the sample plan, but the test items in the proposed exams were not cross referenced to the sample plan. The missing information was provided and a cross referenced sample plan (test outline) was generated by the NRC. Additionally, no information was provided that identified other test items that evaluate the subject areas. The licensee training department has the capability to identify test items by subject area within a computerized system, but the information could not be easily provided to the NRC.

Modifications were made to the examinations proposed by the facility to more effectively evaluate the areas that were covered during the requalification cycle. The test outline for the examinations was developed based on the amount of time spent on each lesson plan, rather than the time spent on each subject area. Modifications were made to cover the subject areas more evenly. For example, a JPM for performing torus makeup from Core Spray was replaced with a JPM to perform torus makeup from HPCI. Both JPMs evaluated EOP support procedures, but HPCI was covered during the requalification cycle, while Core Spray was not. Modifications were also made to cover areas more appropriately. For example, a scenario was modified to evaluate Secondary Containment Control in the dynamic simulator portion of the examination.

No test items in the facility examination bank were designed to evaluate the operators' understanding of plant modifications, industry events, or other operating experiences. The sample plan indicated that these subjects constituted more than fifteen percent of the time spent in requalification training.

3.3 Written Examination Preparation and Administration

The examinations that were originally proposed by the facility did not contain enough questions to meet the guidance of NUREG 1021 for number of points on the examination. The licensee submitted revised examinations that met the minimum requirements of the Examiner Standards, but the NRC was still concerned that the time validation of the examinations was not accurate and that the examinations were too short. The time validation of the questions was discussed during the preparation week and the examination team decided to add several questions to the SRO examinations. The average time for the operators to complete and review the classroom examinations and static scenario examinations were 97 minutes and 102 minutes respectively. The Examiner Standards indicate that each section of the examination should be constructed to take 90 minutes to complete. An additional 30 minutes is allotted for review of the examinations. The amount of time taken for the operators to complete and review the revised examinations indicated that the licensee's original time validation was not accurate.

Revisions were made to many of the written examination questions, mostly to raise the level of knowledge to be evaluated. Questions were reworded to test the operator's comprehension or ability to analyze rather than the operator's memory. Revisions were also required to ensure that the question met the intended test objective.

3.4 Walk-Through Examination Preparation and Administration

All of the facility proposed JPMs required changes to upgrade the performance standards for the individual steps required to perform the tasks. Many of the JPMs required changes to the initial conditions, initiating cues, or to references to make them complete and accurate. These generic problems were identified to the licensee early in the preparation process and the training personnel did a thorough job in correcting the deficiencies prior to the on-site preparation week.

Steps had to be added to most of the JPMs to verify prerequisites and to review precautions and limitations prior to performing the tasks. Failure to review prerequisites, precautions, and limitations was identified as generic weakness during the walk-through examinations. The operators' weakness could be attributed to failure of the training department to train the operators to perform these actions when performing JPMs.

Some of the JPM questions had to be modified or replaced because they did not probe deeply into the operator's knowledge of the system or task. The questions that were revised or replaced, for the most part, required only one or two word answers.

3.5 Dynamic Simulator Examination Preparation and Administration

Many of the ISCTs identified in the scenarios proposed by the facility did not meet the guidance of the Examiners Standards. The ISCTs identified by the facility did not always have plant or public safety significance and often lacked measurable performance indicators. The licensee's identification of ISCTs was also inconsistent from scenario to scenario. During the examination preparation week, invalid ISCTs were deleted and the scenarios were modified to add measurable performance indicators and feedback when possible. For example, a failure of an isolation valve was added to provide a measurable performance indicator and a stuck open relief valve was added to provide feedback on performance of an ISCT.

The scenarios proposed by the facility had to be modified to meet the guidance of the Examiners Standards with respect to EOP usage. Two of the four proposed scenarios did not require decisions or transitions within the EOPs. A previous NRC inspection team identified weaknesses in the licensee's training on use of the EOPs for complex scenarios (Inspection Report No. 50-271/88-200). The scenarios also had to be modified to ensure that all SROs were evaluated in their ability to use Technical Specifications.

The licensee objected to some of the changes proposed by the NRC during the examination preparation week. The licensee's objections were based on the inability of the simulator to realistically model conditions in the secondary containment for a stuck open Scram Discharge Volume (SDV) drain valve. The NRC agreed that the proposed scenario should not be used as part of the requalification examinations, but requested that the scenario be run on an operating crew for observation only. The results of this observation indicated that the operators would not take action to reset the scram with a high radiation levels in the secondary containment even after the stuck open SDV drain valve had been identified The operators' failure to take corrective action to isolate the leak in this situation is considered to be a training weakness.

During administration of the dynamic simulator examinations, the NRC identified a safety concern that the roles and responsibilities of the crew members did not provide independent evaluation of plant operations to assist and advise shift supervision during abnormal and emergency events. Poor communications and coordination of activities contributed to the weaknesses noted in crew performance. The licensee's immediate corrective actions to address this concern are discussed in section 6.0. A previous NRC inspection team identified similar weaknesses in crew performance when dealing with complex casualties (Inspection Report No. 50-271/88-200).

The NRC noted several weaknesses in the facility's evaluation techniques for dynamic simulator examinations. The facility evaluators did not closely observe control board manipulations and very seldom

followed operators to the back panels to observe activities. In several cases, the facility's evaluation of ISCT performance or pass/fail decision basis differed from the NRC's evaluation or basis. For example, when an SRO incorrectly implemented the EOPs and directed emergency depressurization of the reactor unnecessarily, the licensee did not consider his actions as failure of an ISCT, while the NRC did. The licensee's basis for their evaluation was that the unnecessary depressurization did not have an adverse effect on the safety of the public. The operator's incorrect action resulted in an unnecessary challenge to plant systems and, therefore, had an adverse effect on the safety of the plant. Even though the NRC and facility overall pass/fail results for individuals were identical, it appeared that the facility evaluators had a tendency to downplay the safety significance of the performance errors.

3.6 Program Weaknesses

The following program weaknesses were identified during the preparation, administration, and grading of the NRC administered requalification examinations:

- Quality of licensee developed examination materials, including the sample plan (section 3.2), time validation of the written examination materials (section 3.3), JPMs (section 3.4), and dynamic simulator scenarios (section 3.5).

- Rotation practices for ROs in the simulator (section 3.1).

- Evaluation techniques on the dynamic simulator portion of the examination (section 3.5).

- Definition of roles and responsibilities for the crew members that did not provide independent evaluation of plant operations to assist and advise shift supervision during abnormal and emergency events.

- Crew communications and coordination of activities.

4.0 Licensed Operator Medical Conditions

On February 22, 1991, the licensee identified to the NRC that they had failed to notify the NRC that a licensed operator required corrective lenses to perform licensed duties within thirty days of learning of the diagnosis as required by 10 CFR 55.25. The need for corrective lenses is a condition which requires a conditional license in accordance with 10 CFR 55.33(b). The need for corrective lenses was identified during a medical examination on April 12, 1990. The operator's license has been inactive for approximately six years and the operator did not perform licensed duties during the period from April 12, 1990 to February 22, 1991.

The apparent cause of the failure to notify the NRC was a lack of responsibility for making medical notifications within the licensee's organization. There was confusion among the Training, Operations, and Human Resources departments as to which department was responsible for identifying and making medical notifications. Licensee procedures did not clearly identify responsibility for notifying the NRC of changes in medical conditions. Licensee representatives indicated that corrective action would be taken to identify responsibility for notification of changes in licensed operator medical conditions.

The violation of 10 CFR 50.74 and 55.25 in failing to notify the NRC within thirty days of a condition requiring a conditional license is considered a non-cited violation. The licensee-identified violation is not being cited because the criteria specified in Section V.G.1 of the Enforcement Policy were satisfied. The licensee identified the violation. The event had minimal safety significance because the operator did not perform licensed duties during the period in question (271/91-02-02).

5.0 Emergency Operating Procedures

Scope/Background

During the preparation for the examinations, the examiner noted that the Vermont Yankee Emergency Operating Procedures (EOPs) are substantively different from the BWR Owners Group Emergency Procedure Guidelines (EPGs). The Vermont Yankee Procedure Generation Package (PGP) was reviewed to evaluate the justification for the differences from the EPGs. The examiner focused on the Plant Specific Technical Guideline (PSTG) portion of the PGP. The examiner did not have the resources to examine all of the deletions, deviations, and additions, but selected a sample to review and held limited discussions with the VY EOP coordinators. Documents reviewed are listed in Attachment 8.

Many of the items selected for review by the examiner were also questioned during the EOP team inspection in 1988 (Inspection Report 50-271/88-200, Attachment 8 - Reference 1). The EOP team inspection determined that the PSTG differences had not been reviewed by the NRC staff as part of the PGP submittal and requested a resubmittal of the PSTG. The EOP team inspection did not conclude that the justifications were technically adequate.

In the Safety Evaluation Report (SER) issued June 7, 1990 (Attachment 8 - Reference 4), the NRC staff reviewed the licensee PGP submittal, including the resubmitted PGP (Attachment 8 - Reference 3). The SER provided no specific comments on the PSTG portion of the procedure generation package, since it was based on revision 3 of the BWROG EPGs. Instead the NRC staff requested that the "PSTG program description should be revised to conform with Revision 4 of the General Electric Boiling Water Reactor Owner's Group (BWROG) Emergency Procedure Guidelines (EPGs). Deviations from the BWROG Emergency Procedure Guidelines should be documented, justified, and archived for future reference."

In inspection report 50-271/90-16 (Attachment 8 - Reference 5), a review of the licensee actions following the EOP team inspection was performed. In part, the inspection determined that the licensee PGP contained the justifications for deviations, deletions, and additions to the EPGs, but did not perform a technical adequacy assessment of the justifications.

Findings

" current VY PSTG and the VY EOPs implement Revision 4 of the BWR Owners Group Emergency Procedure Guidelines (EPGs). The VY PSTG contains approximately 50 deletions, 135 deviations, and 20 additions to Revision 4 of the EPGs. The justifications for some of the deletions to the EPGs are valid since they apply to design features applicable to other product lines of the BWR not associated with Vermont Yankee (i.e., isolation condenser or Mark III containment type).

Many of the deletions, deviations and additions do not appear to be adequately justified. Many of the concerns that were identified are based on the Lessons Learned from the Special Inspection Program for Emergency Operating Procedures (NUREG-1358), published in April 1989. The Vermont Yankee Procedure Generation Package (VY PGP) does not reference NUREG-1358. The VY PGP also does not reference NUREG/CR-5228 "Techniques for Preparing Flowchart-Format Emergency Operating Procedures," published in January 1989. These documents were issued to provide licensees additional information in preparing Emergency Operating Procedures.

Based on the discussions with the VY EOP coordinators, the inspector was concerned about the limited function of the VY PSTG. NUREG 1358 indicates that the PSTG should be the plant specific document that identifies the equipment or systems to be operated and the list of steps necessary to mitigate the consequences of transients and accidents and to restore safety functions. The PSTG establishes the logic for the symptom based accident mitigation strategy which then is translated into procedures. The VY EOP coordinators indicated that the VY PSTG addresses those plant specific items that are not contained in other station procedures (OP, ON, OT, or Alarm Response Procedures) and does not need to establish the accident mitigation strategy. The VY PSTG does not appear to adequately provide the required logic to assess whether the procedures (other than the EOP flowcharts) contain the required actions to mitigate the symptoms as described in the EPGs. There is also no basis defined in the PSTG to establish the EPG logic contained in procedure OE-3100 Scram. Many of the deviations and deletions appear to stem from the licensee's apparent misconception on the purpose and function of the PSTG.

The licensee took many deviations from the BWROG EPGs and made numerous deletions and additions to the EPGs. The deviations, deletions, and additions listed in Attachment 7 do not have adequate or complete justification, or are not adequately analyzed for impact on the logic of the accident mitigation strategy as defined in the EPGs. Attachment 7 should not be construed to be a comprehensive list of justifications that are not adequate or complete in the VY PSTG.

Conclusion

The Vermont Yankee PSTG contains many additions, deletions, and deviations from revision 4 of the EPGs that do not appear to be adequately justified. Many of the areas of concern focus on the implementation of the RPV pressure control strategy of the BWROG EPGs. Many of the justifications appear to use event based reasons to justify departure from the accident mitigation strategy of the EPGs for symptom based emergency operating procedures. It is not apparent that Vermont Yankee has unique features and/or analysis to support the many additions, deletions, and deviations in the PSTG. Pending additional review this is considered to be an unresolved item (271/91-02-03).

6.0 Exit Meeting and Basis for Continued Operation

An exit meeting was held at the conclusion of the examinations in the Region I office on March 4, 1991. The personnel in attendance are listed in Attachment 1. The NRC results of the individual examinations and the requalification program evaluation were presented. The NRC findings related to licensed operator medical examinations and Emergency Operating Procedures were also presented. The NRC concerns related to the roles and responsibilities of crew members and weaknesses in crew communications and teamwork were discussed. The licensee was given an opportunity to discuss the basis for their satisfactory evaluation of the crew that the NRC evaluated as unsatisfactory. The final result of the meeting was that the licensee's requalification program is considered to be unsatisfactory.

Due to the unsatisfactory requalification program and the concerns related to roles and responsibilities of the crew, the licensee was asked to present a Basis for Continued Operation (BCO). The licensee based their BCO on past performance of licensed operators, recent evaluations of individual operator performance, and accreditation of the operator training programs. They also implemented immediate and short term corrective actions to redefine the roles and responsibilities of the crew members and provide training on teamwork and communications. The licensee formally transmitted their BCO in a letter dated March 8, 1991. The NRC issued Confirmatory Action Letter (CAL) I-91-007 on March 11, 1991, which accepted the licensee's BCO pending the results of operational evaluations of selected operating crews (discussed in section 7.0). The licensee letter and NRC CAL are included as attachments 4 and 5 of this report.

7.0 Operational Evaluations

The NRC conducted operational evaluations of three operating crews to assess the licensee's immediate corrective actions and the acceptability of the BCO on March 9, 1991, and March 19, 1991. The licensee performed evaluations of the two crews evaluated on March 19, 1991, in parallel with the NRC. The purpose of the parallel evaluations was to allow the NRC to assess the licensee's evaluation techniques in light of the differing evaluations of crew performance during the requalification examinations.

All three crews that were evaluated demonstrated satisfactory performance. Communications and coordination of activities among crew members were improved compared to the requalification examinations. Redefining the role of the SE and, more importantly, the SCRO provided the independent assessment of plant operations during emergencies that was lacking during the requalification examinations. These results indicated that the licensee's immediate and short term corrective actions were effective.

The licensee's evaluation results were, in general, similar to those of the NRC. The similar results restored confidence in the licensee's ability to evaluate crew performance and in their ability to continue to safely operate the plant.

Attachments:
1. Persons Contacted
2. Requalification Test Items
3. Licensee's Evaluation Results (dated March 8, 1991)
4. Licensee's Justification for Continued Operation (dated March 8, 1991)
5. NRC Confirmatory Action Letter I-91-007 (dated March 11, 1991)
6. Simulation Facility Report
7. Vermont Yankee PSTG/EPG Revision 4 Differences
8. Emergency Procedure Reference Documents

ATTACHMENT 1

PERSONS CONTACTED

Vermont Yankee Nuclear Power Corporation

W. Murphy, Senior Vice President - Operations (1), (2), (3)
D. Reid, Plant Manager (2), (3)
R. Wanczyk, Operations Superintendent (1), (3)
J. Herron, Operations Supervisor (1), (2), (3), (5)
L. Doane, Assistant Operations Supervisor (1), (3), (4), (5)
R. Spinney, Training Manager (1), (2), (3)
G. LeClair, Operations Training Supervisor (3)
A. Chesley, Simulator Supervisor (3)
G. Ludlam, Licensed Operator Requalification Program Coordinator (1), (2), (3), (4), (5)
M. Sontag, Operations Training Instructor (1), (2), (3), (4), (5)
E. Harms, Operations Training Instructor (1), (3), (4)
L. Amirault, Operations Training Instructor (1), (3), (4)
S. Brown, Operations Training Instructor (1), (4)
F. Helin, Project Engineer (3)
J. Kinsey, Project Engineer (3)
A. Mears, Internal Auditor (3)

Nuclear Regulatory Commission

W. Hodges, Director, Division of Reactor Safety (2), (3)
W. Lanning, Deputy Director, Division of Reactor Safety (2)
J. Johnson, Chief, Projects Branch 3 (2)
J. Rogge, Chief, Section 3A, Division of Reactor Projects (2)
R. Conte, Chief, BWR Section (2), (3), (6)
T. Walker, Senior Operations Engineer (1), (2), (3), (4), (6)
D. Florek, Senior Operations Engineer (2), (4)
C. Sisco, Operations Engineer (2)
H. Eichenholtz, Senior Resident Inspector
T. Hiltz, Resident Inspector (3)
R. Gallo, Chief, Operator Licensing Branch, NRR (2)
B. Wetzel, License Examiner, OLB, NRR (3), (6)
C. Tyner, Examiner (INEL) (1), (4), (6)
T. Morgan, Examiner (INEL) (3), (6)

NOTES:

(1) Attended Entrance Meeting, February 11, 1991
(2) Attended Exit Meeting, March 4, 1991
(3) Attended Exit Meeting, March 19, 1991
(4) Member - Combined Facility/NRC Exam Team
(5) Facility Evaluator
(6) Operational Evaluator

ATTACHMENT 2

REQUALIFICATION TEST ITEMS

Simulator Scenarios

SEG-4, Diesel Trip, Loss of Startup Transformers, Turbine Trip
SEG-5, FRV Controller Failure, Main Steam Line Rupture in Drywell
SEG-7, Recirc Flow Controller Failures, RCIC Steam Line Break
SEG-18, CRD Pump Trip, Loss of Condenser Vacuum, Failure to Scram

Job Performance Measures

20015 - Isolate and Vent the Scram Header
20018 - Terminate and Prevent Injection
20025 - Fill the Torus from HPCI
20301 - Place Battery Charger CA-1 In Service
20106 - Place Standby CRD Flow Control Valve In Service
21202 - Startup an RPS MG Set
21502 - Shift APRM/LPRM Power Supply to Normal Source
20006 - Emergency Fill Main Condenser with Service Water
20032 - Bypass a Group I Isolation and Open the MSIVs
20022 - Perform Manual Insertion of Control Rods
20503 - Startup Torus Cooling from the Remote Shutdown Panel
22303 - Reset a Group III Isolation
26101 - Manually Initiate SBGT
26207 - Transfer Station Loads from Aux to Startup Transformers
26402 - Shutdown the Diesel Generator after Surveillance

Written Examination

Part A - Static Simulator		Part B - Classroom	
SSE-16	SSE-17	SRO	RO
642	194	610	614
303	725	598*	598 *
711	744	028	596
436	149	669	017
618	343	221*	589
294	301	201*	467
192	326	379	221*
750 (SRO only)	254	746	201*
741	742	127*	521
503	268	275	346
716	743 (SRO only)	110*	127*
		373	110*
		407	429
		043	745
		747	401*
		393	392
		401*	410*
		410*	748
		749	681
		428	424*
		98	
		424*	

* Common to RO and SRO Exam

Attachment 3

VERMONT YANKEE NUCLEAR POWER CORPORATION

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

(802) 257-5271

March 8, 1991

TDL 91-004
BVY 91-027

United States Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 20555

Attention: Mrs. Tracy Walker, Lead Examiner
USNRC Region I

References: a) License No. DPR-28 (Docket No. 50-271)
b) NUREG 1021, Operator Licensing Examiner Standards

Subject: Licensed Operator Requalification (LOR)

Dear Mrs. Walker:

A licensed operator requalification examination was jointly administered to twelve license holders at the Vermont Yankee Training Center and Station by the USNRC and the licensee during the week of February 25, 1991. Pursuant to Section ES-601 of reference b, the Vermont Yankee Training Department conducted an LOR training program evaluation. Vermont Yankee's evaluation results are enclosed.

If you have any questions regarding these results, please contact me. Thank you.

Very truly yours,

Randall W. Spinney

Randall W. Spinney
Training Manager

RWS/pjs/910306.1

Enclosure

cc: USNRC Region I Administrator
USNRC Project Manager - VYNPS
USNRC Resident Inspector - VYNPS

~~9103180042~~

VERMONT YANKEE NUCLEAR POWER CORPORATION
REQUALIFICATION PROGRAM EVALUATION BASED ON 1991 EXAMINATION

1. Individual Examination Results

	RO Pass/Fail	SRO Pass/Fail	TOTAL Pass/Fail
Written	4/0	8/0	12/0
JPM/Walkthrough	4/0	8/0	12/0
Simulator	4/0	6/2	10/2
Overall	4/0	6/2	10/2

2. Crew Examination Results

1 of 3 Crews Failed

3. Program Evaluation Results

The facility performed an evaluation of the requalification program based on the facility's examination results. The criteria for program evaluation as specified in ES-601 was used where appropriate. A complete comparison could not be made between facility and NRC results since NRC results were not available. The sample size (12) met the minimum requirement of ES-601. The facility results are:

- One of three crews failed the simulator portion of the examination.
- 100% of the operators passed the written examination.
- Two operators failed the simulator portion of the examination.
- 100% of the operators passed the JPM portion of the examination.
- All operators were trained and evaluated in all positions permitted by their individual licenses.
- Based on feedback from operators, facility observers, and the NRC team, it is felt that all the facility evaluators performed in a satisfactory manner.
- Common weaknesses on JPM's are as follows:
  - 50% of the operators administered JPM 20018 performed unsatisfactorily (Terminate and Prevent Injection per OE 3102 2/3)
  - 66% of the operators administered JPM 21202, question one (1) performed unsatisfactorily. (Purpose of RPS Flywheel)
  - 100% of the operators administered JPM 22302 question two (2) performed unsatisfactorily. (Group III isolation signals)
- Common weaknesses on the written examinations are as follows:
  - 42% of the operators missed question number one on static simulator scenario 16. (Venting containment with isolation signal present)
  - 25% of the operators missed question number seven on static simulator scenario 17. (RWCU pump trip signals)

3. Program Evaluation Results (cont'd)

- 16% of the operators missed question numbers one, three, five, and nine of simulator scenario 17. (AOG recombiner shifting, ADS initiation logic, Load sequencing following an LNP, and steps required to place torus spray in service with a LOCA signal present.)

The Simulator portion of the operating examination revealed the following weaknesses. These identified weaknesses will be addressed.

- Overall Communications and Feedback among crew members was weak
- The Shift Supervisors' direction and use of manpower was weak (Command and Control)
- The CRO was given direction to maintain reactor level using feed/condensate during an LNP. The level went outside the band low prior to the CRO informing the Shift Supervisor that power was not available to the pump.
- The Shift Supervisor discussed using RWCU to reduce reactor water level with a gross fuel element failure present
- Two Senior Reactor Operators conservatively misinterpreted Technical Specification Operability Requirements resulting in a manual scram
- A crew was slow to diagnose an inadvertent RCIC injection
- A crew (several individuals) failed to completely back up a Group III Isolation
- Clarification of an EAL on Emergency Classification Procedure is required (General Emergency)
- An off-normal procedure (ON 3145) requires clarification (previously addressed and being currently implemented)
- A Shift Supervisor failed to correctly read a decision block on an Emergency Operating Procedure resulting in an unnecessary RPV-ED

4. Written Examination Results

The written examination completion times fell within the guidelines of ES-602.

	Classroom	SEG 16	SEG 17
ES-602			
Completion Time	90 minutes (minimum)	45 minutes (minimum)	45 minutes (minimum)
Review Time	30 minutes	15 minutes	15 minutes
	120 minutes (maximum)	60 minutes (maximum)	60 minutes (maximum)
Operator Average			
Completion Time	100 minutes	47 minutes	55 minutes

INDIVIDUALS WRITTEN EXAMINATION RESULTS

Operator	Section A points	Section B points	Overall Score %
1	19 of 20	19 of 20	95%
2	22 of 22	22 of 22	100%
3	20 of 22	22 of 22	95.5%
4	19 of 20	18 of 20	92.5%
5	21 of 22	22 of 22	97.7%
6	19 of 20	20 of 20	97.5%
7	20 of 22	21 of 21	93.2%
8	18 of 22	22 of 22	90.9%
9	19 of 20	20 of 20	97.5%
10	20 of 22	22 of 22	95.5%
11	20 of 22	22 of 22	95.5%
12	20 of 22	21 of 22	93.2%

5. Walkthrough/JPM Examination Results

Operator	JPM	Questions	Score
1	5 of 5	8 of 10	95%
2	4 of 5	10 of 11	82.7%
3	5 of 5	13 of 13	100%
4	4 of 5	10 of 10	85%
5	4 of 5	11 of 12	82.9%
6	4 of 5	9 of 10	82.5%
7	5 of 5	11 of 12	97.9%
8	5 of 5	11 of 12	97.9%
9	5 of 5	8 of 10	95%
10	5 of 5	12 of 12	100%
11	5 of 5	10 of 12	95.8%
12	5 of 5	11 of 12	97.9%

6. Recommendations for Improvement

JPM Evaluation

- The JPM's will be updated to ensure proper performance standards and cues

Simulator Evaluation Guide

- All scenarios will be reviewed to ensure consistent ISCT usage
- All scenarios will be reviewed to ensure proper complexity and depth to require multiple use of EOP's
- Scenarios will be developed that relate to industry events, LER's, and SOER's

Attachment 4

VERMONT YANKEE NUCLEAR POWER CORPORATION

Ferry Road, Brattleboro, VT 05301-7002

BVY 91-25

Warren P. Murphy
Senior Vice President, Operations

(802) 257-5271

March 8, 1991

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

Attn: Document Control Desk

References: a) License No. DPR-28 (Docket No. 50-271)

Dear Sir:

Subject: Operator Licensing Requalification Training Program, Justification for Continued Operation

As requested at the March 4, 1991 meeting with NRC Region I, Attachment 1 provides our Justification for Continued Operation.

Based on the assurances identified in the Attachment, we conclude that a sufficient number of licensed personnel are fully qualified to operate the plant safely and protect the health and safety of the public.

Should you have any additional questions, please do not hesitate to contact us.

Very truly yours,

Vermont Yankee Nuclear Power Corporation

Warren P. Murphy

Warren P. Murphy
Senior Vice President, Operations

/dm

cc: USNRC Regional Administrat[illegible] Region I
USNRC Resident Inspector, [illegible]NPS
USNRC Project Manager, VYNPS

~~9103120164~~

ATTACHMENT 1

LICENSED OPERATOR REQUALIFICATION (LOR) TRAINING PROGRAM JUSTIFICATION FOR CONTINUED OPERATION

DESCRIPTION OF CONDITION AND EVENTS:

During the observation by the NRC of Vermont Yankee's LOR examination process, the NRC determined that the VY LOR training program is unsatisfactory.

Of the twelve individual licensees examined, ten received a Satisfactory grade (83% success rate). However, two of the three crew examinations resulted in Unsatisfactory grades. Therefore, based on the results of the crew examinations, the LOR training program was determined to be unsatisfactory in accordance with the requirements established in NUREG 1021, Section ES-601, "Administration of NRC Requalification Program Evaluations."

The above NRC examination results are consistent with the VY training program results, with the exception of the rating given for the staff crew's Simulator Dynamic Demonstration.

SAFETY SIGNIFICANCE AND JUSTIFICATION FOR CONTINUED OPERATION:

This condition has minimal safety significance in the short term, because there is reasonable assurance that a sufficient number of licensed operators are fully qualified to safely operate the facility and protect the health and safety of the public. This determination is based on the following:

1. Prior to this event, there had been no observation of significant deficiencies in the LOR training program by the NRC.

2. Those individuals (2) and crews (2) who were deemed deficient by the NRC were relieved of their licensed duties. None of the individuals involved will be reinstated until they receive remedial training and satisfy the program requirements of NUREG 1021. None of the NRC-identified crew deficiencies or weakness resulted in a safety-significant condition that was outside the bounds established by Vermont Yankee's EOPs.

3. Within the past six weeks 24 individuals and six crews (including two staff crews) were examined and all received Satisfactory ratings. One crew was deemed marginal in passing, and completed remedial training prior to resuming licensed duties.

4. The LOR examination process consists of three functional areas. In two of these areas [written exam and Job Performance Measures (JPMs)] there was a 100% Satisfactory rating. Both the NRC and Vermont Yankee were in full agreement with these ratings.

5. In NRC's evaluation of the previous LOR training program, all individuals (12) and crews (3) examined received Satisfactory grades. Both the NRC and Vermont Yankee were in full agreement with these ratings. Since that time, there has been no relaxation in our operator training program standards, nor was there a substantial difference between Vermont Yankee and NRC standards with regard to the necessary scope and content of the training program.

6. All operator training programs, including the Licensed Operator Requalification Program, were extensively reviewed and reaccredited in June of 1990 by the National Academy for Nuclear Training.

7. Within the past few weeks, during the NRC's examination of the 1990/1991 Licensed Operator Initial (LOI) candidates, all individuals (4) examined received Satisfactory grades and licenses were granted.

8. As documented in the NRC's Systematic Assessment of Licensee Performance (SALP) since May of 1983 (5 SALP periods), Vermont Yankee has consistently achieved a SALP rating of 1 in the areas of Plant Operations and Training.

9. During the 1989/1990 time period, the licensed operators safely operated the facility, as evidenced by operator performance records that included a good level of knowledge, adherence to licensed conditions, and adherence to procedures. The noteworthy performance by the operators of their licensed duties have been documented in the following NRC audit and inspection reports:

   a) <u>SALP Report No. 50-271/88-99</u>

      "Several licensee strengths were evident during this SALP period. Plant operating history, which included a successful refueling outage and a low plant transient rate attest to good overall management involvement, good operational oversight, a good training program, and a strong orientation at all levels toward nuclear safety."

      "The licensee continued to demonstrate strong and effective management controls that assured safe facility operations, as evidenced by a good plant performance record, adherence to license conditions and safe operating practices, and a demonstrated commitment to safety by operators and supervisor personnel."

      "In general, operator performance during plant transients and routine operations was noteworthy and reflects the success of the operator training program."

      "Plant staff, management, and operators performed well during event identification, classification, and emergency plan implementation."

"The current requalification training program for licensed operators made a positive contribution to safe operation of the plant, as evidenced by operator performance records that included a good level of knowledge, adherence to licensed conditions, and adherence to procedures."

b) NRC Inspection Report No. 50-271/90-02

"Plant operators were observed to display proper attention to detail and a good questioning attitude in carrying out their licensed responsibilities. Plant evolutions were observed to be conducted in accordance with approved plant procedures."

c) NRC Inspection Report No. 50-71/90-03

"On June 1 at 1:53 p.m., an automatic reactor trip occurred from 100% power...Recovery operations were observed by the inspector. Control room personnel and plant support staff response was good. Proper command and control was demonstrated by control room supervisory personnel. The monitoring of plant and equipment conditions was performed in a diligent manner and communicated well between the members of the operating crew. The Shift Engineer was involved and properly integrated into the process. The development by the VYNPS Operations Department of an atmosphere in which licensed operators identify deviations from expected performance is viewed as a strength."

d) NRC Inspection Report No. 50-271/90-15

"A reactor trip during performance of weekly turbine emergency governor trip test was handled well by operators and followup maintenance actions were timely and effective."
"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."

e) NRC Inspection Report No. 50-271/90-07

"The Shift Supervisor/Plant Emergency Director (SS/PED) displayed excellent command and control. He tracked accident information closely and assigned, or reassigned, personnel to follow-up new developments."

10. The Plant Operations Review Committee concluded on March 2, 1991, that there is sufficient confidence in the qualifications and abilities of the licensed operators and their crews to protect the health and safety of the public.

11. The following immediate corrective actions have already been completed:

 a) Based on the NRC Region I assessment of control room staff functions during abnormal and emergency conditions, the following division of duties was initiated on March 1, 1991:

  1. The Shift Engineer's sole duty, during abnormal and emergency conditions, was reinforced so that he provides continual independent technical assessment for the Shift Supervisor.

  2. An additional individual (i.e., communicator) not currently required to meet minimum operating shift compliment (2 SRO's, 2 RO's, SE) was designated to perform the following duties during abnormal and emergency conditions:

   a) NRC and state notifications;
   b) data collection; and
   c) any similar non-licensing activities, as directed by the Shift Supervisor.

  3. Both the Shift Engineer and Communicator are capable of responding to the Control Room within 10 minutes notification.

 b) The management directive on Shift Supervisor command duties and responsibilities was revised and reissued on March 4, 1991 and discussed with the Senior Licensed Operators and Shift Engineers during a meeting on March 5, 1991. Included in that directive was the following:

  "The Shift Supervisor is responsible for ensuring that his crew performs as a team and adheres to the concepts of team work, particularly during off normal conditions. Specifically, he is responsible for stressing and practicing this concept in the simulator during training exercises."

12. The following short-term corrective actions will be completed:

 Vermont Yankee will develop and implement an Instructor Guide (IG) to address lessons learned during the 1991 LOR examination process, relative to command and control functions and communications of shift supervision. Classroom training will be concluded by March 23, 1991. Simulator crew exercises to reinforce that training will be conducted for all operators with active licenses and be completed by March 31, 1991.

13. The following long-term corrective actions will be completed:

a) The VY President and Chief Executive Officer; Senior Vice President, Operations; Plant Manager; and Operations Superintendent will each observe each crew's performance during the 1991 Operator Requalification cycle ending February 1992.

b) A detailed root cause analysis for the failure rate in this NRC monitored exam will be conducted and completed by April 15, 1991, followed by a report to the NRC by April 30, 1991.

/dm

Attachment 5

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION I
475 ALLENDALE ROAD
KING OF PRUSSIA, PENNSYLVANIA 19406

MAR 11 1991

Docket No. 50-271
CAL No. 91-007

Vermont Yankee Nuclear Power Corporation
ATTN: Mr. Warren P. Murphy
Senior Vice President, Operations
RD 5, Box 169
Ferry Road
Brattleboro, Vermont 05301

Gentlemen:

Subject: CONFIRMATORY ACTION LETTER (CAL) I-91-007 UNSATISFACTORY LICENSED OPERATOR REQUALIFICATION TRAINING PROGRAM

The purpose of this letter is to confirm our understanding of those actions which you have taken or will take to ensure that the deficiencies and weaknesses identified in your licensed operator requalification program are promptly corrected and that adequate proficiency of licensed operators is maintained at the Vermont Yankee Nuclear Power Station. These deficiencies and weaknesses were identified during the licensed operator requalification examinations administered by NRC Region I during the week of February 24, 1991, in which two of the three crews and two individuals examined demonstrated unsatisfactory performance. Based on these results, the NRC has determined that the Vermont Yankee requalification program is unsatisfactory.

The examination deficiencies and weaknesses were discussed with you during an exit meeting held with the Deputy Regional Administrator on March 4, 1991 in Region I. Your Justification for Continued Operation forwarded by letter dated March 8, 1991, is acceptable, pending the results of NRC operational evaluations conducted March 9, 1991 and to be conducted March 19, 1991, or later.

Based upon the commitments contained in your March 8, 1991 letter, it is our understanding that you have or will:

1. Remove from licensed duties those individual licensed operators who failed the NRC administered requalification examination, or who were part of the crew failures, during the week of February 24, 1991.

2. Prohibit individuals failing any portion of an NRC requalification examination or who were a part of the crew failures to resume licensed duties before successfully passing an NRC administered requalification examination or operational evaluation, as applicable. This applies during the period that the NRC deems the Vermont Yankee licensed operator requalification program to be unsatisfactory.

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3. Establish division of duties and train shift operating personnel on those duties as defined (Item No. 11) in your letter of March 8, 1991.

4. Train all shift operating personnel on lessons learned from the 1991 licensed operator requalification examination by March 31, 1991.

5. Identify problems and root causes as to why the requalification program was unsatisfactory. This analysis will identify a corrective action program with a schedule of milestones. Submit the root cause analysis with proposed corrective action to NRC Region I by April 30, 1991.

The Vermont Yankee licensed operator requalification program shall be deemed to be unsatisfactory until further correspondence from the NRC changes that status.

Issuance of this Confirmatory Action Letter does not preclude the issuance of an Order formalizing the above commitments. If your understanding differs from that set forth above, please call me immediately. The responses directed by this letter are not subject to the clearance procedures of the Office of Management and Budget, as required by the Paperwork Reduction Act of 1980, PL 86-511.

Thank you for your cooperation.

Sincerely,

Thomas T. Martin

Thomas T. Martin
Regional Administrator

cc: w/encl:
J. Weigand, President and Chief Executive Officer
J. Pelletier, Vice President, Engineering
D. Reid, Plant Manager
J. DeVincentis, Vice President, Yankee Atomic Electric Company
L. Tremblay, Senior Licensing Engineer, Yankee Atomic Electric Company
J. Gilroy, Director, Vermont Public Interest Research Group, Inc.
G. Iverson, New Hampshire Office of Emergency Management
Vermont Yankee Hearing Service List
Public Document Room (PDR)
Local Public Document Room (LPDR)
Nuclear Safety Information Center (NSIC)
K. Abraham, PAO (24) Salp Reports and (2) All Inspection Reports
NRC Resident Inspector (w/SGI)
State of New Hampshire, SLO Designee
State of Vermont, SLO Designee
Commonwealth of Massachusetts, SLO Designee

bcc: w/encl:
Region I Docket Room (with concurrences)
Management Assistant, DRMA (w/o encl)
J. Joyner, DRSS
J. Johnson, DRP
J. Rogge, DRP
H. Eichenholz, SRI - Vermont Yankee
T. Koshy, SRI - Vermont Yankee
M. Conner, SALP Reports Only
K. Brockman, EDO
M. Fairtile, NRR
J. Lieberman, OE
J. Goldberg, OGC
J. Partlow, NRR
D. Holody, RI

ATTACHMENT 6

SIMULATION FACILITY REPORT

Facility Licensee: Vermont Yankee Nuclear Power Corporation

Facility Docket No: 50-271

Requalification Examinations Administered on: February 27 - 28, 1991

Operational Evaluations Administered on: March 9 and 19, 1991

This form is used to report observations. These observations do not constitute audit or inspection findings and are not, without further verification and review, indicative of non-compliance with 10 CFR 55.45(b). These observations do not affect NRC certification or approval of the simulation facility other than to provide information which may be used in future evaluations.

1. The modelling of secondary containment response is very limited. Very few malfunctions are available to evaluate the operators' ability to implement the Secondary Containment Control procedure. The NRC proposed a scenario in which reactor coolant discharged to the secondary containment through stuck open Scram Discharge Volume (SDV) drain valves. This scenario could not be used because the simulator did not model the response of the sumps and area temperatures to the leak.

2. Whenever a significant fuel failure is introduced, radiation levels in the secondary containment increase dramatically even without a primary system discharging into the secondary containment. The licensee explained that the radiation levels were due to "shine" from the primary containment. This response to fuel failure has not been observed on other BWR simulators in Region I; therefore, the examiners questioned the validity of the simulator modelling.

3. The response of the simulator to a Main Steam Line (MSL) rupture in the drywell between the MSIVs (malfunction MS11) was inconsistent. The event was run with all the MSIVs closed (on high MSL flow) and with the outboard MSIV in the ruptured MSL failing to close. In both cases, drywell pressure and temperature increased significantly, but in the latter case, the simulator did not recognize that the leak was isolated. This event was run several times with different failure rates for the malfunctions and with and without complete MSL isolation. The response of drywell pressure and temperature did not appear to be consistent with the conditions of the event.

4. No malfunction exists to fail drywell coolers electrically. An attempt was made to use an existing mechanical malfunction (drive belt failure) and instructor overrides for the indicating lights to make it appear that the drywell cooler had failed to start. This could not be done for all the drywell coolers and drywell response did not appear realistic for the malfunction.

5. ERFIS (SPDS) failed to reset after running a scenario which caused a delay in administration of the operational evaluations.

ATTACHMENT 7

VERMONT YANKEE PSTG/EPG REVISION 4 DIFFERENCES

1. EPG Statement

RPV Control Entry Condition - RPV pressure above high pressure scram setpoint.

VY PSTG Statement

N/A

VY PSTG Justification

It is not appropriate for an event based entry condition to be used for symptomatic based emergency operating procedures. Vermont Yankee used event based operating procedures (ONs and OTs) to address these types of events.

Basis for NRC concern

The VY PSTG does not describe unique design features or provide analysis that would justify deleting this symptom as an entry condition into symptom based emergency operating procedures.

2. EPG Statement

RPV Control Entry Condition - Drywell pressure above high drywell pressure scram setpoint.

VY PSTG Statement

N/A

VY PSTG Justific...

It is not necessary for an event based entry condition to be used for symptomatic based emergency operating procedures. Vermont Yankee used event based operating procedures (ONs and OTs) to address these types of events. The scram procedure would be entered which would cause entry into RPV control if low RPV level conditions exist.

Basis for NRC Concern

The VY PSTG does not describe any unique design features or provide analysis that would justify deleting this symptom as an entry condition into symptom based emergency operating procedures.

3. EPG Statement

RPV Control, RC/P-3 - When either:

- All control rods are inserted to or beyond position 02, or
- It has been determined that the reactor will remain shutdown under all conditions without boron, or
- Cold shutdown boron weight has been injected into the RPV, or
- The reactor is shutdown and no boron has been injected into the RPV.

Depressurize the RPV and maintain a cooldown rate below 100°F/hr.

VY PSTG Statement

When:

- All control rods are inserted to or beyond position 02, or
- Cold shutdown boron weight has been injected into the RPV, and
- RPV level has been restored between 127" and 177"

Proceed to cold shutdown in accordance with Plant Restoration Procedure OP-109.

VY PSTG Justification

The operating crews normally do not have the time or resources necessary to determine if the reactor will remain shutdown under all conditions without boron. Since the only way that the reactor can be assured shutdown with no boron injected is when all rods have been inserted to or beyond position 02, the EPG statement, "It has been determined that the reactor will remain shutdown under all conditions without boron," can be removed from the VY PSTG. The direction to depressurize the RPV is given in OP-109 which is entered when all rods are inserted or cold shutdown weight of boron is injected and RPV level has been restored between 127" and 177".

Basis for NRC Concern

The VY PSTG does not consider the deviation in the context of the overall EPG RPV pressure control strategy as it relates to RPV Control, Primary Containment Control, Secondary Containment Control, Radiation Release Control, and the Contingencies. These procedures depend on the reactor pressure reduction as part of the overall accident mitigation strategy. The EPG considerations for beginning a pressure reduction are that the

reactor will remain shutdown during the cooldown and an emergency situation still exists (page I-4 of EPG). There is no consideration provided in the EPG for RPV level to be restored before a pressure reduction is initiated. Inclusion of the RPV level in the direction to begin a normal depressurization unnecessarily delays actions that could also mitigate the symptoms in other procedures.

It is appropriate to include the statement "It has been determined that the reactor will remain shutdown under all conditions without boron." The VY PSTG does not describe any unique features that would justify not including this statement. The statement does not direct operators to make this judgement, and other BWRs do not require operators to make this determination. If this information is available from either the reactor engineer or the Technical Support Center, then it can be used as part of the accident mitigation strategy.

4. EPG Statement

RC/P-1 Override - If while executing the following steps:

- Boron injection is required, and
- The main condenser is available, and
- There has been no indication of gross fuel failure or steam line break,

Open MSIVs, bypassing pneumatic system and low RPV water isolation interlocks, if necessary, to reestablish the main condenser as a heat sink.

VY PSTG Statement

If:

- MSIV isolation has occurred,
- The main condenser is available, and
- No indication of gross fuel failure or steam line break exists,

Open MSIVs, bypassing low low RPV water level and high steam flow not in RUN isolation interlocks if necessary to reestablish the main condenser as a heat sink.

VY PSTG Justification

Any time the main condenser is available and it is safe to do so, the operator is instructed to use it rather than the suppression pool as the heat sink. The restriction that boron injection is required for establishing the main condenser as the RPV heat sink is an unnecessary

restriction since it could needlessly challenge the suppression pool's heat capacity and delay reestablishment of the main condenser as the heat sink. The high steam flow when not in RUN (unique to VY due to the 105% bypass valve capability) is bypassed to prevent MSIV isolation during the pressure reduction.

Basis for NRC Concern

VY has a design feature of a 105% bypass valve capability and its use should be factored into the accident mitigation strategy. The EPGs support the use of the main condenser as a heat sink. The EPGs also are clear on those conditions which authorize use of defeating isolation interlocks to be able to use the main condenser as a heat sink. For the particular step in question, the EPGs do not allow defeating the MSIV isolation interlocks unless boron injection is required. This occurs when the reactor cannot be shutdown and the suppression pool temperature reaches the boron injection initiation temperature (BIIT). The BIIT is established to assure that the heat capacity temperature limit will not be exceeded when the hot shutdown boron weight is injected into the vessel during an ATWS, the MSIVs are closed, and no torus cooling is available. The VY PSTG defeats an isolation provision without analysis of the consequence of the actions. This may represent an unreviewed safety issue.

5. EPG Statement

N/A

VY PSTG Statement

If torus temperature is above 120°F and the RPV is isolated from the main condenser, commence depressurizing the RPV at normal cooldown rates to <200 psig unless Emergency RPV Depressurization is required.

VY PSTG Justification

VY Technical Specifications bases state that experimental data indicate that excessive steam condensing loads can be avoided if the peak temperature of the torus in maintained below 160°F during any period of relief valve operation. The NRC SER for revision 4 of the EPGs states that the implementation of plant specific procedures should be consistent with its safety analysis or provide the staff with additional information to remedy such deviations. The step is added to comply with the SER requirements.

Basis for NRC Concern

The licensee steps severely complicate the actions for responding to an ATWS event with the MSIVs closed and a relief valve operating. The licensee actions to depressurize the RPV are in direct conflict with the overall EPG strategy to combat ATWS scenarios. During the ATWS, the EPGs do not depressurize the RPV based on torus temperature considerations

unless the torus temperature is impacting the heat capacity temperature limit (HCTL). Based on the VY HCTL curve, this temperature is approximately 195°F. The VY PSTG does not describe unique features regarding the VY torus which would justify ATWS actions different than that contained in the EPGs. In addition had the licensee implemented the pressure control portion of the RPV control in accordance with the EPG guidelines, the procedure would require beginning a cooldown when reactor power is under control which would address the actions covered in the VY technical specifications. The operators are trained not to depressurize the RPV with an ATWS condition, which directly conflicts with the direction given in the suppression pool temperature control procedure.

6. EPG Statement

Secondary Containment Control Entry Condition - Secondary Containment differential pressure at or above 0 inches of water

VY PSTG Statement

N/A

VY PSTG Justification

The secondary containment is designed to withstand an internal overpressure corresponding to 7 inches of water without structural failure or pressure relief. The loss of secondary containment in and of itself does not constitute an unsafe or emergency condition. Concurrent high radiation levels within the secondary containment are required to constitute an emergency condition due to the potential for an uncontrolled release of radioactivity to the environment.

Basis for NRC Concern

The EPG entry conditions are symptomatic of both emergencies and events which may degrade into emergencies. The EPGs specify actions appropriate for both. Entry into procedures developed from the guidelines is not conclusive that an emergency has occurred. Differential pressure at or above 0 inches of water, is symptomatic of a condition which may degrade into an emergency. The VY PSTG does not describe unique features or provide analysis that justifies deleting this entry condition.

7. EPG Statement

SC/T-4.2 - When an area temperature exceeds its maximum safe operating temperature in more than one area Emergency Depressurization is required.

VY PSTG Statement

When a maximum safe operating temperature for a limiting combination is exceeded, Emergency Depressurization is required, enter Contingency 2 and execute it concurrently with this procedure.

VY PSTG Justification

In establishing the maximum safe operating temperature for various areas in secondary containment, the critical plant areas and equipment were determined and the environmental tolerance levels for the equipment obtained. Only those combinations of affected areas which could result in potential loss of both redundant trains of a required safety, or critical function need be considered. This addresses preservation of reactivity control, ECCS initiation, RPV level and pressure control, decay heat removal, ECCS cooling, and post-accident monitoring functions. Reference is also made to Memo, Swensen to Miller, BWROG EPG - Revision 1, dated 6/9/86.

Basis for NRC Concern

The VY distinction of limiting combination rather than more than one area does not address the consideration of a wide-spread problem which may pose a direct and immediate threat to secondary containment integrity. The VY PSTG can allow more than one area above the maximum safe temperature without requiring emergency depressurization. The VY PSTG does not address unique design features that would justify not implementing revision 4 of the EPGs. The PSTG justification also does not address temperature limitations due to personnel access requirements.

8. EPG Statement

SC/R-2.2 - If a primary system is discharging into secondary containment, when an area radiation level exceeds its maximum safe operating radiation level in more than one area, Emergency Depressurization is required.

VY PSTG Statement

N/A

VY PSTG Justification

Required safety or critical functions are environmentally qualified to withstand the radiation levels resulting from a severely damaged core. If a primary system is discharging to secondary containment, a high temperature will occur much sooner than high radiation levels and thus RPV-ED would be performed when a limiting combination is met. Per Swenson to Miller dated 6/9/86, an equipment threatening radiation environment is not considered credible because it would have to exceed post-LOCA levels and core damage is not anticipated for a high energy line break outside containment.

Basis for NRC Concern

Radiation levels above the maximum safe operating in more than one area is a symptom that there is a widespread problem which may pose a direct and immediate threat to plant equipment and to personnel both on and off

site. Reliance on actions in the temperature leg, which do not require emergency depressurization unless a limiting combination is exceeded does not assure that the personnel on and off site are protected. If a limiting temperature combination is not exceeded and there is more than one area above the maximum safe radiation operating level during an unisolated primary system discharge to the secondary containment, emergency depressurization will not be performed to minimize the release of radioactivity to secondary containment. The licensee justification does not address the threat to personnel both on and off site from radiation releases. The licensee justification is based on a high energy line break with no substantial radiological source term. The licensee is using event based information to restrict symptom based procedures.

9. EPG Statement

EPGs require a radiation release control procedure.

VY PSTG Statement

N/A

VY PSTG Justification

The radiation release control guideline has been deleted. The actions prescribed in the EPG guidelines are considered to be redundant to actions already contained within the PSTG and no credible case could be found for utilization of the radiation release guideline. Per Swenson to Miller dated 6/9/86, a large break by itself would result in rapid depressurization, a small break threatening core cooling due to inability to maintain level at high pressure would result in depressurization via the level control procedure, a small break for which cooling is maintained would not result in reaching the RPV depressurization action level.

Basis for NRC Concern

Not all scenarios for primary systems discharging outside primary and secondary containments were addressed since the licensee justification only considers a high energy line break without a significant radiological source term. The radioactive release control procedure is intended to limit radioactivity releases to areas outside of primary and secondary containments. The VY PSTG does not describe unique features which would justify deleting this emergency operating procedure. Technical specification 6.5.A.4 also requires procedures for emergency conditions involving potential or actual release of radioactivity. The licensee is using event based information to restrict symptom based procedures.

10. EPG Statement

C1-3.2 - When RPV water level drops to top of active fuel, if any system, injection subsystem or alternate injection subsystem is lined up with at least one pump running, Emergency Depressurization is required.

VY PSTG Statement

If any system, injection subsystem or alternate injection subsystem is lined up with a pump running, Emergency Depressurization is required.

VY PSTG Justification

If a system(s) is(are) lined up with pumps running, and the operator has determined he cannot maintain RPV water level above TAF, it is prudent to RPV-ED immediately to commence injection with additional sources to regain adequate core cooling.

Basis for NRC Concern

The licensee argument is not based on technical arguments but on "prudence." As long as the core is covered adequate core cooling is assured. In addition the RPV control strategy if implemented in accordance with the EPG guidelines will require the operator to begin a normal cooldown if reactor power is under control. The licensee actions are not a conservative or required action to take under all circumstances. The additional time obtained by delaying emergency depressurization until RPV water level is at the top of active fuel permits recovery actions for other sources of water to avoid an unnecessary emergency depressurization. The VY PSTG does not describe unique features which would justify adding this to the emergency operating procedure.

11. EPG Statement

N/A

VY PSTG Statement

C2-3 (RPV Emergency Depressurization) - If the MSIVs are open and the main condenser is available: Open a minimum of 3 bypass valves. If three or more bypass valves are open then proceed.

VY PSTG Justification

VY has turbine bypass capability equivalent to 105% of rated power. If RPV depressurization can be accomplished using the main condenser, the primary containment suppression system heat margin will be maintained in case it is needed at a later time.

Basis for NRC Concern

The 105% turbine bypass capability is a VY feature that should be considered in the development of the VY EOPs. The EPGs utilize the SRVs as the prime method to RPV emergency depressurize when the procedures indicate that it is required. The EPGs also indicate that, if RPV emergency depressurization is anticipated and if the bypass valves are available, the bypass valves should be used. The justification does not

address why it is acceptable to utilize the turbine bypass valves as the prime method versus the SRVs when emergency depressurization is required. There is no analysis referenced that indicates that the depressurization rate using the BPVs is equivalent to or greater than the capability of the SRVs. Using the bypass valves versus the SRVs for RPV emergency depressurization has an influence on other portions of the procedures (i.e., when establishing the minimum alternate flooding pressure). Use of the BPVs for emergency depressurization in place of the SRVs was not accounted for in the other portions of the PSTG and EOPs.

12. EPG Statement

RC/P Override - If while executing the following steps, RPV water level cannot be determined and less than 4 SRVs are open, then enter emergency depressurization.

VY PSTG Statement

N/A

VY PSTG Justification

PSTG Section RC/L presently contains an override to direct the operator to RPV flooding if water level cannot be determined.

Basis for NRC Concern

The VY PSTG does not direct the operator to enter emergency depressurization if RPV water level cannot be determined and less than 4 SRVs are opened. The RPV flooding procedure does not require the operator to enter emergency depressurization if less than 4 SRVs are opened. The PSTG actions will not allow RPV flooding to take place if emergency depressurization is not performed when RPV water level cannot be determined. The justification does not address why RPV emergency depressurization is not required.

ATTACHMENT 8

EMERGENCY PROCEDURE REFERENCE DOCUMENTS

1. Emergency Operating Procedure Inspection Report 50-271/88-200 dated August 10, 1988.

2. Letter W. Murphy to NRC, "Response to Inspection Report 50-271/88-200," dated September 27, 1988.

3. Letter W. Murphy to NRC, "Submittal of Vermont Yankee Updated Procedure Generation Package," dated March 20, 1989.

4. NRC Safety Evaluation Report for Vermont Yankee Procedure Generation Package dated June 7, 1990.

5. Inspection Report 50-271/90-16 dated December 27, 1990.

6. Vermont Yankee Procedure Generation Package, Revision 5 dated July 1990.

7. Vermont Yankee Emergency Operating Procedures Study Guide Revision 0 dated January 1991.

8. EOP Flowcharts:

   OE-3100, Scram, Revision 6
   OE-3101, RPV Control, Revision 6
   OE-3102, Alternate Level Control, Revision 7
   OE-3103, Drywell Pressure, Temperature and Hydrogen Control, Revision 7
   OE-3104, Torus Temperature and Level Control, Revision 6
   OE-3105, Secondary Containment Control, Revision 8