December 13, 2000

Mr. Michael A. Balduzzi
Vice President, Operations
Vermont Yankee Nuclear Power Corporation
185 Old Ferry Road
P. O. Box 7002
Brattleboro, VT 05302-7002

SUBJECT: VERMONT YANKEE - NRC INSPECTION REPORT 05000271/2000-008

Dear Mr. Balduzzi:

On October 30, 2000, the NRC completed a team inspection of the high pressure coolant injection and reactor core isolation cooling systems. The team also inspected your evaluations of plant changes, tests and experiments. The enclosed report presents the results of that inspection. The findings were discussed with you, Mr. D. Leach, Vice President Engineering, and other members of your staff, on October 30, 2000.

This inspection was an examination of activities conducted under your license as related to safety and compliance with the Commission's rules and regulations and with the conditions of your license. Within these areas, the inspection consisted of a selected examination of procedures and representative records, observation of activities, and interviews with personnel.

The team identified three issues that were evaluated under the risk significance determination process (SDP) and were determined to be of very low safety significance (Green). These issues have been entered in your corrective action program and are discussed in the summary of findings and in the body of the attached inspection report. These involved two examples of non-cited violations of the design control requirements of 10 CFR 50, Appendix B, Criterion III, and a non-cited violation of the design evaluation requirements of 10 CFR 50.59. If you contest these non-cited violations, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN.: Document Control Desk, Washington, D.C. 20555-0001, with copies to the Regional Administrator, Region I, the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, D.C. 20555-0001, and the NRC Resident Inspector at the Vermont Yankee Station.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/NRC/ADAMS/index.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Wayne D. Lanning, Director
Division of Reactor Safety

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

Enclosure: Vermont Yankee - NRC Inspection Report 05000271/2000-008

Attachments: (1) NRC's Revised Reactor Oversight Process
(2) Supplemental Information
(3) List of Documents Reviewed
(4) List of Acronyms Used

cc w/encl:
R. McCullough, Operating Experience Coordinator - Vermont Yankee
G. Sen, Licensing Manager, Vermont Yankee Nuclear Power Corporation
J. A. Hutton, Director-Licensing, AmerGen Energy Company
D. Rapaport, Director, Vermont Public Interest Research Group, Inc.
D. Tefft, Administrator, Bureau of Radiological Health, State of New Hampshire
Chief, Safety Unit, Office of the Attorney General, Commonwealth of Massachusetts
D. Lewis, Esquire
G. Bisbee, Esquire
J. Block, Esquire
T. Rapone, Massachusetts Executive Office of Public Safety
D. Katz, Citizens Awareness Network (CAN)
M. Daley, New England Coalition on Nuclear Pollution, Inc. (NECNP)
State of New Hampshire, SLO Designee
State of Vermont, SLO Designee
Commonwealth of Massachusetts, SLO Designee

Distribution w/encl:
Region I Docket Room (with concurrences)
B. McDermott, DRP - NRC Resident Inspector
H. Miller, RA
J. Wiggins, DRA
G. Meyer, DRP
R. Barkley, DRP
C. O'Daniell, DRP
D. Screnci, PAO
DRS File

Distribution w/encl (**VIA ADAMS**):

J. Shea, RI EDO Coordinator
E. Adensam, NRR
J. Clifford, NRR
R. Pulsifer, NRR
W. Lanning, DRS
Team Members:
A. Della Greca, DRS
F. Arner, DRS
C. Cahill, DRS
S. Peters, DRS
S. Pindale, DRS
J. Yerokun, DRS

DOCUMENT NAME: C:\Vy00-08.wpd
After declaring this document "An Official Agency Record" it **will/will not** be released to the Public.
To receive a copy of this document, indicate in the box: "**C**" = Copy without attachment/enclosure "**E**" = Copy with attachment/enclosure "**N**" = No copy

OFFICE	RI/DRS		RI/DRS		RI/DRP		RI/DRS		RI/DRS	
NAME	A DellaGreca		L Doerflein		GMeyer		J Trapp		W Lanning	
DATE	12/12/00		12/12/00		12/12/00		12/13/00		12/12/00	

OFFICIAL RECORD COPY

U. S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket Nos:	05000271
License Nos:	DPR-28
Report No.:	05000271/2000-008
Licensee:	Vermont Yankee Nuclear Power Corporation
Facility:	Vermont Yankee Nuclear Power Station
Location:	Vernon, Vermont
Dates:	October 2 - 6 and October 16 - 20, 2000
Inspectors:	A. Della Greca, Senior Reactor Engineer, Team Leader, DRS F. Arner, Reactor Engineer, DRS C. Cahill, Reactor Engineer, DRS (part time) S. Peters, Reactor Engineer (Trainee) , DRS S. Pindale, Reactor Engineer, DRS J. Yerokun, Senior Reactor Engineer, DRS B. Gupta, Engineering Contractor
Approved by:	Lawrence T. Doerflein, Chief Systems Branch Division of Reactor Safety

SUMMARY OF FINDINGS

IR 050000271-00-08 on 10/2-6, 16-20, 2000; Vermont Yankee Nuclear Power Corporation (VY); Vermont Yankee Nuclear Power Station; Mitigating Systems.

A regional-based team inspected the high pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) systems using NRC Baseline Inspection Procedure 71111.21, “Safety System Design and Performance Capability,” The team also reviewed the VY’s evaluation of changes, tests and experiments under the 10 CFR 50.59 process using NRC Baseline Inspection Procedure 71111.02, “Evaluations of Changes, Tests, or Experiments.” The significance of the issues is indicated by their color (Green, White, Yellow, or Red) and was determined by the Significance Determination Process (SDP) in Inspection Manual Chapter 609 (see Attachment 1).

A. Inspector Identified Findings

Cornerstone: Mitigating Systems

- Green. The team found that VY, in implementing the results of a calculation they had prepared to address vortex formation in the condensate storage tank (CST), raised the setpoint at which the suction source of the HPCI and RCIC pumps was swapped over from the CST to the suppression pool. This change invalidated the UFSAR statement that committed sufficient inventory in the CST to provide reactor cooling for 8 hours. Therefore, it should have resulted in a 50.59 safety evaluation to determine whether the change adversely impacted the design and licensing basis of the plant. The licensee did not prepare a safety evaluation and in the screening review process they did not provide a justification for not performing a safety evaluation. The team also determined that the 50.59 screening review associated with the CST swap-over setpoint change failed to recognize that the existing TS setpoint for swap-over was no longer conservative. Specifically, the existing TS swap-over setting of greater than or equal to 3%, corresponding to a CST nominal level of 23.36 inches above the tank bottom, was within the vortex range predicted by the vortex calculation and could potentially result in pump degradation. The failure by VY to conduct a safety evaluation for a change to the UFSAR and to change the technical specification was determined to be of very low risk significance (Green) by the SDP phase 1 screening. This conclusion was based on the suppression pool being available to support the 8-hour decay heat makeup requirements to the vessel in hot standby conditions. Therefore, no loss of function due to air entrainment from the CST would have occurred during an actual event. The failure to perform a safety evaluation was considered a non-cited violation of 10 CFR 50.59. The issue was entered in the VY corrective action program. (Section 1R21.1, Design-Mechanical, Electrical and Instrumentation and Control)

- Green. The team found that operating procedures OP-4120 and OP-4121 (HPCI and RCIC system surveillance tests) contained precautions and statements requiring assurance that CST level remained above 15% of the measured volume. The team found that the minimum level of 15% specified in the above procedures allowed the licensee to drop the CST inventory below the height of the CST standpipe by approximately 8,000 gallons and, hence, below the 75,000 gallons specified in the

UFSAR and required by the Technical Specifications. The CST standpipe was originally designed to ensure that sufficient inventory would be reserved for HPCI and RCIC operation. The team determined this issue to be of very low risk significance (Green) by the SDP phase 1 screening process. This conclusion was based upon: (1) the CST level being maintained well above the HPCI and RCIC inventory requirements; and, (2) the team having no evidence that the level had ever dropped below the CST standpipe. The failure by VY to correctly translate design basis assumptions into surveillance procedure acceptance criteria was considered a non-cited violation of 10 CFR 50, Appendix B, Criterion III, Design Control. The issue was entered in the VY corrective action program. (Section 1R21.1, Design-Mechanical, Electrical and Instrumentation and Control)

- Green. The team found that the CST level monitoring transmitters and associated sensing tubing were not heat traced and, therefore, exposed to the ambient conditions of their mounting location. The transmitters perform a safety function in that they are used for the RCIC and HPCI suction switchover from the CST to the suppression pool. The team also found that the mounting location of the transmitters was unheated and not subject to winterization controls. Therefore, the potential existed for the sensing line to freeze during prolonged winter cold spells. Freezing of the sensing line could prevent, or render erratic, the switchover of the RCIC and HPCI suction source to suppression pool. The team determined this issue to be of very low risk significance (Green) by the SDP phase 1 screening process. This conclusion was based upon: (1) the freezing of the line was potentially recognizable by the operators due to erratic CST level indications; (2) there was no evidence that the line ever froze during past winters; and (3) following an event requiring the use of the HPCI and/or RCIC systems, the licensee could manually transfer the suction source from the CST to the suppression pool. VY's failure to ensure the availability of the CST level instrumentation during all environmental conditions was considered an additional example of a non-cited violation of 10 CFR 50, Appendix B, Criterion III, Design Control. The issue was entered in the VY corrective action program. (Section 1R21.2, Operations, Maintenance and Testing)

TABLE OF CONTENTS

Report Details

1. REACTOR SAFETY

Cornerstone: Mitigating Systems

1R21 Safety System Design and Performance Capability (IP 71111.21)

Introduction

The team selected the reactor core isolation cooling (RCIC) and the high pressure coolant injection (HPCI) systems for their review of the design and performance capability of safety systems at Vermont Yankee. The two systems were selected because of their risk significance in event mitigation and core damage prevention. The primary function of the RCIC system is to provide makeup water to the reactor vessel during shutdown and isolation from the feedwater system. The primary function of the HPCI system is to prevent excessive fuel clad temperatures by providing high pressure coolant to the reactor core in the event of a small break loss of coolant accident (LOCA) that does not result in the rapid depressurization of the reactor vessel. The Inspection Procedure used for this effort was IP 71111, Attachment 21.

.1 System Design - Mechanical, Electrical, and Instrumentation and Controls

a. Inspection Scope

The team reviewed the HPCI and RCIC design and licensing basis documents, including applicable portions of the Updated Final Safety Analysis Report (UFSAR), the plant Technical Specifications (TS) and the design basis documents (DBDs). This review was performed to determine the system and component functional requirements during normal and accident conditions. The review also verified that: (1) the system design bases were in accordance with the licensing commitments and regulatory requirements; and (2) the design documents, such as drawings and design calculations were correct. The documents reviewed included engineering analyses, calculations, plant modifications, piping and instrumentation drawings (P&IDs), electrical schematics, instrumentation and control drawings, logic diagrams, and instrument set points.

For selected mechanical and electrical calculations and analyses, the team verified that the assumptions were appropriate, that proper engineering methods and models were used, and that there were adequate technical bases to support the conclusions. When appropriate, the team performed independent calculations to evaluate the document adequacy. For selected plant modifications, the team verified that the ability of the systems to perform their design functions was not adversely affected by the change.

The team also reviewed selected portions of the UFSAR, plant TS, and design documents of interfacing systems, including main steam and electrical power supplies. For these systems, the team examined piping and instrumentation drawings, electrical schematics, and configuration baseline documents and assessed the capability of the supporting systems to satisfy the design functions of HPCI and RCIC systems.

Finally, the team performed plant walkdowns of the HPCI and RCIC systems. The walkdowns were performed to verify that the physical installation was consistent with design bases document assumptions, design drawings and installation specifications. During these walkdowns the team examined the design and condition of major components, including pumps, turbines, and valves. The team also evaluated piping and pipe supports, system instrumentation, valve positions, applicable portions of AC and DC electrical switchgear, DC batteries, heat tracing, operator aids, area heating and ventilation systems, and storage of transient equipment and combustibles.

b. Findings

.1.1 Condensate Storage Tank Suction Source

The licensee prepared calculation VYC-1844, Revision 1, to determine the height of the vortex generated by the HPCI and RCIC pumps while taking suction from the condensate storage tank (CST). The objective of the evaluation was to ensure that the pumps suction source would automatically swap-over to the torus before the vortex could form and adversely impact pump operation. The team found that the process for determining the vortex height was acceptable. The licensee also prepared Calculation VYC-723, Revision 3, “Condensate Storage Tank Level (HPCI) Monitoring,” to evaluate instrument loop accuracy and used the results of this calculation to establish the automatic swap-over setpoint. Again, the team identified no concerns with the process used for determining loop accuracy. However, the team determined that the licensee’s use of the results of these calculations resulted in a conflict with the licensing basis operating requirement specified in the UFSAR and in the Technical Specifications.

UFSAR section 11.8.3 stated that, “... the operating requirement of the CST is a volume equal to an eight hour supply for makeup to the reactor in hot standby via either the reactor core isolation cooling (RCIC) system or the high pressure coolant injection (HPCI) system.” The UFSAR also stated, “... so that the eight-hour supply indicated above will always be available, all other suction pipes are terminated at a level above that required for the eight-hour supply.” The team determined that, based on these statements, the volume between the swap-over setpoint and the suction level of the “other suction pipes” should contain sufficient water to provide makeup coolant to the reactor for a minimum of eight hours. General Electric (GE) originally calculated this volume equivalent to be 65,000 gallons, but recommended that a value of 75,000 gallons be used. Accordingly, 75,000 gallons were specified in the UFSAR and TS. However, the licensee recently revised the required volume equivalent. The change was based on the results of a calculation contained in memorandum VYS-114/97 in which the licensee determined that 56,571 gallons would be sufficient to provide eight hours of makeup water to the vessel for decay heat removal.

Using the results of the vortex and loop accuracy calculations specified above, the licensee set the swap-over point of the RCIC and HPCI pump suction at 33.45 inches above the tank bottom. To ensure that a dedicated volume of water would be available for HPCI and RCIC operation, the standpipe for the core spray and condensate transfer systems were designed to take suction from the CST at a level of 74.66 inches from the bottom of the tank. Subtracting the level of the swap-over setpoint from the termination point of the other suction pipes, the team determined that the CST volume available to the HPCI and RCIC pumps, prior to swapping over to the torus, had a nominal value of approximately 50,441 gallons, which was less than the required values calculated by both GE and the licensee. The available inventory could be even less when instrument accuracy was considered. The team concluded that this reduction in available and usable CST volume conflicted with the existing UFSAR operating requirement of an available eight-hour supply and should have required a 50.59 safety evaluation to determine whether the change impacted the design and licensing basis of the plant. The licensee did not prepare a safety evaluation and did not provide a justification for not performing a safety evaluation in the screening review process.

The team also determined that the 50.59 screening review associated with the CST swap-over setpoint change failed to recognize that the existing TS setpoint for swap-over was no longer conservative. Specifically, the existing TS swap-over setting of greater than or equal to 3% corresponds to a CST nominal level of 23.36 inches above the tank bottom. This level, when considering instrument uncertainty, was well within the vortex range predicted by calculation VYC-1844, Revision 1, for both pumps and, therefore, would not prevent potential pump degradation due to air ingestion from vortexing.

The team determined that the licensee's failure to conduct a safety evaluation for a change to the UFSAR and to change the technical specification was more than minor. The team also determined that this issue was a violation of 10 CFR 50.59. The result of this 10 CFR 50.59 violation was assessed through the significance determination process and was found to be of very low significance (green) because the torus was available to support the 8-hour decay heat makeup requirements to the vessel in hot standby conditions. Therefore, no loss of function due to air entrainment from the CST would have occurred during an actual event. Due to the overall low risk significance, this violation of 10 CFR 50.59 was categorized at Severity Level IV and was treated as a non-cited violation (NCV) consistent with Section VI.A of the NRC Enforcement Policy, issued on May 1, 2000 (65FR25368). **(NCV 05000271/2000-008-01)**

At the conclusion of the inspection, the licensee initiated memorandum VY/LIC 00-36 in which they acknowledged that the current technical specification value was low relative to the swap-over setting required to support HPCI and RCIC operability. Since the actual swap-over level was raised based on the vortex calculation, HPCI and RCIC remained operable. The licensee initiated Event Reports (ERs) No. 2000-1578 and 2000-1596 to enter the issue in the corrective action process. The licensee indicated they planned to submit a license amendment to correct the technical specifications.

.1.2 Surveillance Procedures

Surveillance procedures OP-4120 and OP-4121 (HPCI and RCIC system surveillance tests) contained precautions and statements to ensure that CST level stayed above 15% or termination of the test was required. The team found that the minimum level of 15% specified in the above procedures allowed the licensee to drop the CST inventory below the height of the CST standpipe (see previous section) by approximately 5,500 gallons (8,000 gallons when loop uncertainty was considered). The team also determined that, if the licensee allowed the level to approach the 15% limit in the procedures, the volume reserved for HPCI and RCIC would be approximately 71,700 gallons and, hence, less than the 75,000 gallons required by the technical specifications. The team found this to be a violation of 10 CFR 50, Appendix B, Criterion III, Design Control.

The result of this violation was assessed through the significance determination process and was found to be of very low significance (green) because: (1) the CST level was maintained well above the HPCI and RCIC inventory requirements and (2) the team had no evidence that the level had ever dropped below the CST standpipe. Due to the overall low risk significance, this violation of 10 CFR 50, Appendix B, was treated as a non-cited violation (NCV) consistent with Section VI.A of the NRC Enforcement Policy, issued on May 1, 2000 (65FR25368). **(NCV 05000271/2000-008-02)**

The licensee initiated ER 2000-1509 to enter the issue in the corrective action process and ensure that the procedural guidelines are consistent with the technical specification requirement.

.2 Operations, Maintenance, and Testing

a. Inspection Scope

The team reviewed various documents and plant procedures to verify that the RCIC and HPCI systems were installed, operated, and maintained consistent with the design and licensing bases. The operational readiness and material condition of the selected systems were assessed by conducting system walkdowns and by reviewing appropriate documents, including operating procedures, operator logs, component maintenance history records, preventive maintenance records, surveillance test procedures and results, calibration records, and system health reports. The review also included applicable portions of the emergency operating procedures, vendor documents, portions of the design-bases documents that discussed the operation and maintenance of the systems, and selected event reports related to corrective maintenance and operation of the systems.

The team also interviewed responsible Vermont Yankee personnel, including licensed and non-licensed operators, the system engineer, and maintenance and instrumentation and control personnel, regarding the operation and performance of the selected systems and components.

b. Findings

During the walkdown of condensate storage tank (CST) area, the team observed that the CST level instrumentation had not been freeze-protected. Specifically, the team found that level transmitters LT-5A and B (for HPCI) and LT-12A and B (for RCIC) were mounted in an unheated room on unheated racks and that neither the transmitters nor the sensing tubing had not been heat traced like the other equipment in the area. The transmitters perform a safety function in that they are used for CST level indication and for the RCIC and HPCI suction switchover from the CST to the suppression pool.

The team also found that the CST area (room) was unheated and that the only heat source was the CST itself which was maintained at greater than 50°F, as verified by periodic operator rounds (TI-107-3) per "Auxiliary Operator Round Sheet Outside," VYOPF 0150.01, OP 0150, Revision 33 Page 8 of 11. However, because the CST was insulated, the team concluded that heat radiated by the CST would have only limited effect on the room ambient temperature. The team also concluded that the potential existed for the sensing line to freeze during prolonged winter cold spells. Freezing of the sensing line could prevent, or render erratic, the switchover of the RCIC and HPCI suction source to suppression pool.

The team evaluated this issue through the significance determination process and found it to be of very low risk significance (green) because: (1) the freezing of the line was potentially recognizable by the operators by erratic CST level indications; (2) the team had no evidence that the line ever froze during past winters; and (3) following an event requiring the use of the HPCI and/or RCIC systems, the licensee could manually transfer the suction source from the CST to the suppression pool. The licensee's failure to ensure the availability of the CST level instrumentation during all environmental conditions was considered to be an additional example of a non-cited violation of 10CFR50, Appendix B, Criterion III, Design Control. **(NCV 05000271/2000-008-02)**

The licensee issued ER 2000-1575 to enter the issue in the corrective action process and to ensure that appropriate steps are taken in the future to prevent freezing of the CST level instrumentation.

1R02 <u>Evaluations of Changes, Tests, or Experiments</u> (IP71111.02)

a. <u>Inspection Scope</u>

The team reviewed selected safety evaluations (SEs) performed by Vermont Yankee. The SEs were selected from a list of changes relating to the HPCI and RCIC systems and from other plant changes implemented during the last year. The selection took into consideration safety significance of the change, risk to the structures, systems, and components affected, and impact on the three reactor safety cornerstones (initiating events, mitigating systems, and barrier integrity).

The review was conducted to verify that changes to the facility or procedures as described in the Updated Final Safety Analysis Report (UFSAR), and test and experiments not described in the UFSAR, were reviewed and documented by the licensee in accordance with 10 CFR 50.59. The review also verified that any safety issues pertinent to the changes, tests and experiments had been properly resolved. Additionally, the team verified the licensee's conclusions that the changes tests and experiments did not require prior NRC approval or a license amendment were appropriate. The team conducted discussions with cognizant engineering, as required, and evaluated supporting technical information, including calculations, analyses, and design requirements.

The inspectors also reviewed a sample of changes, tests and experiments for which VY determined that a safety evaluation was not required. This review was performed to verify that VY's threshold for performing safety evaluations was consistent with the requirements of 10 CFR 50.59. Lastly, the team verified that problems associated with the implementation of the safety evaluation program were entered into the corrective action program.

b. Findings

Except as indicated in Section 1R21.1, no other findings were identified.

4. OTHER ACTIVITIES (OA)

4OA1 Identification and Resolution of Problems (IP 71152)

a. Inspection Scope

The team reviewed the licensee's effectiveness in identifying problems associated with the high pressure coolant injection and reactor core isolation systems. The team also reviewed a sample of event reports related to the selected systems to evaluate the adequacy and timeliness of the corrective actions resulting from the identified problems. For selected event reports the team reviewed the adequacy of the operability determinations and verified the completion of the corrective actions.

b. Findings

There were no findings identified.

4OA6 Meetings, Including Exit

.1 Management Meeting

The team presented the inspection results to Messrs. M. Balduzzi, Vice President of Operations and D. Leach, Vice President of Engineering, and other members of licensee management at the conclusion on the inspection on October 30, 2000. The licensee acknowledged the inspection findings presented.

(1) NRC's REVISED REACTOR OVERSIGHT PROCESS

The federal Nuclear Regulatory Commission (NRC) recently revamped its inspection, assessment, and enforcement programs for commercial nuclear power plants. The new process takes into account improvements in the performance of the nuclear industry over the past 25 years and improved approaches of inspecting and assessing safety performance at NRC licensed plants.

The new process monitors licensee performance in three broad areas (called strategic performance areas): reactor safety (avoiding accidents and reducing the consequences of accidents if they occur), radiation safety (protecting plant employees and the public during routine operations), and safeguards (protecting the plant against sabotage or other security threats). The process focuses on licensee performance within each of seven cornerstones of safety in the three areas:

Reactor Safety

- Initiating Events
- Mitigating Systems
- Barrier Integrity
- Emergency Preparedness

Radiation Safety

- Occupational
- Public

Safeguards

- Physical Protection

To monitor these seven cornerstones of safety, the NRC uses two processes that generate information about the safety significance of plant operations: inspections and performance indicators. Inspection findings will be evaluated according to their potential significance for safety, using the Significance Determination Process, and assigned colors of GREEN, WHITE, YELLOW or RED. GREEN findings are indicative of issues that, while they may not be desirable, represent very low safety significance. WHITE findings indicate issues that are of low to moderate safety significance. YELLOW findings are issues that are of substantial safety significance. RED findings represent issues that are of high safety significance with a significant reduction in safety margin.

Performance indicator data will be compared to established criteria for measuring licensee performance in terms of potential safety. Based on prescribed thresholds, the indicators will be classified by color representing varying levels of performance and incremental degradation in safety: GREEN, WHITE, YELLOW, and RED. GREEN indicators represent performance at a level requiring no additional NRC oversight beyond the baseline inspections. WHITE corresponds to performance that may result in increased NRC oversight. YELLOW represents performance that minimally reduces safety margin and requires even more NRC oversight. And RED indicates performance that represents a significant reduction in safety margin but still provides adequate protection to public health and safety.

The assessment process integrates performance indicators and inspection so the agency can reach objective conclusions regarding overall plant performance. The agency will use an Action Matrix to determine in a systematic, predictable manner which regulatory actions should be taken based on a licensee's performance. The NRC's actions in response to the significance (as represented by the color) of issues will be the same for performance indicators as for inspection findings. As a licensee's safety performance degrades, the NRC will take more and increasingly significant action, which can include shutting down a plant, as described in the Action Matrix.
More information can be found at: http://www.nrc.gov/NRR/OVERSIGHT/index.html.

(2) SUPPLEMENTAL INFORMATION

PARTIAL LIST OF PERSONS CONTACTED

Vermont Yankee (VY)

M. Balduzzi	Vice President Operations
M. Ball	Senior Design Engineer
K. Bronson	Plant Manager
P. Corbett	Superintendent Engineering Support
J. DeVincentis	Senior Licensing Engineer
J. Callaghan	Lead Discipline Engineer, Fluid Systems Engineering
M. Desigts	Superintendent Technical Support
J. Dreyfuss	Superintendent System Engineering
S. Goodwin	Lead Discipline Engineer, Mechanical Structural Engineering
K. Hemson	System Engineer
J. Hoffman	Superintendent Design Engineering
R. January	Lead Discipline Engineer, Electrical Engineering
D. Leach	Vice President Engineering
G. Sen	Licensing Manager
M. Smaga	Supervisor Design Engineering
R. Wanckyk	Director Safety and Regulatory Affairs
G. Wierzbowski	Lead System Engineer

Nuclear Regulatory Commission (NRC)

L. Doerflein	Chief, DRS Systems Branch
B. McDermott	Senior Resident Inspector

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened/Closed

05000271/2000-008-01	NCV	Inadequate Safety Evaluation
05000271/2000-008-01	NCV	Inadequate Design Control

(3) LIST OF DOCUMENTS REVIEWED

Design Drawings

B-191301 (Sh. 865-866)	ECCS Analog Trip Division II, Sh 1 and 2, Control Wiring Diagrams
B-191301 (Sh. 1179-1199A)	RCIC System and Components Control Wiring Diagrams
B-191301 (Sh. 1229A)	Torus & CST Level Indication Control Wiring Diagrams
B-191301 (Sh. 1433-1456)	HPCI System and Components Control Wiring Diagrams
B-191301 (Sh. 1457)	Seal Condenser Wiring Shutoff Control Wiring Diagram
B-191301 (Sh. 1177)	RCIC Alternate Shutdown System Instrumentation Logic Diagram
B-191301 (Sh. 866)	ECCS Analog Trip Division II Control Wiring Diagram
G-191169 (Sh. 1-2)	High Pressure Coolant Injection System Flow Diagram
G-191174 (Sh. 1-2)	Reactor Core Isolation Cooling System Flow Diagram
G-191176 (Sh. 1-2)	Condensate Demineralized Water System Flow Diagram
G-191299 (R24)	Analog Tri System - 24 V DC One Line Diagram
G-191299 (R25)	4KV Auxiliary One Line Diagram
G-191300 (Sh. 1-2)	480V Auxiliary One Line Wiring Diagram
G-191301 (Sh. 1-2)	480V Auxiliary One Line Wiring Diagram
G-191372 (Sh. 1-3)	125VDC One Line Wiring Diagram
G-191372 (Sh. 4)	120/240 V Vital AC and Instrument AC One Line Wiring Diagram
G-191372 (Sh. 5)	24V DC Neutr. Mon. & 120V AC RPS One Line
NY 706102 5920-25 (Sh. 1)	Reactor Core Isolation Cooling System Flow Control Diagram
NY 706102 5920-26 (Sh. 2)	Reactor Core Isolation Cooling System Flow Control Diagram
NY 706102 5920-490 (Sh. 3)	Reactor Core Isolation Cooling System Flow Control Diagram
NY 706102 5920-38 (Sh. 1)	High Pressure Core Injection System Flow Control Diagram
NY 706102 5920-39 (Sh. 2)	High Pressure Core Injection System Flow Control Diagram

Engineering Calculations

VYS114/97	ITS Open Item 3.5-06 - Basis for 75000 Gallons CST Requirements
VYC-298 (R12)	Battery Sizing Calculation for Vermont Yankee 125 V Station Batteries A-1 and B-1; Capacity Verification for Battery Charger BC-1-1A, BC-1-1B
VYC 415,	Appendix R, RCIC, HPCI and ECCS Room Cooling
VYC-462C (R0)	RCIC Steam Line Areas High Temperature Setpoint (w/CCN 1)
VYC-488 (R3)	HPCI Stm. Ln. Low Pressure Trip Loop Accuracy (w/CCN 1)
VYC-687C (R0)	HPCI Discharge Flow Hi/Low Trip Ind. Switch Accuracy
VYC-687D (R0)	RCIC Discharge Flow Hi/Low Trip Ind. Switch Accuracy
VYC-687E (R4)	HPCI Stm. Ln. Break DP Ind. Switch Setpoint Accuracy (w/CCN 1)
VYC-704 (R1)	HPCI System Control and Indication Loop Accuracy
VYC-706 (R1)	Condensate Storage Tank Level (RCIC) Monitoring
VYC-709 (R1)	RCIC System Flow Control and Indication Loop Accuracy
VYC-714J (R1)	RCIC Stm. Ln. Break Logic Circuit Time Delay Relay
VYC-714K (R0)	RCIC Stm. Tunnel Hi Temp Logic Circuit Time Delay Relay
VYC-714M (R1)	HPCI Stm. Tunnel Hi Temp Logic Circuit Time Delay Relay
VYC-723 (R3)	Condensate Storage Tank Level (HPCI) Monitoring
VYC-830 (R9)	Voltage Drop Calculation For Distribution Panels DC-1 and DC-2
VYC-886 (R1)	Station Blackout Documentation Analysis
VYC-1188 (R1)	Coordination, 125 VDC, Circuit Breaker, Appendix R (w/CCN 1)
VYC-1258	Design differential pressures for HPCI MOVs

LIST OF DOCUMENTS REVIEWED (Cont.)

VYC-1349 (R1)	125 DC Control Circuit Voltage Drop Study - Batteries A1 & B1
VYC-1726	Reactor Core Isolation Cooling Pump Test Acceptance Values
VYC-1738	HPCI Pump Test Acceptance Criteria
VYC-1790	HPCI/RCIC Turbine Exhaust Vacuum Breaker Sizing Calculations
VYC-1844(R1)	HPCI and RCIC Vortex Height
VYE-1064	RCIC System Hydraulic Calculations
VYPC98-008	Component Level Review of HPCI MOVs for GL 89-10
VYS 77/96	Basis for RCIC Valve Stroke Times

<u>Design Bases Documents</u>

HPCI 99-019 ICI	High Pressure Coolant Injection System, Revision 0
RCIC 2000-009 IC2	Reactor Core Isolation Coolant System, Revision 0
VYS-027 Rev. 7	Separation Criteria
---	Instrument Uncertainty and Setpoints Design Guide (Revision 1)

<u>Engineering and Design Change Requests</u>

95-407	MOV Design Improvements Including Appendix R Hot Short (w/ECN4&5)
96-401	Appendix R - Alternate Shutdown System Redundant Fuses (w/ECN2)
98-402	HPCI and RCIC Turbine Exhaust Vacuum Breaker Modification
97-404	MOV Electrical and Pressure Locking Modification (w/ECN1)
EE 1277	"Replace Obsolete Square D Model 9036-GG1-S2 Level Switch with Square D Model 9036-GG2"
TM-99-007	Replacing Blind Flanges on Reactor Building with Penetration Flange
MM-98-068	Isolation of ES-13-92 and ES-13-100 From SCE Components
MM-2000-12	RCIC Valve V13-16 Packing Modification

<u>Event Reports (ER)</u>

96-0446	Parts of motor inspection per OP5235 were omitted incorrectly
96-0618	MOV V23-20 closed out of control
96-0704	Wrong wire termination point for RCIC Valve V13-15
97-0140	At Low Reactor Pressure HPCI Flow not tested
98-0313	Discrepancy between TS and FSAR for HPCI High Flow Steam Limit
98-1171	Sludge found in RCIC suction line
98-1200	Sludge found in HPCI suction line
98-1274	Unsealed Floor Penetrations in Main Steam Tunnel
98-1792	Incorrect Safety Classification of component Bill of Material Item
98-2052	Discrepancies in design documents
98-2066	IE Bulletin 88-04
98-2108	HPCI Isolated due to Incorrect Inoperable Isolation Valve
98-2132	Valve V23-16 Torque Switch Failure
99-0173	RCIC MOV Calculation non-conservative value for Reactor Pressure
99-0175	RCIC flow element flow curve
99-0497	Failure of PS-23-68C during Surveillance
99-0538	Failure to Record HPCI Pump Discharge Valve Closing Time
99-0607	HPCI and RCIC Surveillance Testing
99-0919	Inadequate Procedural Guidance for Throttle Valve Position

99-1047	Failure of HPCI System to Meet Required Time to Rated Flow
2000-0349	HPCI Exhaust Drain Pot Alarm
2000-0678	Improper Operation of HPCI Overspeed Trip Tappet
2000-0678	Logging Discrepancies - Equipment out of Service Times
2000-0723	GE SIL potential overpressurization of system piping
2000-1262	Incorrect Flow Indicator Referenced by Procedure
2000-1586	Potential Water Hammer in HPCI Discharge Piping

Station Procedures

AP-0020	Control of Temporary and Minor Modification
AP-0022	Setpoint Change Request
AP-6002	Preparing 50.59 Evaluations
AP-6008	Vermont Yankee Design Change
EOP-1	RPV Control
EOP-3	Primary Containment Control
ESP-01-013	10 CFR 50.59 A1 Screening Evaluations
ESP-02-013	10 CFR 50.59 A2 Safety Evaluations
ESP-03-013	10 CFR 50.59 Refresher
OP 0150	Conduct of Operations and Operator Rounds
OP 3126	Shutdown Using Alternate Shutdown Methods
OP 2120	HPCI System
OP 2121	RCIC System
OP 3147	Loss of RBCCW
OP 4120	HPCI System Surveillance
OP 4121	RCIC System Surveillance
PP-7007	Setpoint Control Program
Miscellaneous	Emergency Operating Procedures
Miscellaneous	Calibration Procedures and Results

Safety Evaluations

99-008	Substitution of Monitoring for RBCCW IST Pump/Check Valve Testing
99-020	Using Uchida Correlation to Calculate EQ Profiles for High Energy Line Breaks In Vermont Yankee Reactor Building
99-032	STP 99-003 - Increased Core Flow
99-042	BMO 97-52, Revision 1, Closure - Potential Alternate Cooling System Inventory Loss
2000-001	OP 4033, Revision 0, Standby Fuel Pool Cooling System Heat Exchanger Thermal Performance Test
2000-004	VYDC 99-006 - DE&S Main Steam Break at LO Power
2000-010	FCR 16/305 - Clarifies Statement in FSAR Regarding Wiring Separation in Cable Vault

LIST OF DOCUMENTS REVIEWED (Cont.)

2000-013	FCR 16/259 - Revision to FSAR Section 7.4.5 “Change in Method of Testing ESF Instrumentation Channels”
2000-014	FCR 16/297 - Revision to FSAR Figure 7.3-6 Replacement of Chromalox Analog Temperature Unit”

50.59 Screens

SCR 99C-002	Setpoint Change Request - LT-107-12A&B(M) - RCIC CST Low Level
SCR 99C-068	Setpoint Change Request - LSL-107-5a and LSL-107-5B - CST Level Lo Alarm and Auto Suction Transfer
99-032	Minor Mod HPCI Gland Seal Exhauster/Condenser Seismic Mod
99C-034	RCIC Pump Discharge Flow Diff. Press. Indicating Switch
2000C-001	Cooling Tower Influent (Circ. Water) Hi Rad Monitor
2000C-005	PS-24-1A and 1B
2000C-006	LT-2-3-73A(M)
2000C-007	LT-2-3-73B(M)

Miscellaneous

VY-98-27	Audit Report - Vermont Yankee Design Basis Document Validation
98-157	Quality Assurance Surveillance Report - VY Engineering Calculations
2000-039	Quality Assurance Activity Assessment Report - Engineering Calculations
2000-072	Quality Assurance Activity Assessment Report - 10 CFR 50.59 Safety Evaluations

(4) LIST OF ACRONYMS USED

AC or ac	Alternating Current
CFR	Code of Federal Regulation
CST	Condensate Storage Tank
DBD	Design Basis Document
DC or dc	Direct Current
ER	Event Report
GE	General Electric
HPCI	High Pressure Coolant Injection
LOCA	Loss of Coolant Accident
NCV	Non-Cited Violation
P&ID	Piping and Instrumentation Diagram
NRC	Nuclear Regulatory Commission
RCIC	Reactor Core Isolation Cooling
SDP	Significancy Determination Process
SE	Safety Evaluation
TS	Technical Specifications
UFSAR	Updated Final Safety Analysis Report
VY	Vermont Yankee