June 4, 2007

Mr. Theodore A. Sullivan
Site Vice President
Entergy Nuclear Operations, Inc.
Vermont Yankee Nuclear Power Station
P.O. Box 0500
185 Old Ferry Road
Brattleboro, VT 05302-0500

SUBJECT: VERMONT YANKEE NUCLEAR POWER STATION - NRC LICENSE RENEWAL INSPECTION REPORT 05000271/2007006

Dear Mr. Sullivan:

On May 24, 2007, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection of your application for a renewed license of your Vermont Yankee Nuclear Power Station. The enclosed report documents the result of the inspection which was discussed with members of your staff on May 24, 2007, at a publicly observed exit meeting conducted at the Latchis Theater in Brattleboro, VY.

The purpose of this inspection was to examine the plant activities and documents that supported the application for a renewed license of the Vermont Yankee Nuclear Power Station. The inspection reviewed the screening and scoping of non-safety related systems, structures, and components, as required in 10 CFR 54.4(a)(2), and determined whether the proposed aging management programs are capable of reasonably managing the effects of aging. These NRC inspection activities constitute one of several inputs into the NRC review process for license renewal applications.

The inspection team concluded screening and scoping of nonsafety-related systems, structures, and components, were implemented as required in 10 CFR 54.4(a)(2), and the aging management portions of the license renewal activities were conducted as described in the License Renewal Application. The inspection results supported a conclusion that the proposed activities will reasonably manage the effects of aging in the systems, structures, and components identified in your application. The inspection concluded the documentation supporting the application was in an auditable and retrievable form.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response (if any) will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records component of the NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA by Marsha K. Gamberoni for/

Richard J. Conte, Chief
Engineering Branch 1

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

Enclosure: Inspection Report 05000271/2007006
w/Attachment: Supplemental Information

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U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No.:	50-271
Licensee No.:	DPR-28
Report No:	05000271/2007006
Licensee:	Entergy Nuclear Operations, Inc.
Facility:	Vermont Yankee Nuclear Power Station
Location:	320 Governor Hunt Road Vernon, Vermont 05354-9766
Dates:	January 22 - 26 and February 5 -9, 2007
Inspectors:	Michael Modes, Team Leader Glenn Meyer, Sr. Reactor Engineer Timothy O'Hara, Reactor Engineer Jennifer Bobiak, Reactor Engineer Anne DeFrancisco, Reactor Engineer Douglas Tifft, Reactor Engineer Suresh Chaudhary, Reactor Engineer
Approved by:	Richard J. Conte, Chief Engineering Branch 1

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SUMMARY OF FINDINGS

IR 05000271/2007006; 01/22/2007 - 01/26/2007, 02/05/2007 - 02/09/2007; Vermont Yankee Nuclear Station; Inspection of the Scoping of Nonsafety Systems and the Proposed Aging Management Procedures for the Vermont Yankee Nuclear Station's Application for Renewed License.

This inspection of license renewal activities was performed by seven regional office engineering inspectors. The inspection was conducted in accordance with NRC Manual Chapter 2516 and NRC Inspection Procedure 71002. This inspection did not identify any "findings" as defined in NRC Manual Chapter 0612. Based on the inspection, the applicant made a number of enhancements to the programs, which were documented either via amendments to the license renewal application or through commitments. The inspection team concluded screening and scoping of nonsafety-related systems, structures, and components, were appropriately implemented, and the aging management portions of the license renewal activities were conducted as described in the License Renewal Application. The inspection results supported a conclusion that the proposed activities will reasonably manage the effects of aging in the systems, structures, and components identified in your application. The inspection concluded the documentation supporting the application was in an auditable and retrievable form.

Report Details

4. OTHER ACTIVITIES (OA)

4OA2 Other - License Renewal

a. Inspection Scope

This inspection was conducted by NRC Region I based inspectors, in conformance with the requirements and guidance contained in NRC Inspection Procedure 71002 “License Renewal Inspection,” and Manual Chapter 2516 “Policy and Guidance for the License Renewal Inspection Programs,” in order to ensure there is reasonable assurance that the effects of aging will be managed consistent with the current licensing basis during the period of extended operation. This inspection provides a basis for recommending issuance or denial of a renewed license by evaluating the thoroughness and accuracy of the screening and scoping of nonsafety-related systems, structures, and components, as required in 10 CFR 54.4(a)(2), and whether aging management programs will be capable of managing the identified aging effect in an appropriate manner.

In order to accomplish the goals above the inspection team selected a number of systems for review, using the NRC accepted guidance, in order to determine if the methodology applied by the applicant appropriately captured the nonsafety-systems affecting the safety functions of a system, component, or structure within the scope of license renewal.

The inspection team selected a sample of aging management programs to verify the adequacy of the applicant’s documentation and implementation activities. The selected aging management programs were reviewed to determine whether the proposed aging management implementing process would adequately manage the effects of aging on the system. The inspection team evaluated site-specific information such as surveillance test results, preventative maintenance records, corrective maintenance records. It also reviewed industry operational experience such as generic communications, vendor notifications, etc.

The inspectors reviewed supporting documentation and interviewed applicant personnel to confirm the accuracy of the license renewal application conclusions. For a sample of plant systems and structures, inspectors performed visual examinations of accessible portions of the systems to observe aging effects.

The inspectors also conducted an assist to the Office of Nuclear Regulation related to scoping and screening. The review is documented on Page A13.

b.1. Scoping of Nonsafety-Related Systems, Structures, and Components

The inspectors reviewed the applicant’s program guidance procedures and scoping results. The inspectors determined the applicant’s procedures to be consistent with the NRC accepted guidance in NEI 95-10, Appendix F, Revision 5, Section 3 (nonsafety-

related systems, structures, and components within scope of the current licensing basis), 4 (nonsafety-related systems, structures, and components directly connected to safety-related systems, structures, and components), and 5 (nonsafety-related systems, structures, and components not directly connected to safety-related systems, structures, and components). Also, the inspectors determined that the applicant appropriately utilized the guidance in their process for determining which systems were within scope.

The inspectors reviewed the set of license renewal drawings, which had been color-coded to indicate systems and components in scope for 10 CFR 54.4.(a)(1) and (a)(3). The inspectors interviewed personnel, reviewed license renewal program documents, and independently inspected numerous areas within the plant to confirm that appropriate systems, structures, and components had been included within the license renewal scope; that systems, structures, and components excluded from the license renewal scope had an acceptable basis; and that the boundary for determining license renewal scope within the systems, including seismic supports and anchors, was appropriate.
Cooling Tower #2 Cell 1
The in-plant areas and systems reviewed included the following:

- Reactor Building,
- Turbine Building,
- Intake Structure,
- Control Room Building,
- Cooling Tower #2 Cell 1,
- Condensate Storage Tank Valve and Instrument Enclosure,
- Fuel Oil Storage Tank and Pump House,
- Diesel Generator Rooms,
- High Pressure Coolant Injection (HPCI) System,
- Turbine Building Closed Cooling Water (TBCCW) System,
- Standby Liquid Control (SLC) System,
- Standby Gas Treatment System,
- Heating, Ventilation, and Air Conditioning System,
- Compressed Air/Instrument Air System, and
- Service Water System.

For systems, structures, and components selected regarding spatial interaction (leakage of nonsafety-related components adversely affecting safety-related components), the inspectors identified scoping issues within the Turbine Building. Entergy's scoping had determined that most of the Turbine Building was not within license renewal scope with a few exceptions, i.e., the diesel generator rooms, a few limited areas, and segments of the service water and diesel fuel oil systems. The inspector determined that the scoping of segments of the service water and diesel fuel oil systems were not, in some instances, in accordance with guidance and that safety-related cables for reactor protection system functions had not been appropriately considered. To address these issues, Entergy placed fluid system components within the Turbine Building within the license renewal scope.

For systems, structures, and components selected regarding structural interaction (seismic design of safety-related components dependent upon nonsafety-related components), the inspectors determined that structural boundaries had been accurately determined but not categorized within the license renewal program documents. Specifically, the inspectors determined that Entergy had reviewed applicable isometric drawings to determine the seismic design boundaries and had correctly included the applicable component types in the license renewal application, based on an independent review of a sample of the isometric drawings and the seismic boundary determinations. However, the determined boundaries had been recorded in personal papers but not in license renewal program documents. Accordingly, Entergy personnel subsequently implementing aging management programs would not have the benefit of these determinations and would have needed to redetermine the boundaries, potentially incorrectly. To address this concern, Entergy revised their plans to include the existing seismic boundary determinations within licensee renewal program documents.

Subsequently, the inspectors reviewed a later revision of the program procedure covering scoping of nonsafety-related equipment, and found that the above issues had been addressed. Specifically, the fluid system components within the Turbine Building had been included within the license renewal scope and an attachment had been added to delineate specific structural boundaries. As revised, the inspectors concluded that Entergy had implemented an acceptable method of scoping and screening of nonsafety-related systems, structures, and components and that this method resulted in accurate scoping determinations.

b.2. Aging Management Programs (AMPs)

Buried Piping Inspection

The Buried Piping Program is an existing program credited with managing the effects of corrosion on the pressure-retaining capability of buried steel piping components in the Vermont Yankee Nuclear Station. The aging effects are managed by performance of an inspection either via opportunistic inspection (i.e., during an excavation for modification or repairs) or a focused inspection. The inspection must be performed within the first 10 years of the period of extended operation. Vermont Yankee must also have a record of previous examinations performed.

The inspector selected recent examples of buried piping inspections to sample the effectiveness of the Buried Piping Program. Specifically, the inspector reviewed records from a 2003 event that uncovered a section of service water piping, 24" SW-1B, and a recent containment access building replacement engineering request, to examine the interface between Design Engineering and site excavation teams. In addition, the inspector assessed Entergy's knowledge of the ground chemistry conditions around the buried piping service water areas. The inspector noted that the Entergy procedure describing inspection of buildings and structures did not include specific information as to scheduling a buried piping "focused" inspection. Entergy

subsequently added a planned amendment to their application to add the Generic Aging Lessons Learned requirement of performing a focused inspection within the first 10 years of the period of extended operation (Commitment 44, BVY 07-018).

The inspectors also verified that the applicant performed adequate historic reviews of plant specific and industry experience to determine aging effects. The applicant provided adequate guidance to ensure aging effects are appropriately managed. The inspectors concluded the material condition of the systems, structures and components that require an aging management review, will be adequately maintained consistent with the rule and the applicant's license renewal program.

BWR Stress Corrosion Cracking

The Stress Corrosion Cracking Program is an existing program credited with managing the effects of intergranular stress corrosion cracking in reactor coolant pressure boundary components in the Vermont Yankee Nuclear Station. The aging effects are managed by inspection and flaw evaluation to monitor intergranular stress corrosion and its effects.

The inspectors selected specific components to sample the effectiveness of the Stress Corrosion Cracking Program such as the feedwater nozzles to safe end welds. The inspector selected these components based on the specific characteristics of the welds. The inspector concluded that Entergy had an effective Intergranular Stress Corrosion Cracking Program and that the Generic Letter 88-01, NRC Position on Intergranular Stress Corrosion in BWR Austenitic Stainless Steel Piping, requirements are applied.

The inspectors also verified that the applicant performed adequate historic reviews of plant specific and industry experience to determine aging effects. The applicant provided adequate guidance to ensure aging effects are appropriately managed. The inspectors concluded the material condition of the systems, structures and components that require an aging management review will be adequately maintained consistent with the rule and the applicant's license renewal program.

Boiling Water Reactor Vessel Internals

The Boiling Water Reactor Vessel Internals Program is an existing program that includes inspection, flaw evaluation and repair in conformance with the applicable staff-approved industry Boiling Water Reactor Vessel and Internals Project documents, and monitoring and control of reactor water chemistry, to ensure the long-term integrity of the vessel internal components in the Vermont Yankee Nuclear Station.

The inspector selected specific components to sample the effectiveness of the Boiling Water Reactor Vessel Internals Program. These included the steam dryer and the core shroud components. The inspector chose these components based on the Boiling Water Reactor Owners response to industry and regulatory concerns on shroud cracking, and the fact that the steam dryer, while not being safety-related, is a loose parts concern and was carefully examined as part of the extended power uprate

application process in 2005. The inspector reviewed Entergy's Boiling Water Reactor Vessel Internals Program and confirmed that applicable Electric Power Research Institute recommendations are applied in determining inspection frequency and methods. The empirical data used to back up the computer modeling on steam dryer characteristics was also reviewed by the inspector.

The inspectors also verified that the applicant performed adequate historic reviews of plant specific and industry experience to determine aging effects. The applicant provided adequate guidance to ensure aging effects are appropriately managed. The inspectors concluded the material condition of the systems, structures and components that require an aging management review will be adequately maintained consistent with the rule and the applicant's license renewal program.

Containment Leak Rate

The containment leak rate program is an existing program which is supplemented by the containment inservice inspection program and credited with managing the aging effects due to loss of material and cracking for carbon and stainless steel components, cracking and change in material properties for primary containment elastomer electrical penetration sealant, and loss of material for carbon steel containment penetration components at Vermont Yankee Nuclear Station. As described in 10 CFR Part 50, Appendix J, the testing is required to assure that,

> (a) leakage through primary reactor containment and systems and components penetrating primary containment shall not exceed allowable values specified in technical specifications or associated bases, and
>
> (b) periodic surveillance of reactor containment penetrations and isolation valves is performed so that proper maintenance and repairs are made during the service life of containment, and systems and components penetrating primary containment.

The inspectors reviewed program documentation, integrated leakage rate test data and summary reports, aging management review documents, trending data for electrical penetrations, condition reports, and existing procedures to confirm that the applicant's existing containment leakage rate testing program effectively manages aging effects. The inspectors also verified that the applicant performed adequate historic reviews of plant specific and industry experience to determine aging effects. The containment leak rate program is consistent with NUREG 1801, "Generic Aging Lessons Learned Report" except where the NRC previously issued license amendments approving exceptions to 10 CFR Part 50, Appendix J. Specifically, license amendment 227 approved an extension of the primary containment integrated leak rate testing interval from 10 years to no longer than 15 years on a one time basis, and license amendment 223 approved an exemption to exclude main steam pathway leakage contributions from the overall integrated leakage rate Type A test measurement and from the sum of the leakage rates from Type B and Type C tests. The latter exception to NUREG 1801 was not

identified in the license application. The applicant has amended the application to include this exception (License Renewal Application, Amendment 26, page 6, March 23, 2007).

The inspectors also verified that the applicant performed adequate historic reviews of plant specific and industry experience to determine aging effects. The applicant provided adequate guidance to ensure aging effects are appropriately managed. The inspectors concluded the material condition of the systems, structures and components that require an aging management review, will be adequately maintained consistent with the rule and the applicant's license renewal program.

Diesel Fuel Monitoring

This is an existing program that will be modified for the purpose of aging management. The program is credited with managing the aging effects of loss of material (corrosion) and cracking in fuel oil system components at the Vermont Yankee Nuclear Station. The aging effects will be managed by a combination of periodic chemistry sampling and analysis, and periodic fuel oil tank inspections.

The existing fuel oil monitoring program, associated chemistry procedures, and analysis reports of the condition of fuel oils were reviewed to determine the effectiveness of the existing program. The inspectors reviewed tank inspection records to verify that the results were within the acceptable range. In addition, the inspectors interviewed selected system engineers and performed field walkdowns to independently assess the material condition of the fuel oil systems and identify inconsistencies between the as-built plant configuration and the aging management evaluations and programs. The aging management program credited modifying the existing program to include periodic ultrasonic thickness measurements of the bottom surface of the emergency diesel generator fuel oil storage tank, with approved acceptance criteria.

The inspectors identified that the fire pump diesel fuel oil tank was not included in the program enhancement to perform ultrasonic thickness measurements of the bottom surface of fuel oil tanks. To improve the proposed program enhancements, Entergy committed to include periodic ultrasonic thickness measurements of the bottom surface of the fire pump fuel oil tank and establish acceptance criteria (Commitment 44, BVY 06-009, BVY 07-018, LRA B.1.9). The first examination will occur prior to the period of extended operation. Subsequent test intervals will be determined based on engineering evaluations of the inspection results.

The inspectors identified that the existing fuel oil monitoring program for the security diesel, fire pump diesel, and portable fuel oil storage tanks did not conform to the recommendations in NUREG 1801, Section XI.M30, "Fuel Oil Chemistry," to monitor and control fuel oil quality in accordance with the guidelines of American Society for Testing and Materials (ASTM) Standards. To improve the proposed program enhancements, Entergy committed to perform periodic fuel oil sampling and analysis in accordance with the guidelines of the applicable ASTM Standards, and to establish

acceptance criteria based on those guidelines and engineering evaluations (Commitment 46 and 47, BVY 07-018 and License Renewal Application, Amendment 26, page 5, March 23, 2007).

The inspectors identified that the security diesel fuel oil day tank and associated fuel oil lines had not been reviewed for the effects of aging. Entergy performed an aging management review of the identified components, and modified the proposed program, as appropriate, based on the results of the review (License Renewal Application, Amendment 26, page 4 and 5, March 23, 2007).

The inspectors also verified that the applicant performed adequate historic reviews of plant specific and industry experience to determine aging effects. The applicant provided adequate guidance to ensure aging effects are appropriately managed. The inspectors concluded the material condition of the systems, structures and components that require an aging management review, will be adequately maintained consistent with the rule and the applicant's license renewal program.

<u>Environmental Qualification of Electric Components</u>

The Environmental Qualification of Electrical Components Program is an existing program credited with managing the aging effects in electrical components located in harsh environments subject to the requirements of 10 CFR 50.49 at Vermont Yankee Nuclear Power Station. The aging effects are managed by surveillance and maintenance activities that assure that Environmental Qualification equipment is maintained within its qualification basis and qualified life.

The inspectors reviewed program basis and implementation documents and conducted interviews with plant personnel to assure that the program was adequate and that extended life had been adequately addressed. The inspector reviewed with station personnel a change request for further evaluation of life extension issues and verified that the change was being made within the scope of the program. Based on this, aging effects will be adequately addressed within this program.

The inspectors also verified that the applicant performed adequate historic reviews of plant specific and industry experience to determine aging effects. The applicant provided adequate guidance to ensure aging effects are appropriately managed. The inspectors concluded the material condition of the systems, structures and components that require an aging management review, will be adequately maintained consistent with the rule and the applicant's license renewal program.

<u>Fire Protection</u>

The Fire Protection Program is an existing program modified for the purpose of aging management credited with managing the aging effects in the fire barrier system and in the diesel driven fire pump at the Vermont Yankee Nuclear Power Station. The aging effects are managed by periodic inspection of fire barrier penetration seals, fire barrier walls, ceilings, and floors, and periodic inspection and testing of fire rated doors. Aging

effects are also managed by the periodic inspection and testing of the diesel driven fire pump to ensure that the fuel supply line can perform its intended function.

The inspectors reviewed the existing fire protection program and supporting documents to verify the effectiveness of the program. The inspectors conducted interviews and performed walkdowns of various fire barriers throughout the plant to observe the effectiveness of the existing program. The inspectors also walked down the diesel driven fire pump and the associated fuel supply line. Surveillance procedures were reviewed for completeness and compliance with applicable codes. Program enhancements were reviewed for adequacy and completeness. An acceptable exception to the NUREG-1801 guidance was noted for frequency of inspection of penetration seals. NUREG-1801 states that 10% of each type of seal should be inspected at least once every refueling outage. Vermont Yankee Nuclear Power System program specifies inspection of approximately 25% of seals (regardless of seal type) each operating cycle, with all accessible seals being inspected at least once every four operating cycles. This is acceptable, since each accessible seal will be inspected at least once every four operating cycles, which is a higher frequency than the once every ten refueling outages recommended in NUREG-1801.

The inspectors also verified that the applicant performed adequate historic reviews of plant specific and industry experience to determine aging effects. The applicant provided adequate guidance to ensure aging effects are appropriately managed. The inspectors concluded the material condition of the systems, structures and components that require an aging management review, will be adequately maintained consistent with the rule and the applicant's license renewal program.

Fire Water System

The Fire Water System Program is an existing program modified for the purpose of aging management credited with managing the aging effects in water-based fire protection systems at the Vermont Yankee Nuclear Power Station. The aging effects are managed by periodic testing and inspection of systems and components exposed to water including sprinklers, nozzles, fittings, valves, hydrants, hose stations, standpipes, and above ground and underground piping and components.

The inspectors reviewed program bases documents and implementing procedures to assess the effectiveness of the existing program. The inspectors also conducted interviews and performed walkdowns of portions of the fire water system to observe the effectiveness of the existing program. Program enhancements were reviewed for adequacy and completeness. The inspectors discussed with the applicant several instances of surveillance testing and inspection frequencies that were less frequent than the guidance of NUREG-1801. These surveillances were fire hydrant hose hydrostatic tests, gasket inspections, fire hydrant flow tests, and sprinkler system piping visual inspections. The guidance in NUREG-1801 for these surveillance frequencies was based on the applicable National Fire Protection Association codes. To effectively manage the aging of the fire water system components, Entergy has agreed to meet the surveillance testing and inspection frequencies given in the applicable National Fire

Protection Association codes (Commitment 49, BVY 07-018). The inspectors noted the fire water program acceptance criteria was inconsistent with the guidance in NUREG-1801. NUREG-1801 stipulates, for fire water programs previously accepted by the NRC, that no biofouling exists in sprinkler systems that could cause corrosion in the sprinkler heads. This stipulation was not explicitly included in the Vermont Yankee Nuclear Power System criteria. Vermont Yankee Nuclear Power System does perform a periodic system flush and flow tests, however, biofouling is not specifically reviewed. To more effectively manage the aging of the fire water system components, Entergy has agreed to enhance the periodic test to include a check for fire water system biofouling rather then revise their application to note the exception to NUREG-1801. (Commitment 11, modified, BVY 07-018)

The inspectors also verified that the applicant performed adequate historic reviews of plant specific and industry experience to determine aging effects. The applicant provided adequate guidance to ensure aging effects are appropriately managed. The inspectors concluded the material condition of the systems, structures and components that require an aging management review, will be adequately maintained consistent with the rule and the applicant's license renewal program.

Flow-Accelerated Corrosion

The Flow Accelerated Corrosion Program is an existing program credited with managing the erosion/corrosion effects in systems containing safety and nonsafety-related carbon steel components carrying two-phase or single-phase high-energy fluid $\geq$ 2% of plant operating time at the Vermont Yankee Nuclear Station. The aging effects are managed by a program that is based on Electric Power Research Institute recommendations to predict, detect, and monitor flow accelerated corrosion in plant piping and other pressure retaining components. The program includes an evaluation to determine critical locations, initial operating inspections to determine the extent of thinning at these locations, and followup inspections to confirm predictions, or repair or replace components as necessary.

The inspector selected specific program attributes to sample the effectiveness of the Flow Accelerated Corrosion Program. These included: modifications to the sampling basis due to extended power uprate conditions; the use of CHECWORKS to assist the program manager in determining outage scope; implementation of recommended actions since the 2005 self-assessment (reference Entergy Self-Assessment Report, Vermont Yankee FAC Inspection Program, CR LO-VTY-2003-00327); use of additional methods to determine outage scope, including the use of engineering judgement; program manager knowledge of recent operating experience events and how these were reviewed for VY applicability; etc.

The inspector reviewed the Entergy Flow Accelerated Corrosion Program to determine whether it was effective. The inspector also verified that the Generic Letter 89-08, Erosion/Corrosion-Induced Pipe Wall Thinning, requirements are applied.

The inspectors concluded the applicant had conducted adequate evaluations as well as industry experience and historical reviews to determine aging effects managed by an aging management program. The applicant provided adequate guidance to ensure aging effects are appropriately managed. The inspectors concluded the material condition of the systems, structures and components that require an aging management review, will be adequately maintained consistent with the rule and the applicant's license renewal program.

Heat Exchanger Monitoring

This is a new program under development at the Vermont Yankee Nuclear Station. The program is credited with managing the aging effects of loss of material (corrosion) for heat exchanger tubes, heads, covers, and tube sheets, for those heat exchangers in the scope of license renewal that require periodic monitoring of aging effects, and are not covered by other existing periodic monitoring programs. The aging effects will be managed by periodic visual inspections and non-destructive examinations of selected heat exchangers in this program.

The inspectors identified that the residual heat removal service water pump motor coolers had not been included in the proposed program. Entergy committed to add these heat exchangers into the new program (License Renewal Application, Amendment 26, page 1, March 23, 2007).

The inspectors identified that the proposed program did not specify the criteria to determine which heat exchangers, within the program's scope, would be selected for inspection. In addition, the proposed program did not identify inspection frequency or acceptance criteria basis. To improve the proposed program, Entergy committed to select heat exchangers for inspection based on the materials of construction and associated environments, as well as the type of heat exchanger. At least one heat exchanger of each type, material, and environment combination will be included in the selected population to be examined. In addition, Entergy committed to perform eddy current examinations at a frequency consistent with Electric Power Research Institute recommendations, and evaluate the results against acceptance criteria also based on Electric Power Research Institute recommendations. The program will be initiated prior to the period of extended operation (License Renewal Application, Amendment 26, page 1, March 23, 2007).

The program, as described, will provide inspection and acceptance criteria, and will require evaluation of the inspection results. Inspections will be performed in accordance with approved station procedures. Inspection methods will include visual examinations on accessible heat exchanger heads, covers, and tube sheets, as well as eddy current non-destructive examinations of tube wall thickness, where practical. Tube wall thickness will be trended and corrective actions will be taken if projections indicate that acceptance criteria may not be met at the next inspection. Based on review of the proposed scope, parameters to be monitored, method of monitoring, and acceptance criteria, the inspectors determined that the proposed heat exchanger monitoring program, when implemented at the Vermont Yankee Nuclear Station, will

provide assurance that heat exchangers are routinely evaluated for age-related degradation of loss of material, and will adequately manage the identified aging effect.

The inspectors also verified that the applicant performed adequate historic reviews of plant specific and industry experience to determine aging effects. The applicant provided adequate guidance to ensure aging effects are appropriately managed. The inspectors concluded the material condition of the systems, structures and components that require an aging management review, will be adequately maintained consistent with the rule and the applicant's license renewal program.

Inservice Inspection - Containment Inservice Inspection - Containment

The Containment InService Inspection Program is an existing program modified for the purpose of managing the aging affects in the drywell in the Vermont Yankee Nuclear Station. The aging effects are managed by periodic visual and ultrasonic testing inspections. The inspectors reviewed the existing American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section XI, IWE program used to manage aging effects on the primary containment drywell. The inspector reviewed past drywell inspection results. The inspector reviewed the licensee's response to Generic Letter 87-05 and the commitments contained in their response.

The inspectors reviewed the documentation contained in CAR 91-063 which reports an incident in which water was known to have entered the drywell-to-shield wall air gap and entered the sand bed region. There was a single incident of water intrusion into the gap between the drywell liner and the concrete support structure, in 1991, caused by a leaking condensate line from a valve. It was conclusively determined, at the time, that the water found its way into the sandbed and by-passed the sandbed drains. Although a single incident of water intrusion was not intended to trigger the programmatic discipline discussed in NRC Interim Staff Guidance LR-ISG-2006-0, the inspectors noted a lack of precision in implementing GL 87-05 commitments. The following examples illustrate the weaknesses noted:

- Sand bed drain lines are not routinely monitored for leakage;
- There are no procedures in place to collect leakage and sample it; and
- During this inspection a boroscopic examination of the eight sandbed drains was observed by the inspectors (including the reported results) and it was noted the inspection was performed without specific written inspection or acceptance criteria.

Entergy addressed the above weaknesses. The inspector subsequently reviewed the licensee's planned installation of a collection system on the eight sand bed drains and the licensee's intention to install this system before entering the period of extended operation. Also, the inspector verified that the licensee has added a visual operator check of each of the sand bed drains for leakage conducted once each shift. All these actions are captured in the license renewal tracking system as LRS-16820.

The inspector also performed a walkdown of the drywell to concrete slab seal on the interior of the drywell. Visually, the seal was in good condition. The inspector also looked at the interior surface of the drywell and verified that drywell conditions were being reported, evaluated and repaired in accordance with the Section IX, Subsection IWE program.

Overall, the inspectors also verified that the applicant performed adequate historic reviews of plant specific and industry experience to determine aging effects. The applicant provided adequate guidance to ensure aging effects are appropriately managed. The inspectors concluded the material condition of the systems, structures and components that require an aging management review, will be adequately maintained consistent with the rule and the applicant's license renewal program.

<u>Inservice Inspection - Containment Inservice Inspection - Torus</u>

The Containment Inservice Inspection Program is an existing program modified for the purpose of managing the aging affects in the torus in the Vermont Yankee Nuclear Station. The aging effects are managed by periodic visual and ultrasonic testing inspections, and appropriate analyses and evaluations of the inspection results.

The torus inspection and re-coating, completed in 1998, was reviewed by the inspector. In 1998 the applicant had completed a detailed UT wall thickness survey and had evaluated the condition of the torus wall. The UT data and the analysis had determined that no margin to minimum wall thickness for general corrosion existed on the bottom of the torus . Additionally, the 1998 results revealed that two locations in the torus were below the acceptable, local minimum wall thickness. The applicant had concluded that there was adequate margin remaining in the remainder of the torus and that the localized thinning would not affect the safety function of the torus.

During the 2007 fall outage, the applicant repeated the ultrasonic thickness gauging of the torus. The calculated corrosion rate, based on the two sets of data, when adjusted for NDE uncertainty and the addition of a protective coating subsequent to 1998, was zero. The two prior locations where thin readings had been recorded were repeated; the thickness readings were determined to be acceptable. Additional thickness measurements are planned in subsequent outages to confirm the adequacy of the torus wall thickness.

With the additional work performed in 2007, the inspectors also verified that the applicant performed adequate historic reviews of plant specific and industry experience to determine aging effects. The applicant provided adequate guidance to ensure aging effects are appropriately managed. The inspectors concluded the material condition of the systems, structures and components that require an aging management review, will be adequately maintained consistent with the rule and the applicant's license renewal program.

Inservice Inspection

The Inservice Inspection Program (ISI) is an existing program credited with managing the aging affects in Class 1 and Class 2 piping systems in the Vermont Yankee Nuclear Station. The aging effects are managed by periodic sampling inspections and evaluation of inspection results.

The inspector reviewed the recent implementation of the existing ISI program. The inspector reviewed the last 3 inspection period summary reports and all presently approved relief requests to determine whether the licensee was meeting the requirements of the existing program. Additionally, the inspector reviewed two examples of the licensee's documentation of past indications discovered through inspection and the evaluation of those indications. The first example indication was a feedwater (FW) nozzle indication and the second example was an indication in a vertical weld between 2 reactor vessel plates. Both of the reviewed examples were completed correctly and addressed the conditions discovered. Disposition of the indications was proper and thorough. These examples demonstrate that the existing program has detected aging effects in the applicable systems.

The inspectors also verified that the applicant performed adequate historic reviews of plant specific and industry experience to determine aging effects. The applicant provided adequate guidance to ensure aging effects are appropriately managed. The inspectors concluded the material condition of the systems, structures and components that require an aging management review will be adequately maintained consistent with the rule and the applicant's license renewal program.

Instrument Air Quality

The instrument air quality program is a plant-specific program credited with managing the aging effects in the instrument air system of the Vermont Yankee Nuclear Station. The aging effects are managed by maintaining the instrument air system free from water and significant contaminants, thereby preserving an environment that is not conducive to loss of material.

The inspectors reviewed applicable procedures, test data, corrective action program documents, and performed a walkdown of selected monitoring locations near the control rod hydraulic control units to confirm that the applicant's instrument air quality program effectively manages aging effects.

The inspectors also verified that the applicant performed adequate historic reviews of plant specific and industry experience to determine aging effects. The applicant provided adequate guidance to ensure aging effects are appropriately managed. The inspectors concluded the material condition of the systems, structures and components that require an aging management review will be adequately maintained consistent with the rule and the applicant's license renewal program.

Non-EQ Inaccessible Medium-Voltage Cable

The Non-EQ Inaccessible Medium-Voltage Cable Program is a new program credited with managing the aging effects in inaccessible medium-voltage cables that are exposed to significant moisture simultaneously with applied voltage at the Vermont Yankee Nuclear Power Station. The aging effects are managed by periodic inspection for water collection in cable manholes and conduit, and draining water as needed. Also, in-scope cables will be periodically tested to provide an indication of the condition of the conductor insulation.

At the time of the inspection, the applicant had not completed any of the actions identified in the program. While the inspectors were not able to thoroughly assess the effectiveness of the implementation of the program as a whole, the applicant does have an informal procedure for inspecting cables located in manholes. The inspectors reviewed the procedure, conducted walkdowns, and had discussions with system engineers regarding how the proposed program will identify age-related deficiencies and acceptable testing methods. The inspectors concluded that, if effectively implemented as described in the program description documents, the aging effects in inaccessible medium-voltage cables would be adequately managed.

The inspectors concluded the applicant had conducted adequate evaluations, as well as industry experience and historical reviews, to determine aging effects managed by an aging management program. The applicant provided adequate guidance to ensure aging effects are appropriately managed. The inspectors concluded the material condition of the systems, structures and components that require an aging management review, will be adequately maintained consistent with the rule and the applicant's license renewal program.

Non-EQ Instrumentation Circuits Test Review

The Non-EQ Instrumentation Circuits Test Review Program is a new program credited with managing the aging effects in instrument cables exposed to adverse localized equipment environments at the Vermont Yankee Nuclear Power Station. The aging effects are managed by reviewing results of calibrations or surveillances of instrument circuits for in-scope components. Also, cable testing on in-scope cables will be performed for cables that are disconnected during the associated instrument calibration.

At the time of the inspection, the applicant had not completed any of the actions identified in the program. Therefore, the inspectors were unable to assess the effectiveness of the implementation of this program. However, the inspectors interviewed plant personnel, performed walkdowns, and reviewed associated documentation to assure the proposed program will be capable of managing aging effects. The inspectors concluded that, if effectively implemented as described in the program description documents, the aging effects in instrumentation circuits would be adequately managed.

The inspectors also verified that the applicant performed adequate historic reviews of plant specific and industry experience to determine aging effects. The applicant provided adequate guidance to ensure aging effects are appropriately managed. The inspectors concluded the material condition of the systems, structures and components that require an aging management review, will be adequately maintained consistent with the rule and the applicant's license renewal program.

Non-EQ Insulated Cables and Connections

The Non-EQ Insulated Cables and Connections Program is a new program credited with managing the aging effects in accessible insulated cables and connections exposed to adverse localized environments at the Vermont Yankee Nuclear Power Station. The aging effects are managed by periodic visual inspections of a representative sample of in-scope cables and connections for jacket surface anomalies.

At the time of the inspection, the applicant had not completed any of the actions identified in the program. Therefore, the inspectors were unable to assess the effectiveness of the implementation of this program. The inspectors reviewed drawings, conducted walkdowns, and had discussions with plant personnel to assess the proposed program and assure it will be capable of managing aging effects. The inspectors reviewed EPRI TR-109619, "Guideline for the Management of Adverse Localized Equipment Environments," the guidance document that will be used to develop the program procedures. The inspectors concluded that, if effectively implemented as described in the program description documents, the aging effects in accessible insulated cables and connections would be adequately managed.

The inspectors also verified that the applicant performed adequate historic reviews of plant specific and industry experience to determine aging effects. The applicant provided adequate guidance to ensure aging effects are appropriately managed. The inspectors concluded the material condition of the systems, structures and components that require an aging management review, will be adequately maintained consistent with the rule and the applicant's license renewal program.

Reactor Vessel Surveillance

The Reactor Vessel Surveillance Program is an existing program credited with managing the aging affects in the plates of the reactor vessel beltline area in the Vermont Yankee Nuclear Station. The aging effects are managed by monitoring actual reactor vessel beltline fluence, and measuring the fracture toughness of material specimens from the reactor vessel plates and from specimens from other similar BWR reactor vessel plates under similar conditions.

The inspector reviewed the Reactor Vessel Surveillance program documents and the monitoring which has been completed to date, as well as future monitoring commitments. The inspector also reviewed the licensee's responses to numerous "Requests for Additional Information" from NRR. The inspector discussed the program

with the onsite program owner and confirmed that the licensee is aware of the future requirements and their obligations to adequately monitor the condition of the reactor vessel material.

The inspectors also verified that the applicant performed adequate historic reviews of plant specific and industry experience to determine aging effects. The applicant provided adequate guidance to ensure aging effects are appropriately managed. The inspectors concluded the material condition of the systems, structures and components that require an aging management review will be adequately maintained consistent with the rule and the applicant's license renewal program.

Service Water Integrity

This is an existing program that is credited with managing the aging effects of loss of material (corrosion), cracking, and fouling service water system components at the Vermont Yankee Nuclear Station. The service water system includes the service water, residual heat removal service water, and alternate cooling systems. The aging effects are managed by a combination of surveillance and control techniques required by NRC Generic Letter 89-13, visual inspections and non-destructive examinations of service water piping and components, and by water chemical treatment.

The existing service water program, associated procedures, trending reports, heat exchanger inspection records, eddy current examination results and trending reports, heat exchanger tube plugging trends, and service water system health reports were reviewed to determine the effectiveness of the existing program. The inspectors interviewed selected system and program engineers and performed field walkdowns to independently assess the material condition of the service water systems and identify inconsistencies between the as-built plant configuration and the aging management evaluations and programs.

The inspectors identified that buried service water piping was not included in the existing program of visual inspections and non-destructive examinations. Entergy previously relied on inspection and examination results of accessible in-plant piping to be representative of the condition of the buried piping. To improve internal details of the existing program, Entergy voluntarily committed to the following enhancements for buried service water piping:

- Review industry operating experience on internal pipe inspections
- Develop a formal evaluation of the condition of the piping
- Evaluate available methods to verify pipe integrity
- Perform inspection/testing of the piping prior to the period of extended operation

In addition, the inspectors identified that buried service water piping had not been pressure tested as required by American Society of Mechanical Engineers (ASME) Code, section IWA-5244. Entergy determined the ASME requirement had been mis-interpreted by Entergy, and the buried piping had been tested by an alternate test

method, which verified flow was not impaired. Although use of the alternate test method required prior NRC approval, no relief request had been submitted. Entergy entered the issue into the corrective action program as condition reports CR-VTY-2007-00412 and CR-VTY-2007-00444. Because this was an issue affecting the current plant licensing basis, it was referred to the NRC resident inspection staff for followup.

The inspectors also verified that the applicant performed adequate historic reviews of plant specific and industry experience to determine aging effects. The applicant provided adequate guidance to ensure aging effects are appropriately managed. The inspectors concluded the material condition of the systems, structures and components that require an aging management review, will be adequately maintained consistent with the rule and the applicant's license renewal program.

Masonry Wall

The Masonry Wall program is an existing program as a part of the existing Structural Monitoring Program. The Masonry Program is credited with managing the aging effects in masonry walls in the Vermont Yankee Nuclear Power Station. The aging effects are managed by a program of inspection of masonry walls for cracking, on a frequency of 3 years, to assure that the established evaluation basis for each masonry wall remains valid during the period of extended operation.

The inspectors reviewed the program description, program basis documents, the currently approved station procedures, the results of prior inspections, discussions with cognizant personnel, and a walkthrough visual examination of accessible masonry walls to assess the effectiveness of the current program. The scope of the program includes all masonry walls that perform intended functions in accordance with 10 CFR 54.4, and were covered by NRC I. E. Bulletin 80-11. Masonry walls are inspected for cracking, and elastomers will be monitored for a change in material properties. The inspections are implemented through station procedures. Maintenance history revealed degradation (such as cracks) of masonry block walls along with repairs to specified criteria providing evidence the masonry wall program continues to function. In response to I.E. Bulletin 80-11, "Masonry Wall Design," and Information Notice 87-67, "Lessons Learned from Regional Inspections of Licensee Actions in Response to I.E. Bulletin 80-11," various actions have been taken including program enhancements, follow-up inspections to substantiate masonry wall analyses and classifications, and the development of procedures for tracking and recording changes to the walls. These actions have addressed all concerns raised by I.E. Bulletin 80-11 and Information Notice 87-67, such as unanalyzed conditions, improper assumptions, improper classification, and lack of procedural controls. A review of operating experience indicates that the program is effective for managing the aging effects of masonry walls.

The inspectors also verified that the applicant performed adequate historic reviews of plant specific and industry experience to determine aging effects. The applicant provided adequate guidance to ensure aging effects are appropriately managed. The inspectors concluded the material condition of the systems, structures and components that require an aging management review will be adequately maintained consistent with the rule and the applicant's license renewal program.

Structures Monitoring

The AMP is an existing program that will be further enhanced for the purpose of aging management of structures and structural components at the Vermont Yankee Nuclear Power Station. The program was developed based on guidance in Regulatory Guide 1.160 Revision 2, “Monitoring the Effectiveness of Maintenance at Nuclear Power Plants,” and NUMARC 93-01 Revision 2, “Industry Guidelines for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants,” to satisfy the requirements of 10 CFR 50.65, “Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants.” The scope of the program includes condition monitoring of masonry walls and water-control structures as described in the Masonry Wall Program and in the RG 1.127, Inspection of Water-Control Structures Associated With Nuclear Power Plants. The enhancement to the internal details of the program will include items such as seals, gaskets, process crane rail girders, condensate storage tank (CST) enclosure, carbon dioxide (CO_2) and nitrogen dioxide (NO_2) tank enclosures, CST pipe and diesel generator cable trenches, fuel oil pump house, service water pipe trench, and drywell floor liner seal. These items are not monitored under the current term but require monitoring during the period of extended operation. Aging effects are managed by periodic visual inspections by qualified personnel to monitor structures and components for applicable aging effects. Specifically, concrete structures are inspected for loss of material, cracking, and a change in material properties. Steel components are inspected for loss of material due to corrosion. Component supports will be inspected for loss of material, reduction or loss of isolation function, and reduction in anchor capacity due to local concrete degradation. Exposed surfaces of bolting are monitored for loss of material due to corrosion, loose nuts, missing bolts, or other indications of loss of preload. The scope of the program will be enhanced to include structures that are not monitored under the current term but require monitoring during the period of extended operation.

The inspectors reviewed the program description, program basis documents, the currently approved station procedures, the results of prior inspections, discussions with cognizant personnel, and a walkthrough visual examination of accessible structural items, including reinforced concrete and structural steel members, components and systems to assess the effectiveness of the current program. The scope of the program also includes all masonry walls that perform intended functions in accordance with 10 CFR 54.4, and were covered by I. E. Bulletin 80-11. The inspections included review of station procedures, maintenance history, inspection findings and followup of inspection findings, and current inspection schedules.

The inspectors also verified that the applicant performed adequate historic reviews of plant specific and industry experience to determine aging effects. The applicant provided adequate guidance to ensure aging effects are appropriately managed. The inspectors concluded the material condition of the systems, structures and components that require an aging management review will be adequately maintained consistent with the rule and the applicant’s license renewal program.

Water Chemistry Control – Closed Cooling Water

This is an existing program that is credited with managing the aging effects of loss of material (corrosion), cracking, and fouling in closed cooling water (CCW) system components at the Vermont Yankee Nuclear Station. The aging effects are managed by monitoring and controlling water chemistry in the CCW system.

The existing water chemistry monitoring program, associated chemistry procedures, and historical chemistry parameter trends were reviewed to determine the effectiveness of the existing program. The inspectors interviewed selected system engineers and performed field walkdowns to independently assess the material condition of the CCW system. The water chemistry program is comparable to the program described in NUREG 1801, Section XI.M21, "Closed-Cycle Cooling Water System." In addition, the One-Time Inspection Program will verify the effectiveness of the water chemistry control program by performing internal inspections on selected CCW heat exchangers or other components cooled by CCW, to confirm that unacceptable degraded conditions are not present.

The inspectors also verified that the applicant performed adequate historic reviews of plant specific and industry experience to determine aging effects. The applicant provided adequate guidance to ensure aging effects are appropriately managed. The inspectors concluded the material condition of the systems, structures and components that require an aging management review will be adequately maintained consistent with the rule and the applicant's license renewal program.

c. Conclusions

Based on the inspection, the applicant made a number of enhancements to the programs, which were documented either via amendments to the license renewal application or through commitments.

The inspection team concluded that screening and scoping of nonsafety-related systems, structures, and components, were implemented as required in 10 CFR 54.4(a)(2), and the aging management portions of the license renewal activities were conducted as described in the License Renewal Application. The inspection results supported a conclusion that the proposed activities will reasonably manage the effects of aging in the systems, structures, and components identified in your application. The inspection concluded the documentation supporting the application was in an auditable and retrievable form.

40A6 Meetings, Including Exit

The inspectors presented the inspection results to Mr. T. Sullivan, Vermont Yankee Power Station Vice President, and other members of the licensee's staff in a meeting that was open for public observation on May 24, 2007. The licensee had no objections to the NRC observations. No proprietary information was provided to the inspectors during this inspection. The slide used during this exit meeting is number 26 (ADAMS ML#071520349).

ATTACHMENT

SUPPLEMENTAL INFORMATION

LIST OF DOCUMENTS REVIEWED

Drawings

Vermont Yankee Dwg. G-191150, Revision 20; General Arrangement Reactor Building Sections
Vermont Yankee Dwg. G-191480, Revision 2; Reactor Building Substructures - Sections & Details
Vermont Yankee Dwg. G-191481, Revision 2; Reactor Building Substructures - Plan, Elevations & Details
Vermont Yankee Dwg. G-191482, Revision 4; Reactor Building Foundation Mat Plan - MAS
Vermont Yankee Dwg. G-191483, Revision 4; Reactor Building Foundation Mat Plan - M&R
Vermont Yankee Dwg. G-191484, Revision 4; Reactor Building Foundation Mat - Dowel Plan
Vermont Yankee Dwg. G-191277, Revision 6; Reactor Fuel Storage Pool Piping, Sections & Details
Vermont Yankee Dwg. 5920-14, Revision 3; Arrangement Primary & Secondary Containment
GE Dwg. 729E243, Revision 3, Page 12 of 13
GE Dwg. 729E243, Revision 3, Page 10 of 13
GE Dwg. 729E243, Revision 2, Page 13 of 14
GE Dwg. 729E243, Revision 3, Page 12 of 13
GE Dwg. 919D218, Revision 6, Refueling Bellows
GE Dwg. 728E945, Revision 1, Reactor Well Seal Arrgt.
GE Dwg. 729E509, Revision 1, Lower Drywell Concrete Pours, Primary Containment
Vermont Yankee Dwg. G-191205, Sheet 1 of 5, Revision 5; Reactor Building Composite, Primary Containment
GE Dwg. 729E253, Sheet 3, Revision 1, Typical Drywell Penetration Configurations
Vermont Yankee Dwg. G-191668, Revision 20; Reactor Building - Plans & Details - Sheet 1, Plumbing & Drainage
Vermont Yankee Dwg. 6202-0001, Revision 3; General Plan Pressure Suppression Containment Vessel
VY-E-75-001, Rev 11, Flow Diagram A.O.G.
G-191162, Sheet 2, Rev 25, Flow Diagram, Miscellaneous Systems, Fuel Oil
G-191162, Sheet 3, Rev 22, Flow Diagram, Miscellaneous Systems, Exhaust Stack & Off Gas System
G-191191, Rev 10, Off Gas Piping – Yard
5920 FS-I67A, Rev 4, Off Gas-Stack
G-191159, Sheet 1, Rev 72, Flow Diagram, Service Water System
G-191159, Sheet 2, Rev 85, Flow Diagram, Service Water System
G-191159, Sheet 6, Rev 9, Flow Diagram – AOGCCW
G-191159, Sheet 2, Rev 85, Flow Diagram, Service Water System
VYI-SW-PART15, Sheet 1, Rev 2, Piping Isometric Drawing, Service Water, Turbine Building, South West Condenser, (SW) Part 15

VYI-SW-PART15, Sheet 2, Rev 3, Piping Isometric Drawing, Service Water, Turbine Building, (SW) Part 15
VYI-SW-PART8&8A, Sheet 1, Rev 1, Piping Isometric Drawing, Service Water, Torus Catwalk North Area, (SW) Part 8&8A
VYI-RCW-PART11, Sheet 2, Rev 0, Piping Isometric Drawing, Reactor Coolant Water, Reactor Building, (RCW) Part 11
VYI-SW-PART3, Sheet 2, Rev 0, Piping Isometric Drawing, Service Water, Inside Intake Structure, SW Part 3
B-191301, “Cable and Conduit List”, Sheet 325, Rev 9
G-191159, “Flow Diagram Service Water System”, Sheet 1, Rev 73
G-191163, “Flow Diagram Fire Protection System Inner Loop”, Sheet 1, Rev 39
G-191163, “Flow Diagram Fire Protection System Outer Loop”, Sheet 2, Rev 13
G-191163, “Flow Diagram Fire Protection System Low Pressure CO2”, Sheet 3, Rev 8
G-191163, “High Pressure CO2 Suppression System”, Sheet 4, Rev 0
G-191384, “Yard Duct Runs and Grounding (Site Plan)”, Sheet 1, Rev 43
G-191384, “Electrical Manhole Details”, Sheet 6, Rev 7
G-191162, sheet 2, revision 25, "Fuel Oil Systems Flow Diagram"
G-191159, sheet 1, revision 72, "Service Water System Flow Diagram"
G-191159, sheet 2, revision 89, "Service Water System Flow Diagram"
G-191159, sheet 3, revision 38, "Reactor Closed Cooling Water System Flow Diagram"
Vermont Yankee Dwg. G-191150, Revision 20; General Arrangement Reactor Building Sections
Vermont Yankee Dwg. G-191480, Revision 2; Reactor Building Substructures - Sections & Details
Vermont Yankee Dwg. G-191481, Revision 2; Reactor Building Substructures - Plan, Elevations & Details
Vermont Yankee Dwg. G-191482, Revision 4; Reactor Building Foundation Mat Plan - MAS
Vermont Yankee Dwg. G-191483, Revision 4; Reactor Building Foundation Mat Plan - M&R
Vermont Yankee Dwg. G-191484, Revision 4; Reactor Building Foundation Mat - Dowel Plan
Vermont Yankee Dwg. G-191277, Revision 6; Reactor Fuel Storage Pool Piping, Sections & Details
Vermont Yankee Dwg. 5920-14, Revision 3; Arrangement Primary & Secondary Containment
GE Dwg. 728E945, Revision 1, Reactor Well Seal Arrgt.
Vermont Yankee Dwg. G-191205, Sheet 1 of 5, Revision 5; Reactor Building Composite, Primary Containment

Procedures

OP 0150, Revision 169, 8/30/06; Conduct of Operations and Operator Rounds
ENN-NDE-10.01, Revision 2, 2/13/06; VT-1 Examination
ENN-EP-S-001, Revision 0, 3/19/04; IWE General Visual Containment Inspection (Engineering Standard)
PP 7037, Revision 1, 6/15/05; Safety-Related Coating Program
EN-OE-100, Revision 2, 11/28/05; Operating Experience Program
NE 8604, Revision 1, 4/23/01; Non-code Visual Examination Methods As Good Maintenance Practice
EN-DC-178, Rev 0, System Walkdowns

ENN-MS-S-009-VY, Attachment 7.6, Rev 0, Pipe Line Number and Isometric Cross Reference
ENN-DC-334, Rev 0, Primary Containment Leakage Rate Testing (Appendix J)
EN-DC-334, Rev 0, Primary Containment Leakage Rate Testing (Appendix J)
PP 7006, Rev 9, Primary Containment Leakage Rate Testing Program
SEP-APJ-009, Rev 0, Vermont Yankee Primary Containment Leakage Rate Testing (Appendix J) Program Section
RP 4620, Rev 17, Sampling and Analysis of Instrument and Containment Instrument Air Supplies
AP 0077, “Barrier Control Process”, Rev 5
AP 0092, “Environmental Qualification Document Change Notification”, Rev 7
ENN-DC-164, “Environmental Qualification Program”, Rev 2
ME 176, “Preventive Maintenance Basis”, Rev 0
OP 0023, “Installation and Testing of Cable and Conduit”, Rev 9
OP 0046, “Installation and Repair of Fire Barriers, Penetration Seals, Fire Breaks, and Flood Seals”, Rev 12
OP 4002, “Integrity Surveillance of Fire Detectors and Fire Suppression Systems”, Rev 14
OP 4019, “Surveillance of Plant Fire Barriers and Fire Rated Assemblies”, Rev 23
OP 4103, “Fire Protection Equipment Surveillance”, Rev 40
OP 4104, “Fire Hose Hydro Test Surveillance”, Rev 16
OP 4105, “Fire Protection Systems Surveillance”, Rev 41
OP-4221, “Surveillance of Gas Fire Extinguishing Systems”, Rev 15
OP 4393, “Test of the Cable Vault, Switchgear Rooms, and Intake Structure CO2 Systems,” Rev 20
OP 5235, “AC and DC Motor Maintenance”, Rev 11
OP-4623, revision 29, "Sampling and Treatment of Closed Cooling Water Systems"
RP-4652, revision 5, "Sampling & Testing for Microbiologically Influenced Corrosion"
OP-5265, revision 5, "Service Water Component Inspection and Acceptance Criteria"
PP-7021, revision 2, "Service Water Program"
PP-7601, revision 2, "Service Water Chemical Treatment and Monitoring Program"
OP-4613, revision 35, "Sampling and Testing of Diesel Fuel Oil"
OP-2195, revision 23, "Fuel Oil Storage and Transfer System"
PP-7015, revision 0, "Inservice Inspection Program"
SEP-PT-001, revision 0, "Inservice Inspection Program"

<u>Aging Management Review Technical Basis Documents</u>

LRPD-04, Revision 0, 1/25/06; TLAA Mechanical Fatigue
LRPD-03, Revision 0, 1/25/06; TLAA and Exemption Evaluation Results
AMRC-01, Revision 0, 1/19/06; Aging Management Review of the Primary Containment
Evaluation of Aging Management Programs, Section 4.14.1: Inservice Inspection (ISI) Program, LRPD-02, Revision 1, page158 of 161
Evaluation of Aging Management Programs, Section 4.14.2: Containment Inservice Inspection (CII) Program, LRPD-02, Revision 1, page 162 of 275
Evaluation of Aging Management Programs, Section 4.19: Reactor Vessel Surveillance Program, LRPD-02, Revision 1, page 185 of 275
LRPD-02, Rev 1, Aging Management Program Evaluation Results, Section 4.22, System Walkdown Program

LRPD-02, Rev 1, Aging Management Program Evaluation Results, Section 3.6, Selective Leaching Program
LRPD-02, Rev 1, Aging Management Program Evaluation Results, Section 4.23.2, Water Chemistry Control – BWR Program
LRPD-05, Rev 1, Operating Experience Review Results, Section 3.4.8, Containment Leak Rate Program
LRPD-05, Rev 1, Operating Experience Review Results, Section 4.4.8, Containment Leak Rate Program

Documents

VYNPS Design Basis Document for Containment Pressure Suppression System, Document CPS, Revision 4, 9/27/05
Vermont Yankee, Owner's Activity Report (OAR-1) for Inservice Inspections, October 25, 2002 through May 4, 2004
Report for IWE Inspections Performed for RFO22, 12/2/99

IR & CR

IDR 01-07, 5/2/01
IDR 01-08

Event Reports

ER 20012009
LER 85-007-01, 1985 Appendix J Type B and C Leak Testing Failure Due to Mechanical and Personnel Error
LER 86-007-00, 1985/86 Appendix J Type A Test Failure Due to Penetration Leakage

Work Documents

Emergency Diesel Fuel Oil Storage Tank Sample No. 3482, dated 6-20-2006
Emergency Diesel Fuel Oil Storage Tank Sample No. 7169, dated 12-18-2006
Diesel Fire Pump Fuel Oil Sample No. 7170, dated 12-18-2006
John Deere Diesel Fuel Oil Storage Tank Sample, dated 10-30-2006
John Deere Diesel Fuel Oil Storage Tank Sample No. 0916, dated 2-16-2006
STP-95-02, revision 0, "Hydraulic Performance Testing of Alternate Cooling

Calculations

VYC - 2043, Revision 0, 12/1/99; Evaluation of Primary Containment Localized Thinning and Screening Criteria for ASME XI IWE Inspections
VYM 98/132, 7/21/98; Summary of ASME XI Subsection IWE Torus Shell & Downcomer Inspections Performed During the 1998 Refueling Outage
VYC - 1032, Revision 0, 12/31/91; Torus & Vent System Thickness Requirements
VYM 98/148, 2/6/99; ASME XI Subsection IWE Report on Evaluation of 1998 Torus Inspections
VYM 99/147, 11/28/99; Closure of Inservice Discrepancy Report, IDR 99-07
VYM 99/139, 11/24/99; Closure of Inservice Discrepancy Reports, IDR 99-012 and IDR 99-018
Job Order 98-0113, ASME XI, IWE Miscellaneous Information, 2/6/98

VYC-1854, “Determination of Ampacity for Safety Related Power Cables for the AC Auxiliary Power Distribution System”, Rev 0

NRC Documents

NRC Inspection Report 50-271/91-24, 11/7/91
NRC Ltr. NVY 04-104, 9/30/04; Safety Evaluation of Relief Request ISI-06 for Volumetric Examination of Reactor Pressure Vessel Circumferential Shell Welds in Accordance With Boiling Water Reactor Vessel and Internals Project-05 and Generic Letter 98-05, Vermont Yankee Nuclear Power Station (TAC N0. MC0848)
NRC Ltr. NVY 04-016, 3/5/04; Safety Evaluation of Relief Requests for the Fourth 10-year Interval of the Inservice Inspection Program - Vermont Yankee Power Station (TAC Nos. MB8349 through MB8358)
Generic Letter 89-13, "Safety Related Service Water and Heat Exchanger Problems"
Information Notice 1991-46, "Degradation of Emergency Diesel Generator Fuel Oil Systems"
Information Notice 2006-22, "Ultra-Low Sulfur Fuel Oil Could Adversely Impact Performance"
NUREG 1801, revision 1, "Generic Aging Lessons Learned (GALL) Report"

Licensee Letters

Entergy Ltr. BVY 06-009, 1/25/06; VYNPS License No. DPR-28 (Docket No. 50-271), License Renewal Application
Entergy Ltr. BVY 06-010, 1/25/06; VYNPS License No. DPR-28 (Docket No. 50-271), License Renewal Application Boundary Drawings
Vermont Yankee Ltr. FVY 87-52, 5/7/87; Response to Generic Letter 87-05 - Request For Additional Information - Assessment of Licensee Measures to Mitigate and/or Identify Potential Degradation of Mark I Drywells
Vermont Yankee Ltr. BVY 89-69, 7/1/89; Vermont Yankee Response to NRC Request for Information Regarding Condition of Drywell Paint (IR 89-80)
Vermont Yankee Ltr. BVY 01-82, VYNPS License No. DPR-28 (Docket No. 50-271), Part 21 Reportable Defect
Vermont Yankee Ltr. BVY 06-013, VYNPS Vermont Yankee 2005 Summary Reports for Inservice Inspection and Repairs or Replacements
Entergy Ltr. BVY 03-83, 9/25/03; VYNPS License No. DPR-28 (Docket No. 50-271),
Supplement to Fourth-Interval Inservice Inspection (ISI) Program Plan - Submittal of Relief Request ISI-06
BVY 92-91, Transmittal of 1992 Vermont Yankee Reactor Containment Building Integrated Leakage Rate Test Reports
BVY 90-036, Supplemental Information on Vermont Yankee’s Primary Containment Leak Rate Test Report
BVY 89-64, Transmittal of VY 1989 Primary Containment Leak Rate Test Report
FVY 86-91, Primary Containment Leak Rate Test Report
FVY 83-97, Primary Containment Leak Rate Test Report
WVY 80-16, Primary Containment Leak Rate Testing
BVY 89-17, Vermont Yankee Response to NRC Generic Letter 88-14: “Instrument Air Supply System Problems Affecting Safety-Related Equipment,” February 16, 1989

Vendor Documents

Teledyne Technical Report, TR-7426, Revision 1, 11/21/91; Mark I Torus Shell And Vent System Thickness Requirements For Vermont Yankee Nuclear Station

Work Orders

98-1108, 4/29/98, Torus Seam Repair
97-0734, 9/24/97, UT Thickness Readings - Torus Strainer Mods
2000-060, MM 2000-010, Drywell Seal Repair & UT Readings
05-000968
05-002076
05-005111
05-005114

Corrective Action Program (Note the corrective actions taken as a consequence of our inspection with an asterisk)

CAR 91-063
CR-VTY-1997-00307
CR-VTY-2003-02321
CR-VTY-2005-00048
CR-VTY-2005-00261
CR-VTY-2005-02710
CR-VTY-2002-01913
CR-VTY-2002-00698
CR-VTY-2002-00353
CR-VTY-2001-02066
CR-VTY-2002-02518
CR-VTY-2003-00992
CR-VTY-2001-02009
CR-VTY-2001-02066
CR-VTY-2002-00353
CR-VTY-2002-00698
CR-VTY-2001-01195
CR-VTY-2001-00860
CR-VTY-2001-01169
CR-VTY-2007-00447*
CR-VTY-2007-00445*
CR-VTY-2007-00442*
CR-VTY-2007-00443*
CR-VTY-2007-00257*
CR-VTY-2007-00258*
LO-OEN-2005-00361
CR-VTY-2000-01325
CR-VTY-1996-00664
CR-VTY-2004-01645
CR-VTY-2002-02974
CR-VTY-1997-00127

CR-VTY-2000-1848
CR-VTY-2007-0262
CR-VTY-2007-00412*
CR-VTY-2007-00444*

* Indicates that the CR was generated as a result of this inspection.

Aging Management Programs

Aging Management Review of the Primary Containment, AMRC-01, Revision 0, 1/19/06
TLAA and Exemption Evaluation Results, LRPD-03, Revision 0, 1/25/06
Aging Management Review of the Primary Containment Atmosphere Control and Containment Atmosphere Dilution Systems, AMRM-08, Revision 1, 4/11/06
Aging Management Review of the Reactor Pressure Vessel, AMRM-31, Revision 0, 1/25/06
Vermont Yankee Containment Inservice Inspection (CII) Program, SEP-CI-01
Vermont Yankee Containment Inservice Inspection (CI) Program, Section SEP-CI-001, Rev 0
AMRM-30, Rev 1, Aging Management Review of Nonsafety-related Systems and Components Affecting Safety-related Systems
AMRM-20, Rev 0, Aging Management Review of the Primary Containment Penetrations
AMRC-01, Rev 0, Aging Management Review of the Primary Containment
Reactor Containment Building Integrated Leakage Rate Test, July 1995
AMRM-16, Rev 0, Aging Management Review of the Instrument Air System
Semi-Annual Instrument Air Sampling Report, January 11, 2002
AMRM-02, revision 1, "Aging Management Review of the Residual Heat Removal System"
AMRM-11, revision 0, "Aging Management Review of the Service Water System"
AMRM-12, revision 0, "Aging Management Review of Reactor Building Closed Cooling Water System"
AMRM-15, revision 0, "Aging Management Review of the Fuel Oil System"
AMRM-30, revision 1, "Aging Management Review of Nonsafety-Related Systems Affecting Safety-Related Systems"

Torus Inspection Data Sheets:

YA-UT-112-24, 2 pages
YA-UT-112-46, 2 pages
YA-UT-112-47, 2 pages
YA-IWE-VT-062, 1 page
YA-IWE-VT-111, 2 pages
YA-IWE-VT-188, 11 pages
YA-IWE-VT-146, 2 pages
YA-IWE-VT-189, 2 pages
YA-IWE-UT-228, 1 page
YA-IWE-VT-177, 1 page
YA-IWE-VT-134, 1 page
YA-IWE-VT-135, 1 page
YA-IWE-VT-136, 1 pages
24-9.05-TORUS-01 (UT), 5 pages
Torus & Penetration VT - Bays 1 - 16 interior, 4/8/04, 70 pages
IWE-GV-359, 1 page

IWE-GV-365, 1 page
IWE-GV-360, 1 page
IWE-GV-367, 1 page
IWE-GV-371, 1 page
IWE-GV-392, 1 page

Drywell Inspection Data Sheets:

IWE 8045-01-010, 1 page
IWE 8053-01-063, 2 pages
IWE 8053-01-064, 2 pages
IWE 8045-01-019, 6 pages
IWE 8047-01-079, 1 page
IWE 8047-01-073, 1 page
IWE GV-353, 1 page
IWE GV-354, 1 page
IWE GV-355, 1 page
IWE GV-356, 1 page
IWE GV-357, 1 page
IWE GV-358, 1 page
IWE GV-359, 1 page
IWE 8045-01-01, 1 page
IWE 8045-01-02, 1 page
IWE 8045-01-03, 1 page
IWE 8045-01-04, 1 page
IWE 8045-01-05, 1 page
IWE 8045-01-06, 1 page
IWE 8045-01-017, 1 page
IWE 8045-01-014, 1 page
IWE GV-398, 1 page
IWE 8047-01-031, 9 pages
IWE GV-360, 1 page
IWE GV-393, 1 page
IWE 8047-01-044, 1 page
IWE GV-394, 1 page
IWE 8047-01-043, 1 page
IWE 8047-01-042, 1 page
IWE 8047-01-041, 1 page
IWE GV-383, 1 page
IWE GV-382, 1 page
IWE 8047-01-040, 1 page
IWE 8047-01-051, 1 page
IWE GV-381, 1 page
IWE 8047-01-039, 1 page
IWE GV-380, 1 page
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IWE GV-374, 1 page
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IWE 8047-01-038, 1 page
IWE 8047-01-037, 1 page
IWE 8047-01-036, 1 page
IWE 8047-01-050, 1 page
IWE 8047-01-035, 1 page
IWE GV-372, 1 page
IWE GV-368, 1 page
IWE GV-371, 1 page
IWE GV-370, 1 page
IWE GV-369, 1 page
IWE GV-375, 1 page
IWE GV-365, 1 page
IWE GV-367, 1 page
IWE GV-366, 1 page
IWE GV-364, 1 page
IWE GV-363, 1 page
IWE 8047-01-034, 1 page
IWE 8047-01-049, 1 page
IWE 8047-01-033, 1 page
IWE GV-362, 1 page
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IWE GV-384, 1 page
IWE GV-387, 1 page
IWE GV-386, 1 page
IWE GV-385, 1 page
IWE GV-392, 1 page
IWE GV-400, 1 page
IWE GV-399, 1 page
IWE GV-390, 1 page
IWE GV-391, 1 page
IWE GV-401, 1 page
IWE GV-402, 1 page
IWE GV-389, 1 page
IWE GV-388, 1 page
IWE GV-397, 1 page
IWE GV-396, 1 page
IWE GV-395, 1 page
IWE 8045-01-011, 2 pages
IWE 8045-01-013, 1 page
IWE 8045-01-015, 1 page
IWE 8045-01-012, 1 page
IWE 8047-01-32, 1 page
NDE Rpt #07-001, 2 pages; Sand bed drains, 2/1/07
NDE Rpt #07-002, 2 pages; Sand bed drains, 2/1/07
NDE Rpt #07-003, 2 pages; Sand bed drains, 2/1/07
NDE Rpt #07-004, 2 pages; Sand bed drains, 2/1/07
NDE Rpt #07-005, 2 pages; Sand bed drains, 2/1/07

NDE Rpt #07-006, 2 pages; Sand bed drains, 2/1/07
NDE Rpt #07-007, 2 pages; Sand bed drains, 2/1/07
NDE Rpt #07-008, 2 pages; Sand bed drains, 2/1/07

Visual/Surface Inspection Certification & Eye Examination Records:

King, D.
Griffin, M.

License Renewal Change Requests

Entergy Ltr. BVY 06-043, 5/15/06; VYNPS License No. DPR-28 (Docket No. 50-271), License Renewal Application Amendment No. 2
Entergy Ltr. BVY 06-058, 7/6/06; VYNPS License No. DPR-28 (Docket No. 50-271), License Renewal Application Amendment No. 3
Entergy Ltr. BVY 06-064, 7/14/06; VYNPS License No. DPR-28 (Docket No. 50-271), License Renewal Application Amendment No. 5
Entergy Ltr. BVY 06-063, 7/14/06; VYNPS License No. DPR-28 (Docket No. 50-271), License Renewal Application Amendment No. 4
Entergy Ltr. BVY 06-077, 8/10/06; VYNPS License No. DPR-28 (Docket No. 50-271), License Renewal Application Amendment No. 9
Entergy Ltr. BVY 06-078, 8/15/06; VYNPS License No. DPR-28 (Docket No. 50-271), License Renewal Application Amendment No. 10
Entergy Ltr. BVY 06-079, 8/22/06; VYNPS License No. DPR-28 (Docket No. 50-271), License Renewal Application Amendment No. 11
Entergy Ltr. BVY 06-083, 9/5/06; VYNPS License No. DPR-28 (Docket No. 50-271), License Renewal Application Amendment No. 12
Entergy Ltr. BVY 06-086, 9/19/06; VYNPS License No. DPR-28 (Docket No. 50-271), License Renewal Application Amendment No. 13
Entergy Ltr. BVY 06-088, 9/20/06; VYNPS License No. DPR-28 (Docket No. 50-271), License Renewal Application Amendment No. 14
Entergy Ltr. BVY 06-090, 9/20/06; VYNPS License No. DPR-28 (Docket No. 50-271), License Renewal Application Amendment No. 15
Entergy Ltr. BVY 06-091, 10/17/06; VYNPS License No. DPR-28 (Docket No. 50-271), License Renewal Application Amendment No. 16
Entergy Ltr. BVY 06-096, 10/20/06; VYNPS License No. DPR-28 (Docket No. 50-271), License Renewal Application Amendment No. 17
Entergy Ltr. BVY 06-095, 10/20/06; VYNPS License No. DPR-28 (Docket No. 50-271), License Renewal Application Amendment No. 18, Response to Request for Clarification of SAMA RAI Responses
Entergy Ltr. BVY 06-097, 10/31/06; VYNPS License No. DPR-28 (Docket No. 50-271), License Renewal Application Amendment No. 19
Entergy Ltr. BVY 06-098, 11/9/06; VYNPS License No. DPR-28 (Docket No. 50-271), License Renewal Application Amendment No. 20
Entergy Ltr. BVY 06-099, 11/6/06; VYNPS License No. DPR-28 (Docket No. 50-271), License Renewal Application Amendment No. 21, Response To Request for Clarification of SAMA RAIs

Entergy Ltr. BVY 06-100, 12/4/06; VYNPS License No. DPR-28 (Docket No. 50-271), License Renewal Application Amendment No. 22, Clarification of Aging Management Program and Environmental Report Items

Miscellaneous Documents

Drawing Review (Structural Interaction) for AMRM-30 Aging Management Review of Nonsafety-Related Systems and Components
ME040, PM Basis Identification, Dryer, Air
ME060, PM Basis Identification, Compressor, Air
ME160, PM Basis Identification, Dryer, Air
Apparent Cause Evaluation ER 2003-1812, “Low Megger Results on Vernon Tie Cables,” 09/15/03
E-mail, Leonard Parker (National Grid) to Paul Johnson, “Vernon Substation Periodic Maintenance,” 11/28/06
Fire Hazards Analysis, Rev 7
Fire Protection Commitment Reference Manual, Rev 1
Technical Requirements Manual, Section 3.13, Rev 25
Heat Exchanger Inspection Data Report No. 5265.02, dated 1-24-2007
Heat Exchanger Eddy Current Inspection Summary Report, dated 6-30-2006
LRPD-02, revision 2, "Aging Management Program Evaluation Results"
Service Water System Health Report, 2006 Quarter 3 and Quarter 4
VYSE-APL-2007-009, revision 0, "VY Action Plan for Buried Service Water Piping"
Preventative Maintenance Task M-307 Basis, revision 15, "Cooling Tower Maintenance"
American Society of Mechanical Engineers Section XI 1998 Edition, with 2000 Addenda

Engineering Change Requests

ECR 316, “Perform Evaluation of EQ Equipment for Impact of License Renewal”
Technical Requirements Manual
ECR 490
ER 04-0873, “Vernon Tie Cable Replacement,” Rev 0

BWR Stress Corrosion Cracking

UT Data Sheet, 3-21-95, Exam of Core Spray Nozzle Safe End to Pipe Weld CS4A-MF6A
Liquid Penetrant Examination Report for CS4B-MF6A, 5-12-01
UT Data Sheet, 5-12-01, CS4B-MF6A
UT Data Sheet, 4-10-98, Core Spray Overlay N5B-SE, Safe End to Nozzle
UT Data Sheet, 4-14-04, Core Spray Overlay, N5A-SE, Safe End to Nozzle
2002 ISI Reports, 1-9-03
EPRI BWR-VIP 75, 75A
Relief Request ISI-013, Fourth ISI Interval
Inservice Discrepancy Report for N1B-SE, Recirc Outlet Nozzle to Safe End, 11-11-99 PT Exam
Intergranular Stress Corrosion Cracking Inspection Program, Report JAF-RPT-MULT-01120
Vermont Yankee Intergranular Stress Corrosion Cracking Inspection Program, Engineering Report VY-RPT-06-0004/EN-DC-147, Rev. 1
Drawing “Reactor 8" Diameter Nozzle MK N5A & B, No. 5920-624

ASME Code Case –560, Alternative Examination Requirements for Class I, Category B-J Piping Welds, Section XI, Division I
CR-VTY-2007-00418

BWR Vessel Internals

BWRVIP-06A: BWR Vessel and Internals Project, Safety Assessment of BWR Reactor Internals
BWRVIP Inspection Program, ENN-DC-135, Rev. 5
Engineering Report No. VY-RPT-06-0006, Rev. 0, RV Internals Inspection Program
VYC-3001 Rev. 3, EPU Steam Dryer Acceptance Criteria, ER No. 04-1409 Design Basis Calculation
BWRVIP-76, Core Shroud Inspection and Flaw Evaluation Guidelines
ENN-DC-112, Rev. 5, ER 05-0621, Core Shroud Structural Design-Reliant Welds
VY 2007 Outage Inspection Scope, RFO 26 (2007) Weld Inspection List
CR-VTY-2007-186

Flow-Accelerated Corrosion

Recommendations for an Effective Flow-Accelerated Corrosion Program, EPRI NSAC-202L-R3
EN-DC-315, Rev. 0, Flow Accelerated Corrosion Program
Vermont Yankee Response to NRC Generic Letter 89-08, “Erosion/Corrosion-Induced Pipe Wall Thinning”
Entergy Nuclear Vermont Yankee Monthly Chemistry Plant Data Report, December 2006
Entergy Self-Assessment Report, Vermont Yankee FAC Inspection Program, CR LO-VTY-2003-00327
VY Piping FAC Inspection Program PP 7028-2005 Refueling Outage Inspection Report (RFO 25)
CR LO-VTYLO-2003-00327 CA3
VY Piping FAC Inspection Program PP 7028-2007 Refueling Outage
Engineering Report No. VY-RPT-05-00012, Rev. 0, VY Piping FAC
Draft ENVY Engineering Report VY-RPT-05-00013, Rev. 0, VY Piping FAC Inspection Program - Small Bore Piping Component Inspection Database

Buried Piping

CR-VTY-2003-01957
Event Report 20031957
Repair 24" SW-1B Pipe, Minor Modification 2003-041
Technical Evaluation No. 2003-083, Rev. 1
Ebasco Piping Specification, Ebasco Coating Guide CP-25 Coating Steel with Coaltar Epoxy Coating
ENN-DC-150
10 CFR 50.65 Maintenance Rule Scoping Basis Document, Building and Structures, Rev. 6
ENN-DC-112, Rev. 7, VYNPS Containment Access Building Replacement
2003 Core Bore Impact on 24" SW-1B
Containment Access Building Replacement Engineering Request

REVIEW OF SAFETY EVALUATION REPORT CONFIRMATORY ITEMS

The following items were reviewed as a part of the License Renewal Inspection.

2.3.3.2.a Service Water System

Inspection Item 2.3.3.2.a-1

License renewal drawing LRA-G-191159-SH-01-0, at location H-11, depicts pipe section 2"-SW-566C within the scope of license renewal. The license renewal boundary flag for 2"-SW-566C is located on an unisolable section of pipe. The actual location of the license renewal scope boundary for this pipe section is not clear. Perform an inspection to ensure that the license renewal scope boundary for this pipe section satisfy the applicable requirements of 10CFR 54.4(a)(2).

Inspection Item 2.3.3.2.a-2

LRA section 2.1.2.1.2 (page 2.1-12) states in part “Nonsafety-related piping systems connected to safety-related systems were included up to the structural boundary or to a point that includes an adequate portion of the nonsafety-related piping run to conservatively include the first seismic or equivalent anchor” and “If isometric drawings were not readily available to identify the structural boundary, connected lines were included to a point beyond the safety/nonsafety interface, such as a base-mounted component, flexible connection, or the end of a piping run (such as a drain line).” The following are license renewal boundaries located at safety/nonsafety class breaks. It is not clear that the nonsafety-related piping systems were included up to the structural boundary or to a point that includes an adequate portion of the nonsafety-related piping run to include the first seismic or equivalent anchor. Perform an inspection to ensure that the license renewal scope boundary for these pipe sections satisfy the applicable requirements of 10CFR 54.4(a)(2).

a) LR drawing LRA-G-191159-SH-02-0, location E-8, downstream of valve 419.
b) LR drawing LRA-G-191159-SH-02-0, location E-7, downstream of valve 418.
c) LR drawing LRA-G-191159-SH-02-0, location E-7, downstream of valve 18.
d) LR drawing LRA-G-191159-SH-02-0, location C-3, downstream of valve 203.
e) LR drawing LRA-G-191159-SH-02-0, location J-3, downstream of valve 206.
f) LR drawing LRA-G-191159-SH-01-0, location E-10, downstream of valve 302.
g) LR drawing LRA-G-191159-SH-01-0, location D-11, downstream of valve 23C.
h) LR drawing LRA-G-191159-SH-01-0, location F-13, downstream of valve 22C.
I) LR drawing LRA-G-191159-SH-01-0, location F-14, downstream of valve 22D.
j) LR drawing LRA-G-191159-SH-01-0, location H-11, downstream of valve 20.
k) LR drawing LRA-G-191159-SH-01-0, location H-11, downstream of valve 20A.

<u>Response</u>

There are two types of physical interactions which must be considered when determining if nonsafety related components are in the scope of license renewal per 10CFR 54.4(a)(2). One type of physical interaction is potential spatial interaction with safety related components; the other is where a nonsafety related component is directly connected to a safety related component and is relied upon to provide structural support.

The applicant describes nonsafety components in the Service Water system in license renewal scope in Table 2.3.3.13-B, "Description of Nonsafety-Related System Components Subject to Aging Management Review Based on 10 CFR 54.4(a)(2) for Physical Interactions." For spatial interaction considerations the licensee has included the portion of the system in the service water pump area of the intake structure and the reactor building. For structural support considerations, the applicant has included components outside the safety class pressure boundary, yet relied upon to provide structural/seismic support for the pressure boundary.

The application describes the types of components which are included in the scope for 10 CFR 54.4(a)(2) and subject to aging management review in the service water system in Table 2.3.3-13-42. This table was developed by including all nonsafety-related portions of fluid systems which are located within a building containing safety-related components and all nonsafety-related piping connected to safety-related systems back to the structural boundary using an isometric drawing. In cases where an isometric drawing which depicts the structural boundary is not readily available, connected lines back to a point beyond the safety/nonsafety interface to a base-mounted component, flexible connection, or the end of a piping run (such as a drain line) in accordance with the response to RAI 2.1-2 were included.

Entergy's application does not include drawings and no official record of the exact end of the structural boundaries was maintained. Therefore, during the development of the aging management program for the service water system, the responsible engineer will need to apply the above methodology to be certain the appropriate portions of the system are included for structural support. When boundaries are developed in accordance with the methodology the applicant committed to follow, the result will be a conservative selection of components in scope for 10 CFR 54.4(a)(2).

Entergy determined which components are in scope under 10 CFR 54.4(a)(2) for potential spatial interaction using a conservative spaces approach by system. The licensee stated that if any nonsafety-related portion of a fluid system is located within a building containing safety-related components, the components within the system will be within the license renewal scope. Therefore, the applicant has committed to amend Table 2.3.3.13-B to include the turbine building.

2.3.3.12 John Deere Diesel System

<u>Inspection Item 2.3.3.12-1</u>

The John Deere diesel systems are installed in compliance with 10 CFR 50 Appendix R requirements. Due to a lack of available drawings and/or detailed description of the diesel equipment listed in Table 2.3.3-12 of the LRA, it is difficult to determine if any components subject to AMR may have been omitted from the table. It is recommended that both of the John

Deere diesel systems be inspected to assure all components subject to AMR are included in the list of Table 2.3.3-12 of the LRA. Determine how the fuel oil day tank and supply line interface with each diesel and whether any of their components should be included in the table. Perform an inspection to ensure that the license renewal scope boundaries for these components satisfy the applicable requirements of 10CFR 54.4(a)(3).

Response

The John Deere diesel system components are listed in Table 2.3.3-12 of the License Renewal Application. However, the supporting fuel oil day tank, fiberglass underground storage tank, and supply lines are listed in Table 2.3.3-6, "Fuel Oil (FO) System. The regional inspectors reviewed the John Deere diesel aging management program. Refer to the inspection report for the results of the review.

2.3.3.13.2 Augmented Off-Gas System

Inspection Item 2.3.3.132-1

The LRA states that the augmented off-gas (AOG) system is within the scope of license renewal based on 10 CFR 54.4(a)(2) because of the potential for physical interaction with safety-related components (Table 2.3.3.13-A). The determination of whether a component meets the criterion of 10 CFR 54.4(a)(2) for physical interactions is based on where it is located in a building and its proximity to safety-related equipment or where a structural/seismic boundary exists. This information is not provided on license renewal drawings nor was a detailed description provided in the LRA. Consequently, any omission of AOG components subject to AMR cannot be determined. Perform an inspection to ensure that the license renewal scope boundaries for this system satisfy the applicable requirements of 10CFR 54.4(a)(2) and all the components subject to AMR are included in Table 2.3.3-13-1.

Response

Table 2.3.3.13-B of the License Renewal Application states that the portion of the augmented off-gas (AOG) system associated with the plant stack loop seal is subject to aging management review based on 10 CFR 54.4(a)(2) for physical interactions. The inspector questioned the applicant where the boundaries of this portion of the system were located. Since the boundaries for the portion of the system as described in Table 2.3.3.13-B were not well defined, the applicant committed to amend the table to read "portion of the system inside the plant stack." The inspector walked down the remainder of the system and confirmed that no other portions of the system should have been included based on 10 CFR 54.4(a)(2).

2.3.3.13.e Circulating Water System

Inspection Item 2.3.3.13e-1

The LRA states that the Circulating Water system is within the scope of license renewal based on the potential for physical interaction with safety-related components (10 CFR 54.4(a)(2)) (Table 2.3.3.13-A). The applicant did not provide drawings highlighting in-scope 10 CFR 54.4(a)(2) components stating that the drawings would not provide significant additional information since the drawings do not indicate proximity of components to safety-related

equipment and do not identify structural/seismic boundaries. Without license renewal drawings and/or detailed description of the Circulating Water system, the omission of components subject to AMR cannot be determined (see Table 2.3.3-13-9 of the LRA). Perform an inspection to ensure that the LR scope boundaries for this system satisfy the applicable requirements of 10CFR 54.4(a)(2) and all the components subject to AMR are included in Table 2.3.3-13-9.

Response

The methodology that the applicant has committed to follow in determining which components are in scope under 10 CFR 54.4(a)(2) for potential spacial interaction is conservative and similar to the preventative option described by Section 6 of Appendix F of NEI 95-10, "Industry Guideline for Implementing the Requirements of 10 CFR 54.4." Specifically, the applicant stated that if any nonsafety-related portion of a fluid system is located within a building containing safety-related components, the components within the system will be within the license renewal scope.

The methodology that the applicant has committed to follow in determining which components are in scope under 10 CFR 54.4(a)(2) for structural purposes follow a specific set of criteria. Specifically, the nonsafety-related piping connected to safety-related systems will be traced back to the structural boundary using isometric drawings which depict the structural boundary. In cases where an isometric drawing which depicts the structural boundary is not readily available, the responsible engineer will include connected lines back to a point beyond the safety/nonsafety interface to a base-mounted component, flexible connection, or the end of a piping run (such as a drain line) in accordance with the response to Request for Additional Information (RAI) 2.1-2 of Amendment 8 of the license renewal application (ADAMS Accession Number ML062270057).

Entergy's application does not include drawings, and no official record of the exact end of the structural boundaries was maintained. Therefore, during the development of the aging management program for the service water system, the responsible engineer will need to apply the above methodology to be certain the appropriate portions of the system are included for structural support. When boundaries are developed in accordance with the methodology the applicant committed to follow, the result will be a conservative selection of components in scope for 10 CFR 54.4(a)(2). See response to Inspection Items 2.3.3.2.a-1 and 2.3.3.2.a-2.

2.3.3.13.m Reactor Water Cleanup System

Inspection Item 2.3.3.13m-1

The LRA states that the Reactor Water Clean-up (RWCU) system is within the scope of license renewal based on 10 CFR 54.4(a)(2) because of the potential for physical interaction with safety-related components (Table 2.3.3.13-A). The determination of whether a component meets the criterion of 10 CFR 54.4(a)(2) for physical interactions is based on where it is located in a building and its proximity to safety-related equipment, or where a structural/seismic boundary exists. This information is not provided on license renewal drawings nor was a detailed description provided in the LRA. Consequently, any omission of RWCU components subject to AMR cannot be determined. Perform an inspection to ensure that the license renewal scope boundaries for this system satisfy the applicable requirements of 10CFR

54.4(a)(2) and all the components subject to AMR are included in Table 2.3.3-13-36.

Response

The methodology that the applicant has committed to follow in determining which components are in scope under 10 CFR 54.4(a)(2) for potential spacial interaction is conservative and similar to the preventative option described by Section 6 of Appendix F of NEI 95-10, "Industry Guideline for Implementing the Requirements of 10 CFR 54.4." Specifically, the applicant stated that if any nonsafety-related portion of a fluid system is located within a building containing safety-related components, the components within the system will be within the license renewal scope.

The methodology that the applicant has committed to follow in determining which components are in scope under 10 CFR 54.4(a)(2) for structural purposes follow a specific set of criteria. Specifically, the nonsafety-related piping connected to safety-related systems will be traced back to the structural boundary using isometric drawings which depict the structural boundary. In cases where an isometric drawing which depicts the structural boundary is not readily available, the responsible engineer will include connected lines back to a point beyond the safety/nonsafety interface to a base-mounted component, flexible connection, or the end of a piping run (such as a drain line), in accordance with the response to Request for Additional Information (RAI) 2.1-2 of Amendment 8 of the license renewal application (ADAMS Accession Number ML062270057).

Entergy's application does not include drawings, and no official record of the exact end of the structural boundaries was maintained. Therefore, during the development of the aging management program for RWCU system, the responsible engineer will need to apply the above methodology to be certain the appropriate portions of the system are included for structural support. When boundaries are developed in accordance with the methodology the applicant committed to follow, the result will be a conservative selection of components in scope for 10 CFR 54.4(a)(2). See response to Inspection Items 2.3.3.2.a-1 and 2.3.3.2.a-2.