VERIFICATION OF VYNPS LICENSE RENEWAL PROJECT REPORT

Title of Report: **Aging Management Review of the Reactor Vessel Internals**

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This report documents evaluations related to the VYNPS license renewal project. Signatures certify that the report was prepared, checked and reviewed by the License Renewal Project Team in accordance with the VYNPS license renewal project guidelines and that it was approved by the ENI License Renewal Project Manager and the VYNPS Manager, Engineering Projects.

License Renewal Project Team signatures also certify that a review for determining potential impact to other license renewal documents (based on previous revisions) was conducted for this revision.

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1.0 Introduction

1.1 Purpose

This report is part of the aging management review (AMR) of the integrated plant assessment (IPA) performed to extend the operating license of Vermont Yankee Nuclear Power Station (VYNPS). This report demonstrates the effects of aging on reactor vessel internals (RVI) passive mechanical subcomponents will be adequately managed so that the intended functions will be maintained consistent with the current licensing basis as required by 10CFR 54.21(a)(3). For additional information on the license renewal project and associated documentation, refer to the License Renewal Project Plan. (**Ref. 6.1.1**)

The purpose of this report is to demonstrate that the effects of aging on passive mechanical subcomponents of the VY reactor vessel internals will be adequately managed for the period of extended operation for license renewal. This includes the reactor vessel internals up to the interface with the reactor pressure vessel or other connecting system. The reactor pressure vessel and the nuclear system process barrier are addressed in AMRM-31 and AMRM-33, respectively. The specific boundaries covered by this aging management review report (AMRR) are defined in Section 2.0. The aging effects requiring management for reactor vessel internals subcomponents are identified in Section 3.0. Section 4.0 then evaluates if existing programs and commitments adequately manage those effects.

Applicable aging effects were determined using EPRI report 1003056, *Non-Class 1 Mechanical Implementation Guideline and Mechanical Tools*, herein after referred to as the Mechanical Tools (**Ref. 6.2.1**), and various Boiling Water Reactor Vessel & Internals Program BWRVIP reports (**Ref. 6.3**). The BWRVIP reports provide bases for identification of aging effects based on extensive boiling water reactor operating data and metallurgical analyses. The Mechanical Tools provide the bases for identification of aging effects based on specific materials and environments and documents confirmation of the validity of the aging effects through review of industry experience. The Mechanical Tools were not written to specifically address environments and materials in Class 1 systems. However, the Mechanical Tools are applicable where the materials and environments are the same as the non-Class 1 materials and environments. The reactor vessel internal subcomponents covered in this AMRR include material and environment evaluated in the Mechanical Tools.

Material and environment combinations not addressed in the Mechanical Tools, were addressed using other industry references, including NUREG-1801, *Generic Aging Lessons Learned (GALL) Report* (**Ref. 6.4.1**) and BWR Vessel and Internals Projects Reports **(Ref. 6.3)**.

This aging management review report (AMRR), in conjunction with EPRI report 1003056, documents the identification and evaluation of aging effects requiring management for mechanical components in the reactor vessel internals.

1.2 System Description

The design of the reactor vessel internals is in accordance with the intent of Section III of the ASME Boiler and Pressure Vessel Code. ASME Section III for Class A vessels is used as a guide to determine limiting stress intensities and cyclic loadings for the reactor vessel internals. The material used for fabrication of most of the reactor vessel internals is solution heat-treated unstabilized Type 304 austenitic stainless steel conforming to ASTM specifications. (Section 3.3.4 and 3.3.5.1 of **Ref. 6.1.2**)

The reactor vessel internals are installed inside the reactor pressure vessel to properly distribute the flow of coolant delivered to the vessel, to locate and support the fuel assemblies, and to provide an inner volume containing the core that can be flooded following a break in the nuclear system process barrier external to the reactor pressure vessel. The reactor vessel internals include the following subcomponents: (Section 3.3.4 of **Ref. 6.1.2**)

1. Control rod guide tubes
2. Core plate
3. Core spray lines
4. Differential pressure and standby liquid control line
5. Feedwater spargers
6. Fuel support pieces
7. Incore dry tubes, guide tubes, and local power range monitors (LPRM)
8. Jet pump assemblies and jet pump instrumentation
9. Shroud (including shroud stabilizers)
10. Shroud head and steam separator assembly
11. Shroud support
12. Steam dryer
13. Surveillance sample holders
14. Top guide
15. Vessel head spray line

None of the reactor vessel internals subcomponents are insulated, hence no insulation is discussed in this AMR.

1.2.1 Control Rod Guide Tubes

The control rod guide tubes extend from the top of the control rod drive (CRD) housings up through holes in the core plate. Each guide tube is designed as the lateral guide for a control rod and as the vertical support for a four lobed fuel support piece and the four fuel assemblies surrounding the control rod. The bottom of the guide tube is supported by the CRD housing which in turn transmits the weight of the guide tube, fuel support piece, and fuel assemblies to the reactor pressure vessel bottom head. (Section 3.3.4.3 of **Ref. 6.1.2**)

1.2.2 Core Plate

The core plate is part of the core support assembly. The core plate is a circular stainless steel plate stiffened with a rim and beam structure. The plate provides lateral support and guidance for the control rod guide tubes, peripheral fuel support pieces, and incore guide tubes. The entire assembly is bolted to a support ledge between the central and lower portions of the core shroud. (Section 3.3.4.1.3 of **Ref. 6.1.2**)

1.2.3 Core Spray Lines

Two 100 percent capacity core spray lines enter the reactor pressure vessel through two separate core spray nozzles. The lines divide (T-boxes) immediately inside the reactor pressure vessel. The two halves are routed to opposite sides of the reactor pressure vessel and are supported by clamps attached to the vessel wall. The header halves are then routed downward into the downcomer annulus and pass through the upper shroud immediately below the flange. The flow divides again as it enters the center of the semicircular sparger ring which is routed halfway around the inside of the upper shroud. The ends of the two sparger rings for each line are supported by slip-fit brackets designed to accommodate thermal expansion of the rings. The header routing and supports are designed to accommodate differential movement between the shroud and the vessel. The other core spray line enters the opposite side of the vessel and the sparger rings are at a slightly different elevation in the shroud. (Section 3.3.4.7 of **Ref. 6.1.2**)

1.2.4 Differential Pressure and Standby Liquid Control Line

The differential pressure and liquid control line serves a dual function within the reactor pressure vessel – to inject liquid control solution into the coolant stream and to sense the differential pressure across the core plate. The line enters the reactor pressure vessel at a point below the core shroud as two concentric pipes. In the lower plenum, the two pipes separate. The inner pipe terminates near the lower shroud with a perforated length (sparger) below the core plate assembly. It is used to sense the pressure below the core support during normal operation and to inject liquid control solution when required. The outer pipe terminates immediately above the core plate and senses the pressure in the region outside the fuel assembly channels. (Section 3.3.4.9 of **Ref. 6.1.2**)

1.2.5 Feedwater Spargers

The feedwater spargers are perforated stainless steel headers located in the mixing plenum above the downcomer annulus. A separate sparger is fitted into each feedwater nozzle and is shaped to conform to the curve of the vessel wall. Sparger end brackets are attached to vessel brackets to support the weight of the spargers, and wedge blocks position the spargers away from the vessel wall. (Section 3.3.4.6 of **Ref. 6.1.2**)

1.2.6 Fuel Support Pieces

The fuel support pieces are part of the core support assembly. They are of two basic types - peripheral and four lobed. The peripheral fuel support pieces, which are welded to the core plate, are located at the outer edge of the active core and are not adjacent to control rods. The four lobed fuel support pieces will each support four fuel assemblies, and are provided with orifice plates to assure proper coolant flow distribution to each fuel assembly. The four-lobed support pieces rest in the top of the control rod guide tubes and are supported laterally by the core plate. The control rods pass through slots in the center of the four lobed fuel support pieces. (Section 3.3.4.2 of **Ref. 6.1.2**)

1.2.7 Incore Dry Tubes, Guide Tubes, and Local Power Range Monitors

The incore flux monitor guide tubes are part of the core support assembly. They are extensions of the incore flux monitor housings that run from the lower plenum to the top guide of the core support assembly. These guide tubes provide lateral support for the incore flux monitors (Source, Intermediate, Power, and Transversing). A latticework of clamps, tie bars, and spacers give lateral support and rigidity to the guide tubes. The bolts and clamps are welded after assembly to prevent loosening during reactor operation. These tubes do not provide any pressure boundary. (Section 3.3.4.10 of **Ref. 6.1.2**) See the incore instrumentation discussion below for more details.

For the source range monitoring/intermediate range monitoring (SRM/IRM) detectors, the guide tube/housing contains the dry tube which in turn contains the traveling source/intermediate range probe. The dry tube is the pressure boundary. The dry tubes are inserted through the guide tubes and are held in place below the top guide by spring tension. The dry tubes are bolted to the incore housings under the vessel. (Section 7.5 of **Ref. 6.1.2**)

The local power range monitors (LPRM) have no dry tubes. The detector itself is the pressure boundary. Each detector includes four fixed position power range detectors. The power range detectors are inserted in the incore guide tubes and are held in place below the top guide by spring tension. Detectors are sealed (bolted) to the incore housing under the vessel.

Each LPRM also contains another tube, called either a guide tube or a calibration tube, for a transversing incore probe (TIP). These tubes are not part of the reactor coolant system (RCS) pressure boundary. However, the TIP guide tubes have extensions which continue outside the containment where the rest of the TIP hardware is located. Consequently each guide tube has a ball and a shear valve outside containment as containment isolation valves.

1.2.8 Jet Pump Assemblies and Jet Pump Instrumentation

The jet pump assemblies are located in two semicircular groups in the downcomer annulus between the core shroud and the reactor pressure vessel wall. Each stainless steel jet pump consists of a driving nozzle, suction inlet, throat or mixing section, and diffuser. The driving nozzle, suction inlet, and throat are joined together as a removable unit and the diffuser is permanently installed. High pressure water from the recirculation pumps is supplied to each pair of jet pumps through a riser pipe welded to the recirculation inlet nozzle thermal sleeve. A riser brace is welded to cantilever beams extending from pads on the reactor pressure vessel wall. (Section 3.3.4.4 of **Ref. 6.1.2**)

The jet pump diffuser is a gradual conical section changing to a straight cylindrical section at the lower end. The diffuser is supported vertically by the shroud support ring. The joint between the throat and the diffuser is a slip fit. The throat section is supported laterally by a bracket attached to the riser. The jet pump diffuser section is welded to the shroud support plate and provides a positive seal. (Section 3.3.4.4 of **Ref. 6.1.2**)

Jet pump instrumentation provides indication of jet pump flow during normal operation. Jet pump instrumentation internal to the vessel consists of tubing and brackets to carry pressures and differential pressures to the N8 instrumentation nozzles.

1.2.9 Shroud (including Shroud Stabilizers)

The core shroud is a stainless steel cylindrical assembly which provides a partition to separate the upward flow of coolant through the core from the downward recirculation flow. This partition separates the core region from the downcomer annulus, thus providing a floodable region following a recirculation line break. (Section 3.3.4.1.1 of **Ref. 6.1.2**)

To address the potential for core shroud weld cracking due to intergranular stress corrosion cracking four shroud stabilizers (tie rod assemblies) hold the shroud together. Radial restraints are provided at four elevations to limit the lateral movement of the shroud sections. The shroud hardware limits the displacement of the shroud such that the shroud will maintain its as-designed configuration during all identified operating, transient, and accident conditions. For details of the rods and radial restraints, see Appendix K to the Vermont Yankee UFSAR (**Ref. 6.1.2**)

1.2.10 Shroud Head and Steam Separator Assembly

The shroud head and steam separator assembly is bolted to the top of the upper shroud to form the top of the core discharge plenum. The steam exits from the top of the separator and rises up to the dryers. The separated water exits from under the separator cap and flows out between the standpipes, draining into the recirculation flow downcomer annulus. (Section 3.3.4.1.2 of **Ref. 6.1.2**)

1.2.11 Shroud Support

The shroud support assembly, in the lower plenum, consists of the shroud support ring pad (one circular pad), shroud support ring (core support baffle plate), the shroud support cylinder, shroud support legs (14) and shroud support feet (14). The shroud ring pad is welded to the vessel shell, and the feet are welded to the reactor vessel lower head. The shroud support ring is then welded to the pad, and the 14 legs are welded to the feet. (Section 4.0 of **Ref. 6.1.15**) The shroud support ring and the shroud legs/feet provide redundant support paths for the shroud. (Section K.3.1 of **Ref. 6.1.2**)

1.2.12 Steam Dryer

The steam dryer removes moisture from the wet steam. The wet steam leaving the steam separators flows across the dryer vanes and the moisture flows down through collecting troughs and tubes to the downcomer annulus. A skirt extends down into the water to form a seal between the wet steam plenum and the dry steam flowing out the top of the dryers to the steam outlet nozzles. The dryers rest on steam dryer support brackets attached to the reactor vessel wall and are restricted from lifting by steam dryer holddown brackets which are attached to the reactor vessel closure head over the top of the steam dryer lifting lugs when the head is in place. (Section 3.3.4.5 of **Ref. 6.1.2**)

1.2.13 Surveillance Sample Holders

The surveillance sample holders are welded baskets containing impact and tensile specimen capsules. The baskets hang from brackets on the inside wall of the reactor pressure vessel at the mid-height of the active core and at radial positions chosen to expose the specimens to the same environment and maximum neutron fluxes experienced by the reactor pressure vessel itself, while at the same time avoiding jet pump removal interference. (Section 3.3.4.12 of **Ref. 6.1.2**)

1.2.14 Top Guide

The top guide is part of the core support assembly. The top guide is formed by a series of stainless steel beams joined at right angles to form square openings. Each opening provides lateral support and guidance for four fuel assemblies. Holes are provided at the bottom of the beams to anchor the incore guide tubes. The top guide is positioned by alignment pins which fit into radial slots in the top of the shroud and is secured to the shroud by holddown retainers or latches. (Section 3.3.4.1.4 of **Ref. 6.1.2**)

Per BWRVIP-15 (**Ref. 6.3.3**) and BWRVIP-26 (**Ref. 6.3.6**) the VYNPS top guide is of the BWR-4 design with aligner pins, holddown devices and reinforcement blocks.

1.2.15 Vessel Head Spray

The vessel head cooling spray consists of a spray head mounted on a short length of pipe with a flange at the other end of the pipe. The flange is bolted to a mating flange on reactor vessel head nozzle N6A. The head spray line is no longer connected for use, it is blank flanged at the vessel nozzle. (Section 3.3.4.8 of **Ref. 6.1.2**)

1.3 System and Component Intended Functions

As described in section 3.3.1 of the UFSAR, the power generation objectives of the reactor vessel internals are the following:

a. Maintain partitions between regions within the reactor vessel to provide proper coolant distribution, thereby allowing power operation without fuel damage due to inadequate cooling.
b. Provide positioning and support for the fuel assemblies, control rods, incore flux monitors, and other vessel internals to assure that normal control rod movement is not impaired.

In addition to the reactor vessel internals as intended by UFSAR Section 3.3.1, this AMRR includes the parts of the core spray lines inside the vessel, the differential pressure and standby liquid control line inside the vessel, the feedwater spargers, the incore dry tubes and LPRM, surveillance sample holders, and the vessel head spray line.

> Core spray is a core standby cooling system as discussed in UFSAR section 6. Per ENN-MS-S-009-VY (**Ref. 6.1.3**) the core spray lines must spray water over the fuel assemblies, not just get water into the reactor vessel. Thus, the core spray piping and spargers inside the vessel must distribute flow over the core, and have the function of flow distribution. There is a reactor coolant pressure boundary function in ENN-MS-S-009-VY for core spray, but it is met by components outside the reactor vessel that are not part of this review.

There is no safety function for the DP/SLC piping inside the reactor vessel. This position is explained in detail in BWRVIP-27 (**Ref. 6.3.7**). There is a safety function in UFSAR section 3.8.2 to deliver liquid to the reactor vessel to control reactivity, but it is met by the SLC piping outside the vessel.

The incore instrumentation system (dry tubes and LPRMs) form part of the reactor coolant system pressure boundary.

The feedwater spargers, surveillance capsule holders, and vessel head spray line have no license renewal functions.

The steam dryer has the function to maintain structural integrity, i.e. not to become a loose part. This function is based on industry operating experience as discussed in Section 3.4 of this report.

From the objectives and functions listed above, the following are license renewal intended functions for the reactor vessel internals subcomponents.

1. maintain a floodable volume
2. maintain system pressure boundary (incore dry tubes)
3. provide support for Criterion (a)(1) equipment
4. maintain structural integrity (i.e. don't become a loose part)
5. flow distribution (core spray lines only)

Refer to VYNPS Report LRPD-01, System and Structure Scoping Results (**Ref. 6.1.4**), for additional information on scoping and intended functions of systems and structures for license renewal.

2.0 Screening, Materials, and Environments

2.1 Component Evaluation Boundaries

Passive, long-lived subcomponents that perform a license renewal subcomponent intended function are subject to aging management review. This section reviews each subcomponent of the reactor vessel internals and determines whether or not it is subject to aging management review. If a component is NOT subject to aging management review, the basis is explained in the appropriate subsection. The evaluation boundaries for this AMR include the subcomponent groups identified in Section 1.2. The boundary is typically the weld to the reactor vessel pad for the internals component. The pad, and the weld of the pad to the vessel, is reviewed in AMRM-31, Aging Management Review of the Reactor Pressure Vessel.

The reactor vessel internals items that are subject to aging management review and are reviewed in this AMR include the control rod guide tubes, core plate, core spray lines in the vessel, jet pump assemblies, fuel support pieces, incore guide tubes, incore dry tubes, jet pump assemblies, shroud, shroud support, steam dryer, and top guide. Subcomponents discussed in Section 1.2 that are reviewed in other AMRs are noted in their respective subsection below.

The reactor vessel internals items that are not subject to aging management review are the ΔP/SLC line inside the vessel, feedwater spargers, local power range monitors, jet pump instrumentation inside the vessel, shroud head and steam separator, surveillance capsule holders, and vessel head spray.

Fuel assemblies are internal to the reactor vessel, but they are not subject to aging management review as they are replaced after a limited number of fuel cycles.

Control rod assemblies accomplish their function with a change in configuration and thus are not subject to aging management review in accordance with 10CFR54.21(a)(1)(i).

2.1.1 Control Rod Guide Tubes

The control rod guide tubes (CRGTs) support the fuel and the control rods and as such are subject to aging management review. The CRGTs are reviewed in this AMR.

The CRD housing is reviewed in AMRM-31, Aging Management Review of the Reactor Pressure Vessel. The CRD mechanisms and the rest of the CRD hydraulic system is reviewed in AMRM-33, Aging Management Review of the Reactor Coolant System Pressure Boundary.

2.1.2 Core Plate

The entire core support assembly, including the core plate, is subject to aging management review as it supports the core and control rods. The core plate is reviewed in this AMRR.

2.1.3 Core Spray Lines

The core spray lines are completely subject to aging management review and their function is to distribute flow across the core. The core spray lines inside the vessel are reviewed in this AMR.

The nozzle, thermal sleeve, and brackets attached to the vessel wall are covered in AMRM-31, Aging Management Review of the Reactor Pressure Vessel. The components within the vessel are covered in this AMRR. The remaining class 1 components of the core spray system are evaluated in AMRM-33, Aging Management Review of the Reactor Coolant System Pressure Boundary.

2.1.4 Differential Pressure and Standby Liquid Control Line

This line inside the reactor vessel has no safety related function and is not subject to aging management review. BWRVIP-06 determined that failure of this line would not have an adverse impact on achieving safe shutdown, and no short term nor long-term actions were necessary. The SLC line outside the vessel does have a safety function and is subject to aging management review. The nozzle is reviewed in AMRM-31, Aging Management Review of the Reactor Pressure Vessel. The remainder of the SLC system is reviewed in AMRM-01, Aging Management Review of the Standby Liquid Control System.

2.1.5 Feedwater Spargers

The feedwater lines inside the reactor vessel do not provide any safety function. BWRVIP-06-A (**Ref. 6.3.2**) as accepted by the NRC SER (**Ref. 6.4.8**) reviewed the failure consequences of this subcomponent and determined that disengagement of a feedwater sparger from the inlet nozzle could result in a jet impingement on the steam separators, but there would be no safety related subcomponents in the path of the jet. If the sparger were to fall, it could possibly impact core spray lines or jet pumps, but then would lodge in the annulus. Failure of a feedwater sparger would be quickly detected. Further, the report concluded (Section 4.2 of BWRVIP-06) that even if loose parts were generated “there is no significant safety concern from postulated loose parts.” Consequently this subcomponent is not subject to aging management review.

The feedwater line assemblies outside the reactor vessel are subject to aging management review, but are divided among AMRs. The nozzles, thermal sleeves, and brackets welded to the vessel are in AMRM-31, Aging Management Review of the Reactor Pressure Vessel. The remaining class 1 components of the feedwater system are evaluated in AMRM-33, Aging Management Review of the Reactor Coolant System Pressure Boundary.

2.1.6 Fuel Support Pieces

The fuel support pieces support the core (fuel) and as such are subject to aging management review. The fuel support pieces are reviewed in this AMR.

2.1.7 Incore Dry Tubes, Guide Tubes, and Local Power Range Monitors

The incore guide tubes provide support for the dry tubes and the LPRMs and as such are subject to aging management review. The incore guide tubes are reviewed in this AMR.

The dry tubes for the source and intermediate range monitors are inside the reactor vessel, but are part of the RCS pressure boundary. The dry tubes are subject to aging management review, and are reviewed in this AMR.

The incore housings, as attachments to the reactor vessel, are covered in AMRM-31, Aging Management Review of the Reactor Vessel.

The LPRM have limited lifetimes and are replaced as determined by OP-4407 (**Ref. 6.1.16**) based on calibration current measurements. The TIP guide tubes inside the LPRM are an integral part of the detectors and are also replaced when a detector is replaced. As short lived components, neither the detectors nor the TIP guide tubes are subject to aging management review.

See AMRM-20, Primary Containment Penetrations Aging Management Review, for evaluation of the containment integrity function of the TIP guide tubes and valves.

2.1.8 Jet Pump Assemblies and Jet Pump Instrumentation

The jet pump assemblies form part of the floodable volume around the core and as such are subject to aging management review. The jet pump assemblies are reviewed in this AMR.

The recirculation inlet nozzles, safe ends, and thermal sleeves are reviewed in AMRM-31, Aging Management Review of the Reactor Pressure Vessel. The rest of the recirculation system outside the vessel is reviewed in AMRM-33, Aging Management Review of the Reactor Coolant System Pressure Boundary.

The jet pump instrumentation inside the vessel has no license renewal function and as such is not subject to aging management review.

The jet pump instrumentation nozzles are reviewed in AMRM-31, Aging Management Review of the Reactor Pressure Vessel. The jet pump sensing lines outside the reactor vessel are reviewed in AMRM-33, Aging management review of the Reactor Coolant System Pressure Boundary.

2.1.9 Shroud (including Shroud Stabilizers)

The three shroud cylinders provide support to the core and provide the floodable volume for core cooling. The shroud tie rods and radial restraints are essential to the shroud performing its function. All sections of the shroud, including the shroud repair hardware, are subject to aging management review and are reviewed in this AMRR.

2.1.10 Shroud Head and Steam Separator Assembly

The steam separator assembly and shroud head do not provide any safety function. BWRVIP-06-A (**Ref. 6.3.2**) as accepted by the NRC SER (**Ref. 6.4.8**) reviewed the failure consequences of this subcomponent and determined that cracking to the extent of creating a loose part was unlikely to go undetected. Further, the report concluded (Section 4.2 of BWRVIP-06) that even if loose parts were generated "there is no significant safety concern from postulated loose parts." Recent industry operating experience (Section 3.4) has shown that loose parts generated by the steam dryers can reach the steam lines. However, any loose parts generated by the steam separators would be captured by the steam dryer, and would not reach the steam lines. The conclusion of BWRVIP-06-A therefore remains valid for the steam separators even considering recent operating experience. Consequently this subcomponent is not subject to aging management review.

2.1.11 Shroud Support

The shroud support assembly's sole function is support of the shroud and core, thus the entire assembly is subject to aging management review. This AMRR reviews the shroud support ring (core support baffle plate), the shroud support cylinder, and shroud support legs (14). The shroud support ring pad (1 circumferential pad) and the shroud feet (14) are welded to the vessel and as such are reviewed in AMRM-31, Aging Management Review of the Reactor Pressure Vessel.

2.1.12 Steam Dryer

The steam dryer does not provide any safety function. BWRVIP-06-A (**Ref. 6.3.2**) as accepted by the NRC SER (**Ref. 6.4.8**) reviewed the failure consequences of this subcomponent and determined that cracking to the extent of creating a loose part was unlikely. Further, the report concluded (Section 4.2 of BWRVIP-06-A) that even if loose parts were generated "there is no significant safety concern from postulated loose parts." However, recent industry experience (See Section 3.4 of this report.) has shown that cracking to the extent that generates loose parts can occur, especially after an extended power uprate. This recent operating experience also indicates that the loose parts so generated might interfere with the safety function of other components (MSIVs). Consequently the steam dryer is subject to aging management review as a non-safety component whose failure could affect the safety related function of another component. The license renewal function of the steam dryer is maintaining structural integrity, i.e. not becoming a loose part.

2.1.13 Surveillance Sample Holders

The surveillance capsule holders do not provide any safety function. BWRVIP-06-A (Ref. 6.3.2) as accepted by the NRC SER (Ref. 6.4.8) reviewed the failure consequences of this subcomponent and determined that a loose part such as a surveillance capsule holder is not expected to create an unsafe condition. Consequently this subcomponent is not subject to aging management review.

The surveillance holder bracket pads are welded to the reactor vessel and are reviewed in AMRM-31, Aging Management Review of the Reactor Pressure Vessel.

2.1.14 Top Guide

The top guide supports the fuel assemblies and other core components and as such is subject to aging management review. The top guide is reviewed in this AMR.

2.1.15 Vessel Head Spray

The vessel head spray has been disconnected and is no longer in service. It performs no safety function. Its failure has no impact on other safety related components similar to the feedwater spargers discussed above. Consequently the spray line and spray head inside the vessel are not subject to aging management review.

The spray vessel nozzle and blank flange are reviewed in AMRM-31, Aging Management Review of the Reactor Pressure Vessel.

2.2 Component Materials

This section lists the materials of construction for those subcomponents that were identified in Section 2.1 as subject to aging management review. Where available, detailed material specifications, and the reference from which they were attained, are given. Where only the material type, such as stainless steel or nickel-based alloy, is available, then that is given with its reference. Note that Section 3.3.4 of the UFSAR states “The material used for fabrication of most of the reactor vessel internals is solution heat treated, unstabilized Type 304 austenitic stainless steel, conforming to ASTM specifications.” Other materials of construction of the reactor vessel internals include nickel-based alloys and cast austenitic stainless steel (CASS).

2.2.1 Control Rod Guide Tubes

The control rod guide tubes are type 304 stainless steel per drawing 919D294 in GEK-9608. The control rod guide tube bases are cast austenitic stainless steel, either CF3 or CR8 per section 1.2 of NE 8067.

2.2.2 Core Plate

Item	Material Class	Material	Material Reference
Plate	Stainless Steel	A-268 Type 304	Dwg 5920-1101 (Ref. 6.1.26)
Beams	Stainless Steel	A-240 Type 304	Dwg 5920-1101 (Ref. 6.1.26)
Rim Bolts	Stainless Steel	Not specified	NE 8067 (Section 2.1.2)
Alignment Assemblies	Stainless Steel	A-276 Type 304	Dwg 5920-1101 (Ref. 6.1.26)
Alignment Bolts/Nuts	Stainless Steel	SA193, SA194	BWRVIP-15 Fig. 2.3.2.4

2.2.3 Core Spray Lines

The core spray lines (piping) inside the vessel is either type 304, 304L or 316L austenitic stainless steel per Section 2.0 of BWRVIP-18. Section 3.2.1 of BWRVIP-18 goes on to state that most of the core spray hardware in BWR 3-5s are type 304 stainless, though there is some 304L and 316L used in selected plants. In their SER on the cracking VY observed in two core spray collar to shroud welds, the NRC states “The internal CS piping is a 5-inch diameter, schedule 40 pipe composed of Type 304 stainless steel material.” (**Ref. 6.4.10**)

2.2.4 Differential Pressure and Standby Liquid Control Line

The differential pressure and standby liquid control line inside the reactor vessel is not subject to aging management review as determined in Section 2.1.4.

2.2.5 Feedwater Spargers

The feedwater spargers are not subject to aging management review as determined in Section 2.1.5.

2.2.6 Fuel Support Pieces

The fuel support pieces, both orificed and peripheral, are cast austenitic stainless steel (CASS) per Section 12.0 of NE 8067. BWRVIP-15 and BWRVIP-47 also suggest that both fuel support pieces are CASS.

2.2.7 Incore Dry Tubes, Guide Tubes, and Local Power Range Monitors

The dry tubes for the source and intermediate range monitors are made of Type 304 or Type 316 stainless steel per drawing 729E946 in GEK-9608.

The incore guide tubes are made of type 304 austenitic stainless steel based on Section 3.3.4 of the UFSAR.

The LPRM are not subject to aging management review as determined in Section 2.1.7.

2.2.8 Jet Pump Assemblies and Jet Pump Instrumentation

Section 3.3.4 of the UFSAR says most internals components are constructed of Type 304 stainless steel. The various subcomponents of the jet pumps are made of stainless steel, cast austenitic stainless steel, or nickel-based alloy. Various piece parts are identified in BWRVIP-41 (**Ref. 6.3.9**) and are listed below along with their given material of construction. The names of piece parts vary between references; names given on the table represent the most common names for these parts.

Subcomponent	Material Class	Specific Material	Material Reference
Riser Pipe	Stainless Steel	Type 304 pipe	BWRVIP-41, Table 2.3.4-1
Riser Elbow	Stainless Steel	Type 304 pipe	BWRVIP-41, Table 2.3.4-1
Riser Brace	Stainless Steel	Type 304L or 316L	BWRVIP-41, Table 2.3.1-1

Subcomponent	Material Class	Specific Material	Material Reference
Transition Piece	CASS[1]	Type 304 casting	NE 8067 Section 9.4 BWRVIP-41, Table 2.3.5-1
Suction Inlet Elbow	CASS	Type 304 Casting	BWRVIP-41, Table 2.3.6-1
Suction Inlet Nozzle	CASS	Type 304 Casting	BWRVIP-41, Table 2.3.6-1
Holddown beam	NBA	Inconel X750 Alloy X750	NE 8067 Section 9.2 BWRVIP-41, Table 2.3.2-1
Holddown bolt	Stainless Steel	Type 304 or 316L	BWRVIP-41, Table 2.3.2-1
Mixer flange	CASS[1]	Type 304 casting	NE 8067 9.10 BWRVIP-41, Table 2.3.7-1
Mixer throat (barrel)	Stainless Steel	Type 304 pipe	BWRVIP-41, Table 2.3.7-1
Mixer flare	CASS	Type 304 casting	BWRVIP-41, Table 2.3.7-1
Restrainer bracket	Stainless steel	Type 304 plate	BWRVIP-41, Table 2.3.8-1
Restrainer bracket wedge assemblies	Stainless Steel	Type 304 wrought	BWRVIP-41, Section 2.3.8.4
Diffuser collar	CASS	Type 304 casting	NE 8067 Section 9.10 BWRVIP-41, Table 2.3.9-1
Diffuser shell	Stainless Steel	Type 304 plate	BWRVIP-41, Table 2.3.10-1
Diffuser tailpipe	Stainless Steel	Type 304 pipe	BWRVIP-41, Table 2.3.10-1
Diffuser adapter (top piece)	Stainless Steel	Type 304	BWRVIP-41, Table 2.3.11-1 NE 8067 Section 9.1
Diffuser adapter (bottom piece)	NBA	Alloy 600	NE 8067, Section 9.11 BWRVIP-41, Table 2.3.11-1

The jet pump instrumentation inside the vessel is not subject to aging management review as determined in Section 2.1.8.

2.2.9 Shroud (including shroud stabilizers)

The core shroud is stainless steel per section 3.3.4.1.1 of the UFSAR. It is Type 304 stainless steel per section 3.2 of NE 8067 (**Ref. 6.1.15**).

The shroud radial restraints and top bracket are type 304 stainless steel, the spring rod and top adapter are nickel-based alloy, XM-19, the bottom adapter is nickel-based alloy X750. (Section K.7.1 of the UFSAR)

2.2.10 Shroud Head and Steam Separator

The shroud head and steam separator assembly is not subject to aging management review as discussed in Section 2.1.10.

2.2.11 Shroud Support

The shroud support assembly (ring, cylinder, and legs) is nickel-based alloy, Alloy 600, per section 4.1 of NE 8067. The access hole cover welds are Alloy 82/182 per section 4.3 of NE 8067.

2.2.12 Steam Dryer

The steam dryers are type 304 stainless steel per section 3.3.4 of the UFSAR and per GE specification 21A3317.

2.2.13 Surveillance Sample Holders

The surveillance capsule holders are not subject to aging management review as determined in Section 2.1.13.

2.2.14 Top Guide

The top guide assembly is stainless steel per section 3.3.4.1.3 of the UFSAR. The top guide assembly is type 304 or 304L stainless steel per section 2.1.2 of BWRVIP-25.

2.2.15 Vessel Head Spray

The vessel head spray line is not subject to aging management review as determined in Section 2.1.15.

2.3 Environments

The operating environments experienced by the reactor pressure vessel subcomponents are treated water and neutron fluence on internal surfaces and air-indoor (i.e., containment environment) on external surfaces.

2.3.1 Treated Water

The operating environment experienced by the majority of the reactor vessel internals is treated water on all surfaces, since they are completely contained within the reactor pressure vessel. The reactor coolant system water varies in temperature from less than 212 degrees in small, no flow areas to greater than 500 degrees in the vessel interior. There are four environments based on temperature for treated water.

Treated water. This implies cold (<212 °F) treated water. At this low temperature, moisture may be present on the outside surface of the material.

Treated water greater than 220 ºF. Above this threshold, carbon steel is susceptible to fatigue (Appendix H of **Ref. 6.2.1).**

Treated water greater than 270 ºF. Above this threshold, stainless steel is susceptible to fatigue (Appendix H of **Ref. 6.2.1).**

Treated water greater than 482 ºF. Above this threshold, cast austenitic stainless steel (CASS) is susceptible to reduction of fracture toughness due to thermal embrittlement (Section 3.3.1 of **Ref. 6.2.1)**

For purposes of this report, steam from treated water is considered treated water. VYNPS water chemistry requirements are specified in the Updated Final Safety Analysis Report (UFSAR). Refer to Section 4.1.3 for more information regarding the VYNPS Water Chemistry Program.

2.3.2 Neutron Fluence (Internal)

Subcomponents of the reactor vessel internals immediately adjacent to the core are exposed to neutron fluence in excess of $1X10^{17}$ n/cm^2. The possible effects of this fluence are identified in various BWRVIP documents and are discussed in Section 3.2

2.3.3 Air-indoor (External)

The subject mechanical subcomponents of the reactor vessel internals are located inside the reactor vessel. Most of the items are exposed only to treated water with no external environment.

The exceptions are the incore dry tubes, which are exposed to air-indoor (primary containment atmosphere) on the inside of the tubes. The atmosphere is inerted with nitrogen to a maximum oxygen level of 4% (Section 5.2.6.2 of **Ref. 6.1.1**), making the atmosphere less corrosive than natural air. The ambient temperature ranges from 135°F to165°F. (Table 5.2-1 of **Ref. 6.1.1**). The external surface of the incore dry tubes normally exceeds 212 degrees F and thus is not subject to condensation. For purposes of this AMR, incore dry tubes are assumed to be exposed to air-indoor.

For purposes of this AMR, to maintain consistency with the rest of the reactor coolant system, the internal environment is treated water and the external environment is air-indoor; even though the air is on the ID of the tubes and the treated water is on the OD of the tubes.

3.0 Aging Effects Requiring Management

EPRI report 1003056 (**Ref. 6.2.1**) (Table 4-1) and other industry documents are used in this section to identify and evaluate aging effects. The following aging effects and associated mechanisms were for the material / environment combinations present in the reactor vessel internals.

loss of material	erosion/flow-accelerated corrosion (FAC), crevice and pitting corrosion, microbiologically influenced corrosion (MIC)
cracking	fatigue, flaw growth, flow induced vibration, and stress corrosion (including intergranular stress corrosion and irradiation assisted stress corrosion)
reduction in fracture toughness	thermal embrittlement, radiation embrittlement.
loss of preload	identified by the BWRVIP program as applicable to the core plate rim hold down bolts

For additional information on aging effects, refer to EPRI report 1003056. (Ref. 6.2.1)

Several aging mechanisms can be eliminated based on the material and environment combinations of the reactor vessel internals. These mechanisms are discussed here, and not addressed under each material/environment combination.

Erosion and flow-accelerated corrosion are not applicable to vessel internals. There are no particulate impurities in the reactor coolant to cause erosion, even in the jet pumps where velocity is high. The internals are constructed of stainless steel or nickel-based alloys that are resistant to erosion and flow accelerated corrosion.

MIC is not applicable to the reactor vessel internals due to the high purity of the reactor coolant and the absence of microbes.

The following sections document the determination of aging effects requiring management based on specific subcomponent materials and environments. The review was performed for groups of subcomponents with similar operating environments and materials of construction. The AMR results are tabulated in Attachment 1.

3.1 <u>Stainless Steel and Nickel-based Alloy Exposed to Treated Water</u>

All reactor vessel internals subcomponents are made of either stainless steel or nickel-based alloy. The reactor vessel internals subcomponents of stainless steel are the core shroud, core plate, top guide, control rod guide tubes, jet pump assemblies (partial), core spray lines, and incore flux monitor guide tubes and dry tubes. Nickel-based alloy subcomponents are the core shroud support, jet pump hold-down beam, and the welded core shroud access cover. CASS subcomponents are the fuel support pieces, CRD guide tube bases, and several pieces of the jet pump assemblies. For a complete list of subcomponents and materials, see Attachment 1.

3.1.1 Loss of Material

Stainless steel and nickel-based alloys are inherently immune to general corrosion. Due to physical configuration or small surface defects, system fluid contaminants could concentrate in crevices in the subcomponents. With a high enough concentration of contaminants in the treated water, the stainless steel and nickel-based alloy surfaces may be susceptible to loss of material by pitting and crevice corrosion. Therefore, loss of material (pitting corrosion and crevice corrosion) is an aging effect requiring management for stainless steel and nickel-based alloy subcomponents.

3.1.2 Cracking – Fatigue

Cracking due to thermal fatigue is an aging effect requiring management since the operating temperature exceeds 270ºF. The analysis of fatigue is a time-limited aging analysis (TLAA); for more information on TLAA see Section 4.2.

3.1.3 Cracking – Other than Fatigue

Service loads may result in the growth of pre-service flaws **(Ref. 6.4.3)** or initiation and growth of service-induced flaws. The most susceptible locations for flaw initiation and growth are welded joints. Susceptibility is due to the variations in residual stresses and mechanical properties resulting from the various constituent zones (e.g., composite, unmixed, and heat-affected) within the joint. Therefore, cracking (initiation and growth) is an aging effect requiring management for the period of extended operation.

Cracking from stress corrosion (SCC) is an aging effect that has been observed in BWR internals. Cracking from SCC is an aging effect requiring management for stainless steel and nickel-based alloy components.

Cracking due to flow induced vibration is applicable to the steam dryer based on operating experience (Section 3.4)

3.1.4 Reduction of Fracture Toughness

Under certain conditions, stainless steels and nickel-based alloys are susceptible to an increase in material strength and resultant decrease in low cycle fatigue resistance known as reduction of fracture toughness. Reduction of fracture toughness may be due to radiation embrittlement or thermal embrittlement.

Reduction in fracture toughness due to thermal aging is an aging effect requiring management which is applicable to components fabricated from CASS. The jet pump castings, CRD guide tube bases, and the fuel support pieces are fabricated from CASS and thus are subject to thermal embrittlement. Section 2.2.3 of BWRVIP-41 makes the following statement about the jet pump assemblies.

> “For the jet pump assembly, which is constructed using several cast subcomponents, thermal embrittlement is a potential degradation mechanism. . . . The aging mechanisms also induce embrittlement in duplex stainless steels (e.g. castings). . . . It is important to note that thermal embrittlement does not in itself cause cracking to occur. It

> reduces the structural margin of a material in resisting propagation of cracks due to other initiators like IGSCC or fatigue. In addition, since the cast subcomponents are made in one piece and have no welds, they are not expected to crack; nor is there any particular crack orientation preference."

Thus, reduction in fracture toughness due to thermal embrittlement is an aging effect requiring management for CASS RV internals items.

3.1.5 Loss of Preload

BWRVIP-25 identifies loss of preload of the core plate rim hold down bolts as an aging effect. This aging effect is evaluated for 40 years by a TLAA prepared in the original BWRVIP-25 and the TLAA is extrapolated to 60 years by Appendix B to BWRVIP-25. For additional information on TLAA refer to Section 4.2 of this report and to report LRPD-03, TLAA and Exemption Evaluation Results.

3.2 <u>Stainless Steel and Nickel-Based Alloy Exposed to Neutron Fluence</u>

3.2.1 Reduction of Fracture Toughness

Cast austenitic stainless steel (CASS) is susceptible to reduction of fracture toughness due to radiation embrittlement at the fluence levels received by BWR vessel internals. Thus, reduction in fracture toughness due to radiation embrittlement is an aging effect requiring management for CASS components.

3.2.2 Irradiation-Assisted Stress Corrosion Cracking

The top guide assembly has been identified as being susceptible to irradiation-assisted stress corrosion cracking (IASCC). The onset of IASCC is a time-limited aging analysis (TLAA) addressed in BWRVIP-26. For additional information on TLAA refer to Section 4.2 of this report and to report LRPD-03, TLAA and Exemption Evaluation Results.

3.3 <u>Stainless Steel and Nickel-Based Alloy Exposed to Air-indoor</u>

Only the incore (source and intermediate) dry tubes are exposed to air-indoor. In this case the air is inside the tubes while reactor treated water is outside the tubes. These tubes are made of Type 304 or Type 316 stainless steel. Stainless steel exposed to air (primary containment) has no associated aging effects requiring management.

The following aging effects are identified in Appendix E, Table 4-1, of the Mechanical Tools (**Ref. 6.2.1**) as those that require evaluation for the external surfaces of stainless steel components exposed to air-indoor:

- Loss of Material — wear, fretting, MIC, pitting and crevice corrosion
- Cracking — stress corrosion

However, the relevant conditions do not exist in the external environment of the incore dry tubes for these aging effect(s) to occur.

- Wear and fretting are not applicable as there is no continuous wear against the inside of the incore dry tubes. (The source/intermediate range detectors are normally withdrawn during power operation, only being used during low power startup/shutdown.)
- MIC, pitting corrosion, crevice corrosion and stress corrosion cracking are not applicable to the external surface of the incore dry tubes as the temperature is normally above 212 °F, is not subject to wetting, and not exposed to an aggressive chemical species.

Thus, there are no aging effects requiring management identified for incore dry tube external surfaces exposed to air-indoor.

3.4 Operating Experience

The review of site-specific and recent industry operating experience, documented in LRPD-05, Operating Experience Review Results **(Ref. 6.1.8)**, did not identify any aging effects applicable to the reactor vessel internals not addressed in this aging management review report.

There are several conditions in the VYNPS reactor vessel internals that have been found by inspections and testing. These conditions are consistent with industry experience, which has identified similar conditions at other BWRs.

Ultrasonic inspections conducted in 1996 identified intergranular stress corrosion cracking at the coupling where the core spray piping penetrates the core shroud (see **Ref. 6.1.23**). The NRC issued an SER approving one cycle of operation, with the requirement that the two collar welds with flaws be re-inspected (with satisfactory results) prior to startup from the refueling outage or an acceptable repair be implemented (see **Ref. 6.1.24**). The inspections were performed during the 1998 refueling outage with satisfactory results. (**Ref. 6.1.25**)

A review of industry operating experience (**Ref. 6.2.2**, **Ref. 6.2.3**, and **Ref. 6.2.4**) identified loss of structural integrity (i.e. becoming loose parts) of steam dryers as an aging effect requiring management that was not identified by the Mechanical Tools. Preliminary indications are that steam dryer cracking is due to flow induced vibrations (FIV) associated with the higher steam flows after extended power uprates. This recent operating experience also indicates that the loose parts so generated might interfere with the safety function of other components (MSIVs). This operating experience has resulted in the steam dryers being subject to aging management review even though other reference documentation concludes they are not.

4.0 Demonstration That Aging Effects Will Be Managed

Section 2.0 described the subcomponents within the reactor vessel internals that are subject to aging management review. For those subcomponents, Section 3.0 documented the determination of aging effects requiring management. The aging management review is completed by demonstrating that existing programs, when continued into the period of extended operation, can manage the aging effects identified in Section 3.0. No further action is required for license renewal when the evaluation of an existing program demonstrates that it is adequate to manage the aging effect such that corrective action may be taken prior to loss of system intended functions. Alternately, if existing programs cannot be shown to manage the aging effects for the period of extended operation, then action will be proposed to augment existing programs, or create new programs to manage the identified effects of aging.

Demonstration for the purposes of the aging management review is accomplished by establishing a clear relationship among

1. the subcomponents under review,
2. the aging effects on these items caused by the material-environment-stress combinations which, if undetected, could result in the loss of the intended function such that the system could not perform its function(s) within the scope of license renewal in the period of extended operation, and
3. the credited aging management programs (AMP) whose actions serve to preserve the system intended function(s) for the period of extended operation.

Attachment 1 lists the reactor vessel internals subcomponents subject to aging management review and identifies the aging effects requiring management for the material and environment combinations.

Several aging management programs, in combination will manage the effects of aging precluding the loss of the intended functions of the system components. Section 4.1 discusses these programs in more detail and provides the clear relationship between the subcomponent, the aging effect and the aging management program actions which preserve the intended functions for the period of extended operation. For a comprehensive review of the programs credited for the license renewal of VYNPS and a demonstration of how these programs will manage the aging effects, refer to VYNPS Report LRPD-02, Aging Management Program Evaluation Results. (**Ref. 6.1.5**)

Section 4.1.2 identifies applicable time-limited aging analyses. For more detail on TLAA see LRPD-03, TLAA and Exemption Evaluations (**Ref. 6.1.6**) and LRPD-04, Time-Limited Aging Analyses – Metal Fatigue (**Ref. 6.1.7**).

4.1 <u>Aging Management Programs</u>

4.1.1 Boiling Water Reactor (BWR) Vessel Internals Program

The BWR Vessel Internals Program is a summary program that includes all inspections and flaw evaluations for the reactor vessel internals, including the inspections required by ASME Section XI, Inservice Inspection and the NRC approved boiling water reactor vessel and internals project

(BWRVIP) documents. Monitoring and controlling reactor coolant water chemistry is in accordance with the Water Chemistry Control – BWR Program.

The BWR vessel internals program is credited with managing the aging effects of

- cracking due to flaw growth and stress corrosion cracking
- loss of material due to pitting and crevice corrosion
- loss of material due wear of the jet pump wedges
- reduction of fracture toughness due to radiation embrittlement.

This program applies to internals items fabricated from wrought and cast austenitic stainless steel, nickel-based alloy, and their connecting welds. Numerous inspections and flaw evaluations have been performed and additional inspections are scheduled. (**Ref. 6.1.15**). For additional information on the BWR vessel internals program, refer to VYNPS Report LRPD-02, Aging Management Program Evaluation Results.

4.1.2 Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS)

The Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS) Program is an inspection and flaw evaluation program that is implemented via the BWR vessel internals program. The CASS Program is credited with managing reduction of fracture toughness of cast austenitic stainless steel components in the vessel internals. Individual CASS components are evaluated and scheduled for supplemental inspection in accordance with approved BWRVIP documents. Flaw evaluations and flaw acceptance criteria are also per approved BWRVIP documents. For more information on the Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS) program, see LRPD-02, Aging Management Program Evaluation Results.

4.1.3 Water Chemistry Control - BWR

Reactor vessel internals subcomponents exposed to treated water that are subject to aging management review include forged and cast stainless steel subcomponents, nickel-based alloy subcomponents, and welds. The Water Chemistry Control –BWR Program will mitigate loss of material due to crevice corrosion and pitting corrosion, and cracking by SCC for these components.

The VYNPS Water Chemistry Control Program optimizes the primary water chemistry to minimize the potential for loss of material and cracking by limiting the levels of contaminants in the RCS. For additional information on the water chemistry control – BWR program, see VYNPS Report LRPD-02, Aging Management Program Evaluation Results.

4.2 <u>Time-limited Aging Analyses</u>

Several time-limited aging analyses (TLAA) were identified as applicable to the reactor vessel internals during the preparation of this AMR. The identified TLAA are

- fatigue evaluations of metallic subcomponents

- BWRVIP-25 analyzes the loss of preload of the core plate hold down bolts for 40 years and for 60 years

For additional information refer to LRPD-03, TLAA and Exemption Evaluation Results, (**Ref. 6.1.6**) for the evaluation of TLAA, and to RPD-04, TLAA – Metal Fatigue, (**Ref. 6.1.7**) for the evaluation of metal fatigue for the period of extended operation.

5.0 Summary and Conclusions

The following aging management programs address the aging effects requiring management for the reactor vessel internals

BWR Vessel Internals Program

Thermal and Neutron embrittlement of Cast Austenitic Stainless Steel - CASS

Water Chemistry Control – BWR

For additional review of the programs credited for the license renewal of VYNPS, see VYNPS Report LRPD-02, Aging Management Program Evaluation Results.

For additional review of the TLAA credited for the license renewal of VYNPS, see VYNPS Reports LRPD-03, TLAA and Exemption Evaluation Results and LRPD-04, TLAA - Mechanical Fatigue.

Attachment 1 contains the aging management review results for the reactor vessel internals subcomponents.

In conclusion, the programs described in Section 4.0 will provide reasonable assurance that the effects of aging on the VYNPS reactor vessel internals will be managed such that the intended functions will be maintained consistent with the current licensing basis throughout the period of extended operation.

6.0 References

6.1 VYNPS References

6.1.1 VYNPS Report LRPG-01, Vermont Yankee NPS License Renewal Project Plan, Revision 0.

6.1.2 VYNPS Updated Final Safety Analysis Report, Revision 18.

6.1.3 ENN-MS-S-009-VY, Vermont Yankee Site Specific Guidance and System Safety Function Sheets, Revision 0, 03/22/2005

6.1.4 LRPD-01, System and Structure Scoping Results

6.1.5 LRPD-02, Aging Management Program Evaluation Results

6.1.6 LRPD-03, TLAA & Exemption Evaluations Results

6.1.7 LRPD-04, TLAA – Metal Fatigue

6.1.8 VYNPS Report LRPD-05, Operating Experience Review Results, Revision 0.

6.1.9 VYNPS Report AMRM-31, Aging Management Review of the Reactor Pressure Vessel Revision 1

6.1.10 VYNPS Report AMRM-33, Aging Management Review of the Reactor Coolant Pressure Boundary Revision 1

6.1.11 VYNPS Report AMRM-20, Aging Management Review of the Primary Containment Penetrations

6.1.12 VYNPS Procedure PP 7027, Reactor Vessel Internals Management Program Rev 1, 9/27/2002

6.1.13 GEK-9608, Operation and Maintenance Instructions, Reactor Assembly for Vermont Yankee Nuclear Power Station December, 1970 Program Implementing Procedure, Rev 3, 12/02

6.1.14 VY Procedure PP 7015, Rev. 3, September, 2003, Vermont Yankee Inservice Inspection Program

6.1.15 VY Procedure NE 8067, Revision 1, September 27, 2002, Reactor Vessel Internals Inspection Details

6.1.16 VYNPS Procedure OP-4407, Rev. 11, 1/17/02, LPRM Lifetime Management

6.1.17 Piping Specification, VYNP-V1-III-P-1, November 13, 1967; Ebasco Specification 62-65T, General Power Piping

6.1.18 Licensed Operator Training Program Student Handout, LOT-00-299H, Rev. 0, September, 2001

6.1.19 VY Procedure OP4612, Sampling and Treatment of the Reactor Water System Rev 23, September, 2000

6.1.20 VY Procedure OP 2199, Hydrogen Water Chemistry System Rev 1, September 2003

6.1.21 NVY 98-153, October 26, 1998, R.P. Croteau, Nuclear Regulatory Commission, to G. A. Maret, VYNPC, Jet Pump riser circumferential weld inspections at Vermont Yankee Nuclear Power Station (TAC NO. MA1681) (SER attached)

6.1.22 General Electric Specification 21A3317, Revision 0, 6/16/66, Standard Requirements for Steam Dryer Units

6.1.23 Calculation VYC-1536, VY Core Spray Annulus Piping Flaw Evaluation

6.1.24 USNRC SER for Related to Flaw Evaluation of Core Spray Internal Piping, dated November 20, 1996

6.1.25 AP 0028 Commitment NVY96176_05, Reinspect 5 Core Spray (CS) piping welds that were visually in the 96 Outage. Also visually reinspect 2 core spray piping welds that exhibited possible cracking. See Letter BVY 97-123 for details.

6.1.26 Drawing 5920-1101 (TEK B.327B), Revision 2, Core Support

6.2 Industry Documents

6.2.1 EPRI report 1003056, Revision 3, *Non-Class 1 Mechanical Implementation Guideline and Mechanical Tools*, November 2001 (a.k.a. the Mechanical Tools).

6.2.2 GE Services Information Letter (SIL) Number 644, *BWR/3 steam dryer failure*, August 21, 2002

6.2.3 GE Services Information Letter (SIL) Number 644, Supplement 1, *BWR Steam Dryer Integrity*, September 5, 2003

6.2.4 GE Services Information Letter (SIL) Number 644, Revision 1, *BWR steam dryer integrity*, November 9, 2004

6.3 Boiling Water Reactor Vessel and Internals Project (BWRVIP) documents

6.3.1 BWRVIP-03, *BWR Vessel and Internals Project Reactor Pressure Vessel and Internals Examination Guidelines*, Revision 3, EPRI Report TR-105696-R1, December, 2000

6.3.2 BWRVIP-06-A, *Safety Assessment of BWR Reactor Internals* EPRI Report 1006598, March 2002

6.3.3 BWRVIP-15, *Configurations of Safety-Related BWR Reactor Internals*, EPRI Report TR-106368, March 1996

6.3.4 BWRVIP-18, *BWR Core Spray Internals Inspection and Flaw Evaluation Guidelines*, EPRI Report TR-106740, July, 1996

6.3.5 BWRVIP-25, *BWR Core Plate Inspection and Flaw Evaluation Guidelines*, EPRI Report TR-107284, December 1996

6.3.6 BWRVIP-26, *BWR Top Guide Inspection and Flaw Evaluation Guidelines*, EPRI Report TR-107285, December 1996

6.3.7 BWRVIP-27, *BWR Standby Liquid Control System/Core Plate ΔP Inspection and Flaw Evaluation Guidelines*, EPRI Report TR-107286, April 1997

6.3.8 BWRVIP-38, *BWR Shroud Support Inspection and Flaw Evaluation Guidelines*, EPRI Report TR-108823, September 1997

6.3.9 BWRVIP-41, *BWR Jet Pump Assembly Inspection and Flaw Evaluation Guidelines*, EPRI Report TR-108728, October 1997

6.3.10 BWRVIP-42, *LPCI Coupling Inspection and Flaw Evaluation Guidelines*, EPRI Report TR-108726, December 1997

6.3.11 BWRVIP-47, *BWR Lower Plenum Inspection and Flaw Evaluation Guidelines*, EPRI Report TR-108724, February 1998

6.3.12 BWRVIP-49, *Instrument Penetration Inspection and Flaw Evaluation Guidelines*, EPRI Report TR-108695, March 1998

6.3.13 BWRVIP-62, *Technical Basis for Inspection Relief for BWR Internal Components with Hydrogen Injection*, EPRI Report 108705, December 1998

6.3.14 BWRVIP-75, *BWR Vessel and Internals Project Technical Basis for Revisions to Generic Letter 88-01 Inspection Schedules (BWRVIP-75)*, EPRI Report TR-113932, October 1999

6.3.15 BWRVIP-76, *Core Shroud Inspection and Flaw Evaluation Guidelines*, EPRI Report TR-114232, November 1999

6.3.16 BWRVIP-79, *BWR Water Chemistry Guidelines – 2000 Revision* EPRI Report TR-103515-R2, Final Report, February 2000.

6.4 Codes, Regulations, and NRC documents

6.4.1 NUREG-1801, Volumes 1 and 2, *Generic Aging Lessons Learned (GALL) Report*, U.S. Nuclear Regulatory Commission, April 2001.

6.4.2 NUREG-1803, *Safety Evaluation Report Related to the License Renewal of the Edwin I. Hatch Nuclear Plant, Units 1 and 2*, December, 2001.

6.4.3 *Nondestructive Examination Standards, Technical Basis and Development of Boiler and Pressure Vessel Code, ASME Section XI, Division 1*, EPRI-NP-1406-SR, May 1980.

6.4.4 Code of Federal Regulations, Title 10, Part 50, Appendix G: *Fracture Toughness Requirements*, December 19, 1995.

6.4.5 Code of Federal Regulations, Title 10 Part 54, *Requirements for Renewal of Operating Licenses for Nuclear Power Plants*

6.4.6 NRC Letter, W.H. Bateman to C. Terry, BWRVIP Chairman, *NRC Approval Letter BWRVIP-06-A*, September 16, 2003

6.4.7 US Nuclear Regulatory Committee, Generic Letter 94-03, *Intergranular Stress Corrosion Cracking of Core Shrouds in Boiling Water Reactors*, July 25, 1994

6.4.8 NRC Letter, G.C. Lainas to C. Terry, BWRVIP Chairman, Safety Evaluation of the BWR Vessel and Internals Project BWRVIP-06 Report (TAC No. M93936), 15 September 1998

6.4.9 ACRSR-2091, Mario V. Bonaca to Nils J. Diaz, Report on the Safety Aspects of the License Renewal Application for the Dresden 2 and 3 and Quad Cities 1 and 2 Nuclear Power Stations, 16 September 2004

6.4.10 NRC letter (NVY 96-176), C. Craig Harbuck to Donald A. Reid, 20 November 1996, Review of Core Spray System Piping Collar-to shroud weld flaw evaluation and Core Spray System Inspection Plan at Vermont Yankee Nuclear Power Station (TAC NOS. M96671 and M96689)

Attachment 1: Aging Management Review Results - Reactor Vessel Internals

Component Type	Intended Function	Material[1]	Environment	Aging Effect Requiring Management	Aging Management Program
Control rod guide tubes Tubes	Support for Criterion (a)(1) equipment	Stainless steel 304 SS	Treated water >270ºF (int)	Loss of material	Water chemistry control - BWR
				Cracking – fatigue	TLAA – metal fatigue
				Cracking	BWR vessel internals Water chemistry control - BWR
Control rod guide tubes Bases	Support for Criterion (a)(1) equipment	CASS CF3 or CF8	Treated water >482ºF (int)	Loss of material	Water chemistry control - BWR
				Cracking – fatigue	TLAA – metal fatigue
				Cracking	BWR vessel internals Water chemistry control - BWR
			Treated water >482 °F and neutron fluence	Reduction of fracture toughness	CASS
Core plate	Support for Criterion (a)(1) equipment	Stainless steel	Treated water >270ºF (int)	Loss of material	Water chemistry control - BWR
Plate, Beams		Type 304 / 304L		Cracking – fatigue	TLAA – metal fatigue
Blocks, Plugs Alignment assemblies		Stainless Steel		Cracking	BWR vessel internals Water chemistry control – BWR

Attachment 1: Aging Management Review Results - Reactor Vessel Internals

Component Type	Intended Function	Material[1]	Environment	Aging Effect Requiring Management	Aging Management Program
Core plate Rim bolts	Support for Criterion (a)(1) equipment	Stainless steel	Treated water >270ºF (int)	Cracking	BWR vessel internals Water chemistry control – BWR
				Loss of Preload	TLAA – loss of preload
Core spray lines	Flow distribution	Stainless steel Type 304	Treated water >270ºF (int)	Loss of material	Water chemistry control – BWR
				Cracking – fatigue	TLAA – metal fatigue
				Cracking	BWR vessel internals Water chemistry control - BWR
Fuel support pieces Orificed supports Peripheral supports	Support for Criterion (a)(1) equipment	CASS	Treated water >482ºF (int)	Loss of material	Water chemistry control - BWR
				Cracking – fatigue	TLAA – metal fatigue
				Cracking	BWR vessel internals Water chemistry control - BWR
			Treated water >482 °F and neutron fluence	Reduction of fracture toughness	CASS
Incore dry tubes	Pressure boundary	Stainless steel Type 304 or Type 316	Treated water >270ºF (ext)	Loss of material	Water chemistry control – BWR
				Cracking – fatigue	TLAA – metal fatigue
				Cracking	BWR vessel internals Water chemistry control - BWR
			Air-indoor (int)	None	None

Attachment 1: Aging Management Review Results - Reactor Vessel Internals

Component Type	Intended Function	Material[1]	Environment	Aging Effect Requiring Management	Aging Management Program
Incore guide tubes	Support for Criterion (a)(1) equipment	Stainless steel Type 304	Treated water >270ºF (ext)	Loss of material	Water chemistry control – BWR
				Cracking – fatigue	TLAA – metal fatigue
				Cracking	BWR vessel internals Water chemistry control - BWR
Jet pump assemblies	Floodable volume	Stainless steel	Treated water >270ºF (int)	Loss of material	Water chemistry control – BWR
Risers, riser braces				Cracking – fatigue	TLAA – metal fatigue
Riser hold down bolts				Cracking	BWR vessel internals
Mixer barrels & adapters,		Type 304			Water chemistry control - BWR
Restraint brackets, wedges, bolts		304, 304L, 316L			
Diffusers & tailpipes					
Adapter upper rings		Stainless steel			
Jet pump assemblies	Floodable volume	Nickel-based alloy	Treated water >270ºF (int)	Loss of material	Water chemistry control - BWR
Holddown beams		Alloy X-750		Cracking – fatigue	–TLAA – metal fatigue
Adapter lower ring		Alloy 600		Cracking	BWR vessel internals Water chemistry control - BWR
Jet pump castings	Floodable volume	CASS	Treated water >482ºF (int)	Loss of material	Water chemistry control – BWR
Transition piece		Type 304		Cracking – fatigue	TLAA – metal fatigue
Inlet elbow/nozzle Mixer flange & flare				Cracking	BWR vessel internals Water chemistry control - BWR
Diffuser collar			Treated water >482 °F and neutron fluence	Reduction of fracture toughness	CASS

Attachment 1: Aging Management Review Results - Reactor Vessel Internals

Component Type	Intended Function	Material[1]	Environment	Aging Effect Requiring Management	Aging Management Program
Shroud Upper, central, and lower sections	Floodable volume	Stainless steel Type 304	Treated water >270ºF (int)	Loss of material	Water chemistry control - BWR
				Cracking – fatigue	TLAA – metal fatigue
				Cracking	BWR vessel internals Water chemistry control – BWR
Shroud repair hardware	Support for Criterion (a)(1) equipment	Stainless steel and Nickel-based alloy	Treated water >270ºF (int)	Loss of material	Water chemistry control - BWR
Tie rod assemblies top bracket		Type 304		Cracking – fatigue	TLAA – metal fatigue
radial restraints spring rod / top adapter bottom adapter		Type 304 XM-19 alloy X-750		Cracking	BWR vessel internals Water chemistry control - BWR
Shroud support	Support for Criterion (a)(1) equipment	Nickel-based alloy	Treated water >270ºF (int)	Loss of material	Water chemistry control - BWR
ring, cylinder, and legs	Floodable volume	Alloy 600		Cracking – fatigue	TLAA – metal fatigue
access hole cover		Weld = Alloy 82/182		Cracking	BWR vessel internals Water chemistry control - BWR
Steam dryers	Structural integrity	Stainless steel Type 304	Treated water >270°F (int)	Cracking	BWR vessel internals

Attachment 1: Aging Management Review Results - Reactor Vessel Internals

Component Type	Intended Function	Material[1]	Environment	Aging Effect Requiring Management	Aging Management Program
Top guide assembly	Support for Criterion (a)(1) equipment	Stainless steel Type 304 / 304L	Treated water >270ºF (int)	Loss of material	Water chemistry control - BWR
				Cracking – fatigue	TLAA – metal fatigue
				Cracking	BWR vessel internals Water chemistry control – BWR

1 Material and material references are found in Section 2.2 of this report.