VERIFICATION OF VYNPS LICENSE RENEWAL PROJECT REPORT

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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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TABLE OF CONTENTS

1.0 Introduction 4
- 1.1 Purpose 4
- 1.2 System Description 4
- 1.3 System and Component Intended Functions 8

2.0 Screening 10
- 2.1 Component Evaluation Boundaries 10
- 2.2 Materials for Subcomponents Subject to Aging Management Review (SAMR) 12
- 2.3 Environments for the Reactor Vessel Subcomponents 15

3.0 Aging Effects Requiring Management 17
- 3.1 Low Alloy Steel, Including Carbon Steel, Exposed to Treated Water 18
- 3.2 Low Alloy Steel, including Carbon Steel, exposed to Neutron Fluence 19
- 3.3 Low Alloy Steel, Including Carbon Steel, Exposed to Air-indoor 20
- 3.4 Stainless Steel Cladding 20
- 3.5 Stainless Steel and Nickel-Based Alloys Exposed to Treated Water 21
- 3.6 Stainless Steel and Nickel-Based Alloys Exposed to Neutron Fluence 22
- 3.7 Stainless Steel and Nickel-Based Alloys Exposed to Air-indoor 22
- 3.8 Bolting 23
- 3.9 Operating Experience 24

4.0 Demonstration That Aging Effects will be Managed 25
- 4.1 Aging Management Programs 25
- 4.2 Time-Limited Aging Analyses 28

5.0 Summary and Conclusions 29

6.0 References 30

Attachment 1: Aging Management Review Results 33

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 pressure vessel (RPV) passive subcomponents will be adequately managed so that the intended functions will be maintained consistent with the current licensing basis (CLB) as required by 10 CFR 54.21(a)(3). For additional information on the license renewal project and associated documentation, refer to License Renewal Project Plan. (**Ref. 12**)

The purpose of this report is to demonstrate that aging effects for passive mechanical subcomponents of the VYNPS reactor pressure vessel will be adequately managed for the period of extended operation associated with license renewal. This aging management review report (AMRR) includes the reactor pressure vessel and associated interior and exterior integral attachments. The approach for demonstrating management of aging effects is to first identify the components that are subject to aging management review in Section 2.0. The next step is to define the aging effects requiring management for the system components in Section 3.0. Section 4.0 then evaluates if existing programs and commitments adequately manage those aging effects.

Applicable aging effects were determined using EPRI report 1003056 *Non-Class 1 Mechanical Implementation Guideline and Mechanical Tools* (**Ref. 2**); herein after referred to as the Mechanical Tools**.** This EPRI report provides 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 pressure vessel subcomponents covered in this AMR include materials and environments evaluated in the Mechanical Tools.

Other industry references, including NUREG-1801, Generic Aging Lessons Learned (GALL) Report (**Ref. 1**) and BWR Vessel and Internals Project Reports BWRVIP-05 (**Ref. 6**), 74 (**Ref. 5**), 86 (**Ref. 28**), and 116 (**Ref. 29**), were used to address material and environment combinations not addressed in the Mechanical Tools.

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 pressure vessel.

1.2 System Description

As described in UFSAR Section 4.2, the reactor vessel is designed and fabricated in accordance with ASME Boiler and Pressure Vessel Code, Section III (1965 Edition with Summer 1966 Addenda), its interpretations, and applicable requirements for Class A vessels as defined therein. (Table 4.1-1 of **Ref. 4**) The vessel contains the following subcomponents.

1. Reactor vessel shell and heads

2. Reactor vessel cladding
3. Reactor vessel nozzles, safe ends and thermal sleeves
4. Control rod drive penetrations
5. Incore flux detector penetrations
6. Reactor vessel internal attachments
7. Reactor vessel supports
8. Reactor vessel exterior attachments
9. Reactor vessel pressure boundary bolting
10. Reactor vessel insulation

1.2.1 Reactor Vessel Shell and Heads

The reactor vessel shell is a welded vertical cylinder with hemispherical heads. The cylindrical shell and hemispherical heads are fabricated of low alloy steel plate. The vessel bottom head is welded directly to the vessel shell. Full penetration welds are used at all joints including nozzles throughout the vessel, except for some nozzles and penetrations of less than 3-inch nominal size. (Section 4.2.4.1 of **Ref. 4**)

The flanged reactor vessel upper head is secured to the vessel shell by studs and nuts (See Section 1.2.9). The head and vessel flanges are low alloy steel forgings. The flanged joint between the vessel and vessel head is sealed by two concentric stainless steel seal rings designed for no detectable leakage. Taps are provided between the two rings and outside the outer ring to indicate seal leakage. (Section 4.2.4.1 of **Ref. 4**)

1.2.2 Reactor Vessel Cladding

The cladding is not part of the pressure boundary; rather it provides a protective barrier to minimize corrosion of the low alloy steel and to minimize the release of corrosion products. The reactor vessel shell and heads are clad on the interior with stainless steel weld overlay (0.125” minimum). Some of the low alloy steel nozzles are fully clad, some are partially clad, and some are unclad as identified in Section 2.2.3. The sealing surfaces of the reactor vessel head and shell flanges are weld overlay clad with austenitic stainless steel**.**

1.2.3 Reactor Vessel Nozzles, Safe Ends, and Thermal Sleeves

The vessel nozzles are low alloy steel forgings made in accordance with ASTM A508 as modified by ASME Code Case 1332-3, Paragraph 5. Nozzles of 3-inch nominal size or larger are full penetration welded to the vessel. Nozzles of less than 3-inch nominal size may be partial penetration welded as permitted by ASME Code, Section III. Nozzles which are partial penetration welded are nickel-chromium-iron forgings made in accordance with ASME SB-166 as modified by Code Case 1336. The vessel top head nozzles are provided with flanges with small groove facing. The drain nozzle is of the full penetration weld design and extends 14 inches below the bottom outside surface of the vessel. Feedwater inlet nozzles have thermal sleeves similar to those shown in the detail of Figure 4.2-2. The nozzle provided for the control rod drive hydraulic return line is capped. The cap is connected to the safe end with a full penetration weld. Nozzles connecting to stainless steel piping have "safe ends" of stainless steel of types which are compatible with the material of the mating pipe. Nozzles for connecting carbon steel piping are clad through at least the thickness of the vessel wall or one-half the diameter of the nozzle bore, whichever is less. (Section 4.2.4.1 and Figure 4.2-2 of the UFSAR) Table 1.2-1 gives a complete listing of all vessel nozzles, safe ends and thermal sleeves.

Table 1.2-1 Reactor Vessel Nozzles, Safe Ends and Thermal Sleeves

Nozzle #	Description	Qty	Size (inches)	Nozzle Type	Safe End	Thermal sleeve
N1[1]	Reactor recirculation outlets[1]	2[1]	28[1]	Forged, full penetration[2]	Yes[4]	No
N2[1]	Reactor recirculation inlets[1]	10[1]	12[1]	Forged, full penetration[2]	Yes[4]	Yes[5]
N3[1]	Main steam outlets[1]	4[1]	18[1]	Forged, full penetration[3]	Yes[4]	No
N4[1]	Feedwater inlets[1]	4[1]	10[1]	Forged, full penetration[2]	Yes[4]	Yes[5]
N5[1]	Core spray inlets[1]	2[1]	8[1]	Forged, full penetration[2]	Yes[4]	Yes[5]
N6A[1]	Head spray[1]	1[1]	6[1]	Forged, full penetration[3]	No – flanged[10]	No
N6B[1]	Head instrument[1]	1[1]	6[1]	Forged, full penetration[3]	No – flanged[10]	No
N7[1]	Head vent[1]	1[1]	4[1]	Forged, full penetration[3]	No – flanged[10]	No
N8[1]	Jet pump instruments[1]	2[1]	4[1]	Forged, full penetration[3]	Yes[4]	No
N9[1]	CRD return[1]	1[1]	3[1]	Forged, full penetration[2]	No – capped[7]	No
N10[1]	Core DP/SLC[1]	1[1]	2[1]	Forged, full penetration[2]	No[3]	No[8]
N11 & N12[1]	Instrumentation[1]	4[1]	2[1]	Insert, partial penetration[3]	Yes[4]	No
N13 & N14[1]	RPV flange leakoff[1]	2[1]	1[1]	Drilled penetrations[1]	No[6]	No
N15[1]	Drain[1]	1[1]	2[1]	Full penetration[9]	Yes[4]	No

1. The nozzle numbers, description, quantity, and size came from drawing 919D294 in **Ref. 19** and Table 4.2-2 of the UFSAR [**Ref. 4**]
2. Nozzle configurations taken from drawing 919D294 in **Ref. 19**.
3. Nozzle configurations taken from drawing 104R940 in **Ref. 19**.
4. Nozzles connected to stainless steel piping have "safe ends" of stainless steel. Nozzles connected to carbon steel pipe have safe ends made of low allow steel.
5. The recirculation inlet nozzles, feedwater inlet nozzles, and core spray inlet nozzles have thermal sleeves. (Figure 4.2-2 of **Ref. 4**)
6. The flange leakoff nozzles (N13 and N14) consist of drilled penetrations in the vessel flange (between the two seal rings and outside the outer ring) with nickel-based alloy piping inserted in the holes.
7. The control rod drive hydraulic return line nozzle was capped and the associated piping removed due to stress corrosion cracking. (Section 4.2.4.1 and Figure 4.2-2 of **Ref. 4**).

The cap is connected directly to the nozzle with a full penetration weld. The safe end and thermal sleeve were removed.

8. The nozzle for the core differential pressure and standby liquid control pipe is designed with a transition that provides an annular region between the nozzle and the inner standby liquid control line to minimize thermal shock effects on the reactor vessel in the event that use of the Standby Liquid Control System is required. (Section 4.2.4.1 of the UFSAR)
9. The drain nozzle is of the full penetration weld design and extends 14 inches below the bottom outside surface of the vessel. [Section 4.2.4.1 of the UFSAR]
10. The vessel top head nozzles are provided with flanges with small groove facing. [Section 4.2.4.1 of the UFSAR]

1.2.4 Control Rod Drive Penetrations

There are 89 control rod drive penetrations (6 inch) in the reactor vessel bottom head. A control rod drive stub tube is inserted into each penetration from inside the reactor vessel and is secured by a partial penetration weld. The control rod drive (CRD) housings are then inserted through the stub tube and welded to the end of the stub tube inside the reactor vessel. Each CRD housing transmits loads to the bottom head of the reactor vessel. (Section 4.2.4.6, table 4.2.1, and figure 4.2-2 of **Ref. 4**) The housings extend below the reactor vessel and terminate in flanges to which the drive mechanisms are bolted.

1.2.5 Incore Flux Detector Penetrations

There are 30 incore flux detector penetrations (2”) in the bottom head of the reactor vessel. An instrument housing is inserted in each penetration and partial-penetration welded to the inside of the reactor vessel head. The instrument housing terminates in a flange at the lower end. Either a dry tube (for source and intermediate range detectors) or a local power range monitor (LPRM) assembly is inserted into the instrument housing. The dry tube or LPRM bolts to the flange at the bottom of the incore housing to complete the pressure boundary. Traveling incore probes (TIPs) travel in guide tubes inside the local power range monitors. (Sections 4.2.4.8, 7.5, and Table 4.2.1 of **Ref. 4**)

1.2.6 Reactor Vessel Internal Attachments

There are multiple attachments to the reactor pressure vessel, for supporting various internal components. These internal attachments include the following.

Internal Attachment	Quantity	Reference
Shroud support ring pad	1	Figure 2.9.2.4 of **Ref. 39**
Shroud support feet	14	Drawing 5920-252 **Ref. 41**
Jet pump riser support pads	20	Table 2.4-3 of the UFSAR
Guide rod brackets	2	Table 2.4-3 of the UFSAR
Steam dryer brackets	4	Table 2.4-3 of the UFSAR
Dryer holddown brackets	4	Table 2.4-3 of the UFSAR
Feedwater sparger brackets	8	Table 2.4-3 of the UFSAR
Core spray brackets	4	Table 2.4-3 of the UFSAR
Surveillance specimen holder brackets	6	Table 2.4-3 of the UFSAR

1.2.7 Reactor Vessel Supports

The reactor vessel is supported by a low-alloy steel skirt. The top of the skirt is welded to the bottom of the vessel. The skirt is then supported by a concrete and steel pedestal which carries the load through the drywell to the reactor building foundation slab.

The reactor vessel is laterally and vertically supported and braced to make it as rigid as possible without impairing movement required for thermal expansion. (Section 4.2.4.3 of **Ref. 4**) Vessel stabilizers are connected between the reactor vessel stabilizer brackets and the top of the shield wall surrounding the vessel. (Section 4.2.4.4 of **Ref. 4**)

1.2.8 Reactor Vessel Exterior Attachments

There are multiple external attachments to the reactor pressure vessel. (The support skirt and stabilizer brackets were discussed in Section 1.2.7) The external attachments include the following.

External Attachment	Quantity	Reference
Head lifting lugs	4	Table 4.2.3 of the UFSAR
Insulation supports	2	Table 4.2.3 of the UFSAR
Insulation support brackets	24	Table 4.2.3 of the UFSAR
Thermocouple pads	32	Table 4.2.3 of the UFSAR

1.2.9 Reactor Vessel Pressure Boundary Bolting

The reactor vessel upper head is secured to the reactor vessel shell by studs and nuts. The studs pass through the head closure flange and are threaded into the vessel closure flange. The vessel flanges are sealed as discussed in Section 1.2.1 above. [Section 4.2.4.1 of the UFSAR]

Other bolting reviewed in this AMR are the CRD mechanism to CRD housing bolts, the incore dry tube to incore housing bolts, the LPRM assembly to incore housing bolts, and the vessel nozzle flange to connecting flange bolts for nozzles N6 and N7.

1.2.10 Reactor Vessel Insulation

The lower head and cylindrical shell insulation is permanently installed. The insulation panels for the cylindrical shell of the vessel are held in place by vessel insulation supports located at two elevations on the vessel. The support brackets for each support are full-penetration welded to the vessel at 12 evenly spaced locations around the circumference. (Section 4.2.4.9 of the UFSAR)

1.3 <u>System and Component Intended Functions</u>

As described in UFSAR Section 4.2.1, the power generation objectives of the reactor vessel are:

(1) to contain the reactor core, reactor internals, and the reactor coolant moderator, and
(2) to serve as a high integrity barrier against leakage of radioactive materials to the drywell.

There are no safety objectives for the reactor vessel in the UFSAR; however, VY Vermont Yankee Site Specific Guidance and System Safety Function Sheets, ENN-MS-S-009-VY, give the safety functions for each system at VY. The safety functions of the reactor coolant system (nuclear boiler system) are the following. (**Ref. 3**)

1. Provide and maintain a high integrity reactor coolant pressure boundary inside and out to the first isolation outside primary containment to prevent leakage of radioactive materials.
2. Provide for primary containment isolation/boundary.
3. Provide flow paths for ECCS system injection into the vessel.
4. Contain/structurally support the reactor core, reactor internals, reactor coolant moderator and reactivity control portions of the system.
5. Relieve any overpressure that occurs during abnormal operational transients and over pressurization of the nuclear system via four (4) SRVs and three (3) SVs.
6. Provide pressure relief (in conjunction with ADS) via four (4) SRVs to allow for core cooling.
7. Provide means for emergency and alternate cooling (i.e., feed and bleed) via the core spray spargers and nozzles.
8. Provide valid signals to interfacing plant systems necessary for reactivity control/RPS, SBGT initiation, containment isolation, ARI/RPT and emergency core cooling initiation to prevent the onset and mitigate the consequences of an accident.
9. Provide indication for operators to initiate and control systems used during and following accident and abnormal conditions or to monitor the status of safety systems during design basis accident events (e.g., R.G. 1.97 Category 1 variables).
10. Provide steam quenching capability and primary containment integrity following a LOCA (by directing SRV discharges below the water level of the suppression pool via the discharge lines and actuation of vacuum breakers).
11. Provide structural integrity for safety-related portions of safety-related systems.

The intended functions for the reactor pressure vessel subcomponents subject to aging management review are the following.

- reactor coolant system pressure boundary
- support for Criterion (a)(1) equipment
- corrosion protection (for cladding)

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

2.0 Screening

2.1 Component Evaluation Boundaries

The major components of the reactor pressure vessel include the reactor pressure vessel shell, lower head, upper closure head, cladding, flanges, studs, nuts, nozzles and safe ends. The component evaluation boundaries for this AMRR are the welds between the safe ends and attached piping and the interface flanges for bolted connections. Thermal sleeves that are welded to vessel nozzles or safe ends are reviewed in this AMRR. The control rod drive stub tubes, control rod drive housings, and incore housings are also included in this AMRR. The vessel support skirt, vessel interior welded attachments, and vessel exterior attachments are also considered in this AMRR. Each reactor pressure vessel subcomponent is discussed below. All reactor pressure vessel subcomponents subject to aging management review in this AMRR, along with their materials of construction, are listed in Attachment 1.

2.1.1 Reactor Vessel Shell and Heads

The reactor pressure vessel shell, lower head and upper closure head are subject to aging management review and are reviewed in this AMRR. The reactor internals are evaluated in AMRM-32, Aging Management Review of the Reactor Vessel Internals.

The upper head and vessel flanges, including the studs and nuts, are subject to aging management review and are reviewed in this AMRR. See Section 2.1.9 for discussion of the studs and nuts. The RPV o-rings are periodically replaced. Therefore, these o-rings are not subject to aging management review per 10 CFR 54, section 54.21(a)(1)(ii). The flange leak-off lines are evaluated in AMRM-33, Aging Management Review of the Reactor Coolant System Pressure Boundary.

2.1.2 Reactor Vessel Cladding

The reactor vessel cladding is subject to aging management review and is reviewed in this AMRR.

2.1.3 Reactor Vessel Nozzles, Safe Ends, and Thermal Sleeves

All vessel nozzles and welds that attach the nozzles to the vessel are subject to aging management review and are reviewed in this AMRR. All vessel nozzle safe ends, including the welds that attach the nozzles to safe ends are subject to aging management review and are reviewed in this AMR. The welds that attach safe ends to attached piping and the attached Class 1 piping and valves out to the ASME Section XI IWB inspection boundary are evaluated in AMRM-33, Aging Management Review of the Reactor Coolant System Pressure Boundary.

Thermal sleeves attached to the recirculation inlet nozzles, feedwater inlet nozzles, and core spray inlet nozzles are subject to aging management review and are reviewed in this AMRR. The internal piping attached to these nozzles/thermal sleeves is reviewed in AMRM-32, Aging Management Review of the Reactor Vessel Internals.

2.1.4 Control Rod Drive Penetrations

The control rod drive stub tubes, control rod drive housings and the bolts for the housing flange are subject to aging management review and are reviewed in this AMRR. The mating flange on the control rod drive mechanism and the remainder of the control rod drive hydraulic system are evaluated in AMRM-33, Aging Management Review of the Reactor Coolant System Pressure Boundary. The control rod guide tubes and the thermal sleeve that locks the guide tube to the housing are reviewed in AMRM-32, Aging Management Review of the Reactor Vessel Internals.

2.1.5 Incore Detector Penetrations

The incore housings are subject to aging management review and are reviewed in this AMRR. The incore guide tubes, dry tubes and LPRM assemblies are evaluated in AMRM-32, Aging Management Review of the Reactor Vessel Internals.

2.1.6 Reactor Vessel Internal Attachments

Vessel interior welded attachments (as described in Section 1.2.6) are subject to aging management review and are reviewed in this AMRR. The subcomponents attached to these attachments (not welded to the vessel) are evaluated in AMRM-32, Aging Management Review of the Reactor Vessel Internals.

2.1.7 Reactor Vessel Supports

The support skirt and vessel stabilizer brackets are subject to aging management review and are reviewed in this AMRR. The reactor pressure vessel stabilizers are evaluated in AMRC-06, Aging Management Review of Bulk Commodities.

2.1.8 Reactor Vessel Exterior Attachments

External attachments are subject to aging management review if they are load bearing attachments connected to pressure retaining portions of the vessel. The lifting lugs do not bear significant weight during power operation and are not subject to aging management review. The thermocouple pads and insulation support brackets bear insignificant weight and are not subject to aging management review. The refueling bellows support is connected to the outer surface of the vessel flange, beyond the pressure boundary, and is not subject to aging management review.

2.1.9 Reactor Vessel Pressure Boundary Bolting

The pressure boundary bolting discussed in section 1.2.9 is subject to aging management review and is reviewed in this AMRR. Bolting in this AMR includes the reactor pressure vessel closure flange bolting, the CRD closure bolting, the incore detector closure bolting, and the three nozzles on the upper head flange bolting.

2.1.10 Reactor Vessel Insulation

The vessel insulation is reviewed in AMRC-06, Aging Management Review of Bulk Commodities. (**Ref. 47**)

2.2 Materials for Subcomponents Subject to Aging Management Review (SAMR)

This section lists the materials of construction for those subcomponents that were identified in Section 2.1 as subject to aging management review.

2.2.1 Reactor Vessel Shell, Heads, and Flanges

The reactor vessel shell and heads are made of low alloy steel, A533 Grade B cc1339-2. The closure flanges are forged of low alloy steel, SA508 Class 2 cc1332. (UFSAR table 4.2-1 and **Refs. 17, 31, and 32**)

2.2.2 Reactor Vessel Cladding

The vessel shell and heads are weld overlay clad with austenitic stainless steel which consists of a minimum of two layers and a minimum of 0.125 inch total thickness after all machining. The first layer is deposited with a composition equivalent to ASTM A371, Type ER309, and the second layer has a composition equivalent to ASTM A371, Type ER308, except that the carbon content does not exceed 0.08%.

The sealing surfaces of the reactor vessel head and shell flanges are weld overlay clad similar to the vessel shell and heads except with a minimum of 0.25-inch total thickness after all machining, including the area under the seal grooves. (Section 4.2.4.1 of the UFSAR), (NE8067 Appendix A, Section 19.1, **Ref. 22**), (GE specification 21A1115, Ref. 31)

2.2.3 Reactor Vessel Nozzles, Safe Ends, and Thermal Sleeves

The materials for the vessel nozzles, safe ends, and thermal sleeves are given in table 2.2-1.

Table 2.2-1

Nozzle #[1]	Description[1]	Nozzle Material	Clad	Safe End Material	Thermal Sleeve or Cap or Blank Flange Material
N1	Reactor recirculation outlets	LAS, A508 Cl2 cc1332[2]	Yes[10]	SS, A182 F316[4]	NA
N2	Reactor recirculation inlets	LAS, A508 Cl2[2]	Yes[10]	SS, A182 F316[4]	Thermal sleeve: Type 304 Austenitic Stainless Steel[2]
N3	Main steam outlets	LAS, A508 Cl2[2]	Partial[9]	CS, SA516 Grade 70[5]	NA
N4	Feedwater inlets	LAS, A508 Cl2[2]	Partial[9]	CS, SA508 Class 2[6]	Thermal Sleeve: SS, SA312 Grade 304 NBA, Alloy 600[2]
N5	Core spray inlets	LAS, A508 Cl2[2]	Yes[10]	NBA, MS-16 (Inconel)[7]	Thermal sleeve: Type 304 Austenitic Stainless Steel[2]

Nozzle #[1]	Description[1]	Nozzle Material	Clad	Safe End Material	Thermal Sleeve or Cap or Blank Flange Material
N6A	Head spray	LAS, A508 Cl2[2]	Yes[11]	NA[11]	Blank Flange: A182 Grade F304[8] Blank Flange - A182 Grade F304[8]
N6B	Head instrument	LAS, A508 Cl2[2]	Yes[11]	NA[11]	Blank Flange: A182 Grade F304[8]
N7	Head vent	LAS, A508 Cl2[2]	Yes[12]	NA[12]	Blank Flange: A182 Grade F304[8]
N8	Jet pump instruments	LAS, A508 Cl2[2]	Yes	SS, A336 class F8[4]	NA
N9	CRD return	LAS, A508 Cl2[2]	Yes[10]	NA	Cap: SA182 Grade 316L[2]
N10	Core DP/SLC	LAS, A508 Cl2[2]	Yes[10]	A336 Class 1 F8[4]	NA
N11 & N12	Instrumentation	NBA, SB 166 cc 1336[2]	No	A479 Type 316(SE)[4]	NA
N13 & N14	RPV flange leakoff	NBA, SB-166[3]	No[13]	NA[13]	NA
N15	Drain	LAS, A508-1, Nuclear[3]	Partial[12]	LAS, A508-1[12]	NA

1. The nozzle numbers and description repeat the table in section 1.2.3 and came from drawing 919D294 in Reference 19 and Table 4.2-2 of the UFSAR (**Ref. 4**)
2. Nozzle and thermal sleeve material from Table 4.2.1 of the UFSAR and the certified test reports, **Ref. 32**.
3. Nozzle material taken from Section 3 of the certified test reports, **Ref. 32**
4. Material taken from the PP7015 (**Ref. 17**) Appendix C, Table 5
5. Drawing 5920-483, **Ref. 40**
6. Drawing 5920-241, **Ref. 50**
7. Drawing 5920-624, **Ref. 36**
8. Section 8.12 of GE specification 21A1115, **Ref. 31**
9. Nozzles for connecting carbon steel piping are clad through at least the thickness of the vessel wall or one-half the diameter of the nozzle bore, whichever is less. (Section 4.2.4.1 of the UFSAR)
10. Cladding is shown on drawing 919D294 of **Ref 19**.
11. Nozzles 6A and 6B are shown on drawing 5920-243 (**Ref. 42**)
12. Nozzles 7 and 15 are shown on drawing 5920-244 (**Ref. 43**)
13. RPV leakoff, nozzles N13 and N14, are shown on drawing 5920-324 (**Ref. 44**)

2.2.4 Control Rod Drive Penetrations

The control rod drive stub tubes are made of nickel-based alloy, SB167 cc 1336 and the control rod housings are Type 304 austenitic stainless steel. (Drawing 919D294 of **Ref. 19** and Table 4.2.1 and figure 4.2-2 of the UFSAR)

2.2.5 Incore Detector Penetrations

The incore housings are type 304 or type 316 stainless steel. (Drawing 729E946 of **Ref. 19** and Table 4.2.1 of the UFSAR)

2.2.6 Reactor Vessel Internal Attachments

Internal Attachment	Material	Reference
Shroud Support ring pad	NBA, Alloy 182	Sec. 4.2 of **Ref. 22**
Shroud support feet	NBA, Alloy 600	Sec. 4.6 of **Ref. 22**
Jet pump riser support pads	SS, E308L	Sec. 9.6 of **Ref. 22**
Guide rod brackets	SS, E308L	Sec. 5.5 of **Ref. 22**
Steam dryer brackets	SS, SA240 Type 304	Drawing 5920-329 (**Ref. 45**)
Dryer holddown brackets	SS, Type 304	Section 3.3.4 of the UFSAR
Feedwater sparger brackets	SS, SA240 Type 304	Drawing 5920-330 (**Ref. 46**)
Core spray brackets	SS, Type 304	Section 3.3.4 of the UFSAR
Surveillance specimen holder brackets	SS, Type 304	Section 3.3.4 of the UFSAR

2.2.7 Reactor Vessel Supports

The support skirt is made of low alloy steel, A533 Grade B. (Appendix C, Table 5 of Ref. 17) The vessel stabilizer brackets are also made of A533 Grade B per Section 8.11.2 of the vessel specification, **Ref. 31**.

2.2.8 Reactor Vessel Exterior Attachments

The reactor pressure vessel exterior attachments are not subject to aging management review as discussed in section 2.1.8.

2.2.9 Reactor Vessel Pressure Boundary Bolting

The reactor vessel closure studs, nuts and washers/bushings are made of low alloy steel, A-540 Grade B. (Appendix C, Table 5 of **Ref. 17**)

The CRD flange closure bolting (capscrews and washers) is low alloy steel, SA193 Grade B7. (page 3.4-27 of the UFSAR)

The bolts for the three flanges on the upper head (N6A, N6B, N7) are low alloy steel, A193 Grade B7. (Section 8.12.3 of **Ref. 31**) The nuts are low alloy steel SA194 Grade 2H. (Section 8.12.3 of **Ref. 31**) Material is consistent with the piping specification, **Ref. 23**.

The incore detector closure bolting consists of a stainless steel nut and washer to connect the stainless steel flange to the dry tube. The flange bolts that hold the dry tube or LPRM assembly to the incore housing are stainless steel, SA182 F304 or F316. (**Ref. 49**)

2.2.10 Reactor Vessel Insulation

The reactor pressure vessel insulation is not reviewed in this AMRR as discussed in section 2.1.10.

2.3 Environments for the Reactor Vessel Subcomponents

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

2.3.1 Treated Water

The majority of the components reviewed in this report have the internal environment of treated water. 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. 2).**

Treated water greater than 270 ºF. Above this threshold, stainless steel is susceptible to fatigue (Appendix H of **Ref. 2).**

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. 2)**.

For purposes of this report, steam is considered treated water. VYNPS water chemistry requirements are specified in the Updated Final Safety Analysis Report (UFSAR). Treated reactor water is described in Section 4.3 of the EPRI BWR Water Chemistry guidelines (BWRVIP-79) for normal water chemistry. **(Ref. 8)** Refer to Section 4.1.10 for more information regarding the VYNPS Water Chemistry Program.

The vessel support skirt is the one subcomponent that is normally below 212 °F.

2.3.2 Neutron Fluence

The region of the reactor vessel immediately around the core, the beltline, is exposed to neutron radiation in excess of $1x10^{17}$ n/cm^2. Section 3.2.1 provides further discussion of the radiation effect on the beltline region.

2.3.3 Air-indoor (External)

Mechanical subcomponents of the reactor pressure vessel are located in the primary containment (drywell), which is a sheltered and controlled environment. The atmosphere is inerted with nitrogen to a maximum oxygen level of 4% (Section 5.2.6.3 of the UFSAR), making the atmosphere less corrosive than air. Normal drywell temperature during plant operation is between 135°F and 165°F. (Section 5.2.3.2 of the UFSAR). For purposes of this AMR, no credit is taken for the nitrogen, and all subcomponents of the reactor pressure vessel that are exposed to containment atmosphere are conservatively assumed to be exposed to air-indoor.

External surfaces of most of the reactor vessel subcomponents normally exceed 212 °F and thus do not have moisture present on those surfaces.

Subcomponents completely within the reactor pressure vessel (internal vessel attachments and thermal sleeves) have the internal environment of treated water and no external environment. External vessel attachments have the external environment of air-indoor with no internal environment.

3.0 Aging Effects Requiring Management

Industry reports (BWRVIPs), NUREG-1801, and EPRI report 1003056 are used in this section to identify and evaluate aging effects for the reactor pressure vessel. The following aging effects, and the associated aging mechanisms, were identified for the material/environmental combinations present in the reactor pressure vessel.

loss of material	general corrosion, galvanic corrosion, erosion, flow accelerated corrosion, crevice corrosion, selective leaching and pitting corrosion,
cracking	fatigue, flaw growth, stress corrosion cracking (SCC) and intergranular attack (IGA),
reduction of fracture toughness	thermal and radiation embrittlement and
loss of preload	various mechanisms (for bolting)

For additional information on aging effects, refer to EPRI report 1003056. **(Ref. 2)**

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

Galvanic corrosion is not applicable due to the materials chosen for class 1 components.

Erosion and flow-accelerated corrosion are not applicable to the reactor pressure vessel as the components are made of non-susceptible materials.

Selective leaching is not applicable to the reactor vessel since the susceptible materials (zinc-copper alloys, aluminum alloys, gray cast iron) are not present.

Thermal embrittlement is not applicable to the reactor vessel as none of the VYNPS reactor vessel subcomponents are made of the susceptible material - duplex ferritic-austenitic stainless steel castings (CASS).

Loss of pre-load for bolting, in agreement with the Mechanical Tools, is a design driven effect. Loss of pre-load leads to gasketed closure leakage, but does not defeat the function of the joint to maintain the pressure boundary. Consequently loss of preload is not an aging effect requiring management.

Cracking due to flaw growth is managed by the inspection requirements for Class 1 components in accordance with ASME Section XI, Subsection IWB. Because inservice inspection per ASME Section XI is required in accordance with 10 CFR 50.55a, cracking due to flaw growth is not identified on the tables in Attachment 1.

The following sections document the determination of aging effects requiring management based on the 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 Low Alloy Steel, Including Carbon Steel, Exposed to Treated Water

The reactor pressure vessel subcomponents of carbon steel or low alloy steel clad with stainless steel and exposed to treated water are the upper head, bottom head, shell, closure flange on the vessel, closure flange on the upper head, and all or part of various nozzles as identified in Table 2.2-1. The base metal is discussed in this section and the cladding is discussed in Section 3.3.

The reactor pressure vessel subcomponents of unclad carbon steel or low alloy steel exposed to treated water are all or part of various nozzles and safe ends as identified in Table 2.2-1, and the flanges on the upper head nozzles.

All of these subcomponents are exposed to treated water on the inside and air - primary containment on the outside.

3.1.1 Loss of Material

Unclad low-alloy and carbon steel internal surfaces exposed to treated water are susceptible to general corrosion. These surfaces are also susceptible to loss of material due to pitting and crevice corrosion in the presence of high oxygen levels and contaminants. Therefore, loss of material (general corrosion, pitting corrosion and crevice corrosion) is an aging effect requiring management for RPV unclad low alloy steel items exposed internally to treated water.

The carbon steel or low alloy steel base metal of those components clad with stainless steel is not exposed to the treated water, hence loss of material is not a concern for the base metal.

3.1.2 Cracking - Fatigue

Carbon steel and low alloy steel items, clad or unclad, exposed to treated water are susceptible to cracking by fatigue whenever the temperature exceeds 220 degrees °F. ASME Section III, Subsection NB requires calculation of cumulative usage factors (CUF), and the usage factors must be less than one for the period of extended operation. Cumulative usage factor assessment is a time-limited aging analysis (TLAA). For more information on TLAA, see Section 4.12. Cracking due to fatigue is discussed in VYNPS Report LRPD-04, TLAA - Mechanical Fatigue.

3.1.3 Cracking – Other Than Fatigue

Service loads may result in the growth of pre-service flaws **(Ref. 9)** or initiation and growth of service-induced flaws. The most susceptible locations for flaw initiation and growth are the welded joints. Susceptibility is due to the variations in residual stresses and mechanical properties resulting from the various constituent zones within the joint. Therefore, cracking (flaw initiation and growth) within the welded joints is an aging effect requiring management for carbon steel and low alloy steel components, clad or unclad, for the period of extended operation. However, because inservice inspection per ASME Section XI is required in accordance with 10 CFR 50.55a, cracking due to flaw growth is not identified on the tables in Attachment 1.

Cracking has occurred in cladding of BWR vessels, including the VYNPS reactor vessel. See Section 3.9, Operating Experience for more details. Cracking of the cladding is not expected to propagate into the low-alloy steel base metal. This is summarized in the NRC SER for Hatch Plant (**Ref. 11**) as follows, "As for SCC of the low-alloy steel vessel shells, BWRVIP-05, BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations, (**Ref. 6**) and BWRVIP-60, Evaluation of Crack Growth in BWR Low Alloy Steel RPV Internals, (**Ref. 30**) indicate that even if cracks were to emanate from the vessel cladding, they are not expected to propagate into the low-alloy steel of the reactor vessel. BWRVIP-05 and BWRVIP-60 have been reviewed and approved by the staff." Cracking of the base material due to cracking in the cladding is not an aging effect requiring management for low alloy steel clad with stainless steel.

Stress corrosion cracking and intergranular attack are not significant aging mechanisms for low alloy steel in treated water.

3.2 <u>Low Alloy Steel, including Carbon Steel, exposed to Neutron Fluence</u>

3.2.1 Reduction of Fracture Toughness

Low-alloy steels subjected to high levels of high-energy neutrons are susceptible to increase in material strength and resultant decreased low cycle fatigue resistance known as reduction of fracture toughness. Reduction of fracture toughness is typically measured in terms of Charpy transition temperature shift and Charpy upper-shelf energy decrease. Based on the materials and projected fluence levels, only reactor pressure vessel shell subcomponents in the beltline region are susceptible to reduction of fracture toughness.

The beltline is defined by 10 CFR 50 Appendix G, Fracture Toughness Requirements **(Ref. 10)** to be the region of the reactor pressure vessel that directly surrounds the effective height of the active core and adjacent regions of the reactor pressure vessel that are predicted to experience sufficient neutron irradiation damage to be considered in the selection of the most limiting material with regard to radiation damage. In addition, 10 CFR 50 Appendix H (**Ref. 35**) does not require material surveillance testing for ferritic materials unless the peak neutron fluence at the end of the design life exceeds 1.0×10^{17} n/cm^2. The beltline may thus be alternatively defined as reactor pressure vessel ferritic materials with an end-of-life fluence that exceeds 1.0×10^{17} n/cm^2.

At VYNPS, the beltline for 40-years consists of four plates (1-14, 1-15, 1-16 and 1-17) and their connecting welds, all adjacent to the active fuel zone. There are no nozzles in the beltline region for the current term of operation **(Ref. 5)**. The beltline has been re-evaluated for 60 years (see VYNPS report LRPD-03, TLAA and Exemption Evaluation Results) by extrapolating the data found in **Refs. 33 and 34**. No nozzles will be added to the beltline region due to additional fluence incurred during the period of extended operation at the uprated power. The plate and weld material currently in the beltline for 40-years remain the only materials in the beltline for the period of extended operation. Reduction of fracture toughness applies equally to clad or unclad low alloy steel; however, there are no unclad components in the beltline region.

Reduction of fracture toughness due to radiation embrittlement is an aging effect requiring management for the base metals and weld metals within the beltline region. Analysis of reduction of fracture toughness is a TLAA addressed in VYNPS report LRPD-03, TLAA and Exemption Evaluation Results.

3.3 Low Alloy Steel, Including Carbon Steel, Exposed to Air-indoor

Low alloy steel components exposed only to air are the vessel OD attachments; including the stabilizer attachment brackets and the vessel support skirt and its attachment weld. The reactor pressure vessel's external surfaces (shell, heads, and nozzles) are exposed to air-indoor.

3.3.1 Loss of Material

External ferritic steel surfaces exposed to air-indoor are susceptible to loss of material only if the surface is exposed to moisture. (Appendix E of **Ref. 2**). The reactor vessel is over 212 °F during plant operation and consequently the material is not exposed to moisture. The vessel support skirt is typically less than 212 °F.

Therefore, loss of material due to general corrosion is considered an aging effect requiring management only for the reactor vessel support skirt.

3.3.2 Cracking - Fatigue

Cracking due to fatigue is discussed in Section 3.1.2 above.

3.3.3 Cracking – Other Than Fatigue

External exposure to air-indoor does not contribute to cracking of low alloy steel or carbon steel. Therefore, cracking is not an aging effect requiring management for carbon steel and low alloy steel external surfaces exposed to air-indoor.

3.3.4 Reduction of Fracture Toughness

Reduction of fracture toughness is discussed in Section 3.2.1 above for the neutron fluence environment.

3.4 Stainless Steel Cladding

The reactor vessel internal cladding is considered an independent subcomponent at VYNPS. The internal environment of the cladding is treated water. Since the cladding is welded to the base metal, there is no external environment to address. Other stainless steel subcomponents are addressed in Sections 3.4 and 3.5.

3.4.1 Loss of Material

Due to physical configuration or small surface defects, system fluid contaminants could concentrate in crevices within the reactor pressure vessel. With a high enough concentration of contaminants in the treated water, the cladding is susceptible to loss of material by pitting and crevice corrosion **(Ref. 2)**. Therefore, loss of material (crevice and pitting corrosion) is an aging effect requiring management for stainless steel cladding.

3.4.2 Cracking - Fatigue

At or near attachment welds, cracking of cladding due to fatigue is an aging effect requiring management.

3.4.3 Cracking – Other Than Fatigue

A high concentration of contaminants in treated water may cause stainless steel cladding and attachment welds to the cladding to crack due to stress corrosion cracking/intergranular attack. Stress corrosion cracking of cladding at VYNPS has occurred as discussed in Section 3.9, Operating Experience. While the cladding does not directly contribute to the primary pressure boundary, clad cracking may expose the underlying ferritic steel to treated water, which may lead to corrosion of the base metal. However, as discussed in Section 3.1.3, the cladding cracks will not propagate into the underlying ferritic steel

Cracking (SCC) is an aging effect requiring management for stainless steel cladding.

3.4.4 Reduction of Fracture Toughness

Reduction in fracture toughness due to radiation embrittlement is an aging effect that can affect the strength of the pressure boundary in the reactor pressure vessel beltline region. However, the stainless steel cladding is not credited for adding strength to the reactor vessel, only for protecting the low alloy steel from treated water. Therefore, reduction of fracture toughness due to radiation embrittlement is not an aging effect requiring management for the reactor vessel's stainless steel cladding.

3.5 <u>Stainless Steel and Nickel-Based Alloys Exposed to Treated Water</u>

Reactor pressure vessel stainless steel and nickel-based alloy subcomponents exposed to treated water include the control rod drive housings, control rod drive stub tubes, incore housings, thermal sleeves, instrumentation nozzles, nozzle safe ends, and various internal attachments. The stainless steel safe ends were buttered with nickel-based alloy weld material prior to welding to the associated nozzle (**Ref. 4**). Internal attachments are fabricated of stainless steel, nickel-based alloy, or ferritic steel clad with stainless steel. (**Ref. 5**)

The control rod drive hydraulic return line nozzle was capped and the associated piping removed due to stress corrosion cracking. (Section 4.2.4.1 and Figure 4.2-2 of **Ref. 4**) The cap for this line is forged austenitic stainless steel.

3.5.1 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 or pits within the reactor pressure vessel. With a high enough concentration of contaminants in the treated water, the stainless steel and nickel-based alloy internal surfaces may be susceptible to loss of material by pitting and crevice corrosion. Therefore, loss of material (pitting corrosion and crevice corrosion) are aging effects requiring management for stainless steel and nickel-based alloy steel items exposed to treated water.

3.5.2 Cracking - Fatigue

Cracking by fatigue is an aging effect that applies to stainless steel and nickel-based alloy items designed in accordance with ASME Section III, Subsection NB. The ASME Design Code requires the calculation of cumulative usage factors (CUF) and the usage factors must be less than one for the period extended operation. Cracking due to thermal fatigue is an aging effect requiring management for the stainless steel and nickel-based alloy Class 1 components since the operating temperature of the valves and the piping exceeds 270 ºF.

Therefore, cracking due to fatigue is an aging effect requiring management for stainless steel and nickel base alloys in a treated water environment. Fatigue (cumulative usage factor assessment) is a time-limited aging analysis (TLAA). For more information on TLAA, see Section 4.2.

3.5.3 Cracking – Other Than Fatigue

Cracking at welded joints by growth of fabrication flaws **(Ref. 9)** due to service loads is an aging effect requiring management for stainless steel and nickel-based alloy subcomponents exposed to treated water.

Cracking from stress corrosion and intergranular attack is an aging effect requiring management for stainless steel and nickel-based alloys. Stainless steel thermal sleeves are susceptible to intergranular stress corrosion cracking (IGSCC). In particular there is industry operating experience of recirculation inlet thermal sleeve cracking.

Therefore, cracking (flaw growth, SCC/IGSCC) is an aging effect requiring management for stainless steel and nickel-based alloy steel items. However, because inservice inspection per ASME Section XI is required in accordance with 10 CFR 50.55a, cracking due to flaw growth is not identified on the tables in Attachment 1.

3.6 Stainless Steel and Nickel-Based Alloys Exposed to Neutron Fluence

3.6.1 Reduction of Fracture Toughness

Reduction in fracture toughness due to radiation embrittlement is an applicable aging effect in the reactor pressure vessel beltline region. However, there are no stainless steel (other than the vessel clad discussed in section 3.3 above) or nickel-based alloy items in the beltline region. Therefore, reduction of fracture toughness is not an aging effect requiring management for reactor pressure vessel stainless steel and nickel-based alloy subcomponents.

3.7 Stainless Steel and Nickel-Based Alloys Exposed to Air-indoor

Reactor pressure vessel stainless steel and nickel-based alloy subcomponents exposed externally to air-indoor include the control rod drive housings, incore housings, instrumentation nozzles, and nozzle safe ends.

3.7.1 Loss of Material

Stainless steel and nickel-based alloys exposed to air (primary containment atmosphere) environment are not susceptible to loss of material.

3.7.2 Cracking - Fatigue

External exposure to air-indoor does not promote fatigue as the temperature/pressure changes caused by the external environment are small. The temperature/pressure changes associated with the treated water contribute to the fatigue of the vessel subcomponents and are discussed in Section 3.4.2 above. Therefore, cracking due to fatigue is not an aging effect requiring management for stainless steel and nickel-based alloy subcomponents due to external exposure to air-indoor.

3.7.3 Cracking – Other Than Fatigue

External exposure to air-indoor does not contribute to cracking of stainless steel or nickel-based alloys. Therefore, cracking is not an aging effect requiring management for stainless steel and nickel-based alloy subcomponents externally exposed to air-indoor.

3.7.4 Reduction of Fracture Toughness

External exposure to air-indoor does not contribute to reduction of fracture toughness. Therefore, reduction of fracture toughness is not an aging effect requiring management for stainless steel and nickel-based alloy subcomponents externally exposed to air-indoor.

3.8 Bolting

The reactor pressure vessel closure bolting for the vessel head to shell, the closure bolting for the three upper head nozzle flanges, and the closure bolting for the CRD housing flanges are all constructed of low alloy steel. The closure bolting (bolts, nut, washer, and flange) on the incore detector housings are all stainless steel.

3.8.1 Loss of Material

Loss of material is an aging effect requiring management for the reactor vessel closure bolting. This bolting is exposed to water and steam during refueling operations and subsequent plant heatup.

Loss of material for other pressure boundary bolting is associated with boric acid wastage of low alloy steel bolting. As VYNPS does not normally use boric acid in the reactor coolant system, this is not an aging effect requiring management for the other closure bolting in this AMR. Stainless steel bolting is not susceptible to loss of material.

External low-alloy steel surfaces exposed to air-indoor are susceptible to general corrosion when the temperature is below 212 °F. This bolting is exposed to these conditions only during plant shutdown, and general corrosion during shutdown is accounted for in the metal corrosion allowance. Metal corrosion allowance is a TLAA, see section 4.2 for more information on TLAA.

3.8.2 Cracking – Fatigue

All of the identified bolting is susceptible to cracking by fatigue. Therefore, cracking due to fatigue is an aging effect requiring management for pressure boundary bolting. Fatigue analysis is a time-limited aging analysis (TLAA). For more information on TLAA, see Section 4.2.

3.8.3 Cracking – Other Than Fatigue

All of the identified bolting is susceptible to cracking by stress corrosion cracking (SCC) as identified in Appendix F of the Mechanical Tools (**Ref. 2**).

3.8.4 Reduction of Fracture Toughness

Reduction of fracture toughness is only applicable to materials in the reactor vessel beltline region. As there is no bolting in the beltline region, reduction of fracture toughness is not an aging effect requiring management for closure bolting.

3.9 Operating Experience

The review of site-specific and recent industry operating experience, documented in LRPD-05, Operating Experience Review Results **(Ref. 7),** did not identify any aging effects not addressed in this aging management review report. Cracking in the VYNPS reactor vessel cladding and an indication in one of the reactor vessel shell welds have been found by inservice inspection. Each is discussed below.

3.9.1 Vessel Cladding

A visual inspection of the reactor vessel head, conducted in the 1992 outage as prompted by GE SIL 539, found linear rust indications in the head cladding. Independently, the visual inspection of the vessel found additional linear rust indications. (Section 12 of Appendix A of **Ref. 22**, **Ref. 24**) The visual indications prompted additional inspections by ultrasonic testing. Evaluations concluded that the cracks were due to intergranular stress corrosion of the cladding and that there was no penetration into the base metal. Details can be found in **Ref. 24 and 25**. The cracks are being tracked for measurement of growth and are periodically re-inspected as part of the Reactor Vessel Internals Management Program (**Ref. 14**). These results are consistent with known industry operating experience and the position that clad cracking does not propagate into base metal as discussed in section 3.1.1 of this report.

3.9.2 Reactor Vessel Weld Indication

Reactor vessel plate 1-15 had one indication in the 1995 inspection. The indication is located in the plate below weld EF which joins plates 1-12 and 1-15. The weld is outside the core region. The indication is acceptable for continued service per calculation package YAEC-25Q-301, which was submitted to the NRC by VYNPS letter BVY 96-119 (**Ref. 37**), and approved by the NRC in their letter of 11 Oct 96 (**Ref. 38**). For further discussion of this flaw analysis, see LRPD-04, TLAA – Mechanical Fatigue.

4.0 Demonstration That Aging Effects will be Managed

Section 2.0 described the subcomponents within the reactor pressure vessel 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 the system intended functions. Alternately, if existing programs cannot be shown to manage the aging effects for the period of extended operation, then action(s) will be proposed to augment existing program(s), or create new programs to manage the identified effects of aging.

Demonstration for the purposes of this license renewal technical evaluation is accomplished by establishing a clear relationship among

1. the components under review,
2. the aging effects on these items caused by the material-environment 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 pressure vessel subcomponents subject to aging management review and identifies the aging effects requiring management for the material and environment combinations. The following programs, in combination will manage the effects of aging, thereby precluding loss of the intended functions of the reactor pressure vessel. Section 4.1 discusses these programs in detail and provides the clear relationship between the reactor vessel subcomponent, the aging effect and the aging management program actions that preserve the intended functions for the period of extended operation.

For a comprehensive review of the programs credited for license renewal of VYNPS and a demonstration of how these programs will manage aging effects, see VYNPS Report LRPD-02, Aging Management Program Evaluation Results.

Time-limited aging analyses (TLAA) that have been identified for the reactor pressure vessel are described in section 4.2.

4.1 <u>Aging Management Programs</u>

4.1.1 BWR CRD Return Line Nozzle Program

The BWR CRD Return Line Nozzle Program is credited with managing cracking for the CRD return line nozzle. This program includes system modification (the line has been removed and the nozzle capped), inservice inspection, and water chemistry control. Monitoring and controlling reactor coolant water chemistry is in accordance with the Water Chemistry Control – BWR Program. For additional information on the BWR CRD Return Line Nozzle Program, see VYNPS Report LRPD-02, Aging Management Program Evaluation Results.

4.1.2 BWR Feedwater Nozzle Program

The BWR Feedwater Nozzle Program is credited with managing cracking of the feedwater nozzles. This program augments the ISI program to incorporate the additional inspections recommended in GE-NE-523-A71-0594-A, Revision 1, May 2000, Alternate BWR Feedwater Nozzle Inspection Requirements, as recommended by NUREG-1801 Program XI.M5. Monitoring and controlling reactor coolant water chemistry is in accordance with the Water Chemistry Control – BWR Program. For additional information on the BWR Feedwater Nozzle Program, see VYNPS Report LRPD-02, Aging Management Program Evaluation Results.

4.1.3 BWR Penetrations Program

The BWR Penetrations Program is credited with managing cracking for the SLC/DP and instrumentation nozzles. The program includes inspection and flaw evaluation in accordance with BWRVIP-27-A, BWR Standby Liquid Control System/Core Plate Delta-P Inspection and Flaw Evaluation Guidelines and BWRVIP-49-A, Instrument Penetration Inspection and Flaw Evaluation Guidelines. Monitoring and controlling reactor coolant water chemistry is in accordance with the Water Chemistry Control – BWR Program. For additional information on the BWR Penetrations Program, see VYNPS Report LRPD-02, Aging Management Program Evaluation Results.

4.1.4 BWR Stress Corrosion Cracking Program

The BWR Stress Corrosion Cracking Program is credited with managing cracking (SCC) of BWR stainless steel piping (safe ends) of greater than or equal to 4 inches nominal diameter. This program includes the inspection recommendations of BWRVIP-75-A as accepted by the NRC (**Ref. 48**). Monitoring and controlling reactor coolant water chemistry is in accordance with the Water Chemistry Control - BWR Program. For additional information on the BWR Stress Corrosion Cracking Program, see VYNPS Report LRPD-02, Aging Management Program Evaluation Results.

4.1.5 BWR Vessel ID Attachment Weld Program

The BWR Vessel ID Attachment Weld Program is credited with managing cracking for the pressure vessel internal attachment welds. This program includes inspection and flaw evaluation per the guidelines of BWRVIP-48-A (approved by the NRC), Vessel ID Attachment Weld Inspection and Evaluation Guidelines and per ASME Section XI. Monitoring and controlling reactor coolant water chemistry is in accordance with the Water Chemistry Control - BWR Program. For additional information on the BWR Vessel ID Attachment Welds Program, see VYNPS Report LRPD-02, Aging Management Program Evaluation Results.

4.1.6 BWR Vessel Internals Program

The BWR Vessel Internals Program is designed to manage cracking of the reactor vessel internals and includes inspection and flaw evaluation in accordance with the BWRVIP documents. The following subcomponents of the reactor pressure vessel are managed by the BWR Vessel Internals Program: the CRD housings, the CRD stub tubes, core spray nozzle thermal sleeves, recirculation inlet nozzle thermal sleeves, and the SLC nozzle to safe end weld. The BWR Vessel Internals Program is credited with managing cracking for these stainless steel and nickel-based alloy components. (Section 19 of Appendix A of **Ref.** 14)

For more details on the reactor vessel internals, see VYNPS Report AMRM-32, Aging Management Review of the Reactor Vessel Internals. For additional information on the BWR Vessel Internals Program, see VYNPS Report LRPD-02, Aging Management Program Evaluation Results.

4.1.7 Inservice Inspection Program

The reactor pressure vessel is included in the VYNPS ASME Section XI, Subsections IWB, IWC, and IWD, Inservice Inspection (ISI) Program (**Ref. 17**). The VYNPS ISI Program is a nondestructive examination program that provides for the implementation of ASME Code, Section XI, 1998 Edition, 2000 Addenda Subsection IWB, in accordance with the provisions of 10 CFR 50.55a. The VYNPS ISI program has been augmented to include recommended inspections from various BWRVIP documents and other industry documents that are reflected in Section XI of NUREG-1801.

The Inservice Inspection Program is credited for managing cracking and loss of material for numerous subcomponents of the reactor pressure vessel. All components crediting ISI are listed in Attachment 1.

For additional information on the Inservice Inspection Program, see VYNPS Report LRPD-02, Aging Management Program Evaluation Results.

4.1.8 Reactor Head Closure Studs Program

The Reactor Head Closure Stud Program manages cracking and loss of material of the reactor head closure studs. This program includes a combination of nondestructive examination and vessel bolting/unbolting procedures. The program includes preventive actions and inspection techniques for BWRs. For additional information on the Reactor Head Closure Studs Program, see VYNPS Report LRPD-02, Aging Management Program Evaluation Results.

4.1.9 Reactor Vessel Surveillance Program

This program monitors changes in the fracture toughness of ferritic materials in the reactor pressure vessel beltline region caused by exposure to neutron radiation. This monitoring is accomplished through surveillance material testing in accordance with BWRVIP guidelines and in accordance with Table 4.2.4 of the VYNPS UFSAR (**Ref. 4**). Fracture toughness test data are obtained from material specimens exposed in surveillance capsules, which are withdrawn periodically from the reactor pressure vessel. This program is based on programs documented in the following industry topical reports

- BWRVIP-86A, BWR Integrated Surveillance Program (**Ref. 28**)

  BWRVIP-116, Integrated Surveillance Program Implementation for License Renewal (**Ref. 29**)

The Reactor Vessel Surveillance Program is credited with managing reduction of fracture toughness (radiation embrittlement) of the vessel beltline shell and welds.

For additional information on the Reactor Vessel Surveillance Program, see VYNPS Report LRPD-02, Aging Management Program Evaluation Results.

4.1.10 Water Chemistry Control – BWR Program

Class 1 subcomponents exposed to treated water that are subject to aging management review include carbon steel subcomponents, wrought and forged stainless steel subcomponents, nickel-based alloy subcomponents, and welds. The Water Chemistry Control – BWR Program will manage the following aging effects.

- Loss of material due to general corrosion (carbon and low alloy steel)
- Loss of material due to crevice corrosion and pitting corrosion (all materials)
- Cracking by SCC and intergranular attack (IGA) (stainless steel and nickel-based alloys)

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

4.2 Time-Limited Aging Analyses

The following time-limited aging analyses (TLAA) have been identified for the reactor pressure vessel.

- The reactor pressure vessel is exposed to elevated temperatures, thermal cycling, and the associated metal thermal fatigue. The evaluation of metal fatigue is a TLAA.
- Corrosion allowances are provided for all exposed surfaces of carbon and low alloy steels. The evaluation of the corrosion allowance for the period of extended operation is a TLAA.
- The reactor pressure vessel beltline receives neutron radiation in excess of $1x10^{17}$ n/cm^2 and as such is subject to reduction in fracture toughness. The evaluation of reduction of fracture toughness due to radiation embrittlement is a TLAA.

For additional information, refer to VYNPS Report LRPD-03, TLAA and Exemption Evaluation Results for the evaluation of TLAA and to LRPD-04, TLAA – Mechanical Fatigue for the evaluation of the 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 pressure vessel.

VYNPS Program

ASME Section XI Inservice Inspection

Water Chemistry Control – BWR

Reactor Head Closure Studs

BWR Vessel ID Attachment Welds

BWR Feedwater Nozzles

BWR Control Rod Drive Return Line Nozzle

BWR Stress Corrosion Cracking

BWR Penetrations

BWR Vessel Internals

Reactor Vessel Surveillance

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

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

In conclusion, programs described in Section 4.0 will provide reasonable assurance that the effects of aging on the VYNPS reactor pressure vessel 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

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    Manufacturer’s Serial No. B-4698
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43. Drawing 5920-244, Revision 6 4 dia Vent Nozzle Mk N7 2 dia Drain Nozzle MK N15
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Attachment 1: Aging Management Review Results
Reactor Pressure Vessel

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs
Bolting					
Closure flange studs, nuts, washers and bushings	Pressure boundary	Low alloy steel A540 Grade B[3]	Air-indoor (ext)	Loss of material	Reactor head closure studs
				Cracking – fatigue	TLAA – metal fatigue
				Cracking	Reactor head closure studs
Incore housing bolting flange bolts flange nut and washer	Pressure boundary	Stainless steel A182 Grade F304[2]	Air-indoor (ext)	Cracking - fatigue	TLAA – metal fatigue
				Cracking	Inservice Inspection
Other pressure boundary bolting flange bolts and nuts (N6A, N6B, N7) CRD flange capscrews and washers	Pressure boundary	Low alloy steel Bolts: SA193 Grade B7[10] Nuts: SA194 Grade 2H[2] SA193 Grade B7[11]	Air-indoor (ext)	Cracking – fatigue	TLAA – metal fatigue
				Cracking	Inservice Inspection

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs
Heads and Shell					
Dome - bottom head	Pressure boundary	Low alloy steel with SS cladding A533 Grade B[2,3]	Treated water >220ºF (int)	Loss of material	Water chemistry control - BWR
				Cracking – fatigue	TLAA – metal fatigue
				Cracking	Inservice inspection Water chemistry control - BWR
			Air-indoor (ext)	None	None
Dome - upper head	Pressure boundary	Low alloy steel with SS cladding A533 Grade B[2,3]	Treated water >220ºF (int)	Loss of material	Water chemistry control - BWR
				Cracking – fatigue	TLAA – metal fatigue
				Cracking	Inservice inspection Water chemistry control - BWR
			Air-indoor (ext)	None	None
Flanges (closure) upper head vessel shell	Pressure boundary	Low alloy steel with SS cladding SA508 Class 2[2,3,4]	Treated water >220ºF (int)	Loss of material	Water chemistry control - BWR
				Cracking – fatigue	TLAA – metal fatigue
				Cracking	Inservice inspection Water chemistry control - BWR
			Air-indoor (ext)	None	None
Reactor vessel shell Upper Shell Intermediate nozzle shell Lower shell	Pressure boundary	Low alloy steel with SS cladding A533 Grade B[2,3]	Treated water >220ºF (int)	Loss of material	Water chemistry control - BWR
				Cracking - fatigue	TLAA – metal fatigue
				Cracking	Inservice inspection Water chemistry control - BWR
			Air-indoor (ext)	None	None

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs
Reactor vessel shell Intermediate beltline shell	Pressure boundary	Low alloy steel with SS cladding A533 Grade B[2,3]	Treated water >220ºF (int)	Loss of material	Water chemistry control - BWR
				Cracking – fatigue	TLAA – metal fatigue
				Cracking	Inservice inspection Water chemistry control
			Neutron fluence	Reduction in fracture toughness	Reactor vessel surveillance TLAA – neutron fluence
			Air-indoor (ext)	None	None
Nozzles and Penetrations					
CRD housings	Pressure boundary	Stainless steel TP304[9]	Treated water >270ºF (int)	Loss of material	Inservice inspection Water chemistry control - BWR
				Cracking – fatigue	TLAA – metal fatigue
				Cracking	BWR Vessel Internals Water chemistry control - BWR
			Air-indoor (ext)	None	None
CRD stub tubes	Pressure boundary	Nickel-based alloy SB167 cc 1336[9]	Treated water >270ºF (int)	Loss of material	Inservice inspection Water chemistry control - BWR
				Cracking – fatigue	TLAA – metal fatigue
				Cracking	BWR Vessel Internals Water chemistry control - BWR

ATTACHMENT 1: Aging Management Review Results
Reactor Pressure Vessel

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs
In-core housings	Pressure boundary	Stainless steel TP304, TP316[9]	Treated water >270ºF (int)	Loss of material	Inservice inspection Water chemistry control - BWR
				Cracking – fatigue	TLAA – metal fatigue
				Cracking	Inservice inspection Water chemistry control - BWR
			Air-indoor (ext)	None	None
Nozzles Recirc outlets (N1) Recirc inlets (N2)	Pressure boundary	Low alloy steel with SS cladding SA508 Class 2[4]	Treated water >220ºF (int)	Loss of material	Water chemistry control - BWR
				Cracking – fatigue	TLAA – metal fatigue
				Cracking	Inservice inspection Water chemistry control - BWR
			Air-indoor (ext)	None	None
Nozzles Main steam (N3)	Pressure boundary	Low alloy steel with partial SS cladding SA508 Class 2[4]	Treated water >220ºF (int)	Loss of material	Inservice inspection Water chemistry control - BWR
				Cracking – fatigue	TLAA – metal fatigue
				Cracking	Inservice inspection Water chemistry control - BWR
			Air-indoor (ext)	None	None
Nozzles Feedwater (N4)	Pressure boundary	Low alloy steel with partial SS cladding SA508 Class 2[4]	Treated water >220ºF (int)	Loss of material	Inservice inspection Water chemistry control - BWR
				Cracking – fatigue	TLAA – metal fatigue
				Cracking	BWR feedwater nozzle
			Air-indoor (ext)	None	None

ATTACHMENT 1: Aging Management Review Results
Reactor Pressure Vessel

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs
Nozzles Core Spray (N5) Head Spray (N6A) Head Instr. (N6B) Head vent (N7) Jet pump inst. (N8)	Pressure boundary	Low alloy steel with SS cladding SA508 Class 2[4]	Treated water >220ºF (int)	Loss of material	Water chemistry control – BWR
				Cracking – fatigue	TLAA – metal fatigue
				Cracking	Inservice inspection Water chemistry control - BWR
			Air-indoor (ext)	None	None
Nozzle CRD return (N9)	Pressure boundary	Low alloy steel with SS cladding SA508 Class 2[4]	Treated water >220ºF (int)	Loss of material	Water chemistry control - BWR
				Cracking - fatigue	TLAA – metal fatigue
				Cracking	BWR CRD return line nozzle
			Air-indoor (ext)	None	None
Nozzle Core DP/SLC (N10)	Pressure boundary	Low alloy steel with SS cladding SA508 Class 2[4]	Treated water >220ºF (int)	Loss of material	Water chemistry control - BWR
				Cracking – fatigue	TLAA – metal fatigue
				Cracking	BWR penetrations Water chemistry control - BWR
			Air-indoor (ext)	None	None
Nozzles –Instrumentation (N11, N12)	Pressure boundary	Nickel-based alloy SB-166[4]	Treated water >270ºF (int)	Loss of material	Inservice inspection Water chemistry control - BWR
				Cracking – fatigue	TLAA – metal fatigue
				Cracking	BWR penetrations Water chemistry control - BWR
			Air-indoor (ext)	None	None

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs
Nozzles Flange leakoff (N13, N14)	Pressure boundary	Nickel-based alloy SB166[4]	Treated water >270ºF (int)	Loss of material	Inservice inspection Water chemistry control - BWR
				Cracking – fatigue	TLAA
				Cracking	Inservice inspection Water chemistry control – BWR
			Air-indoor (ext)	None	None
Nozzles Drain (N15)	Pressure boundary	Low alloy steel with partial SS cladding SA508 Class 1[4]	Treated water >220ºF (int)	Loss of material	Inservice inspection Water chemistry control – BWR
				Cracking - fatigue	TLAA – metal fatigue
				Cracking	Water chemistry control – BWR Inservice inspection
			Air-indoor (ext)	None	None
Safe Ends, Thermal Sleeves, Flanges, Caps					
CAP CRD return line (N9)	Pressure boundary	Stainless steel SA182 Grade 316L[7]	Treated water >270ºF (int)	Loss of material	Inservice inspection Water chemistry control - BWR
				Cracking – fatigue	TLAA – metal fatigue
				Cracking	BWR CRD return line nozzle Water chemistry control - BWR
			Air-indoor (ext)	None	None

ATTACHMENT 1: Aging Management Review Results
Reactor Pressure Vessel

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs
Flanges head nozzle flanges (N6, N7) blank flanges (N6)	Pressure boundary	Stainless steel A182 Grade F304[2]	Treated water >270ºF (int)	Loss of material	Inservice inspection Water chemistry control - BWR
				Cracking - fatigue	TLAA – metal fatigue
				Cracking	Inservice inspection Water chemistry control - BWR
			Air-indoor (ext)	None	NA
Safe ends >=4” recirc outlet (N1) recirc inlet (N2)	Pressure boundary	Stainless steel A182 F316[3]	Treated water >270ºF (int)	Loss of material	Inservice Inspection Water chemistry control – BWR
				Cracking – fatigue	TLAA – metal fatigue
				Cracking	BWR stress corrosion cracking Water chemistry control – BWR
			Air-indoor (ext)	None	None
Safe ends >=4” Main steam (N3) Feedwater (N4)	Pressure boundary	Carbon steel SA516 Grade 70[6] ASTM A508 Class 1[12]	Treated water >220ºF (int)	Loss of material	Inservice Inspection Water chemistry control – BWR
				Cracking – fatigue	TLAA – metal fatigue
			Air-indoor (ext)	None	None
Safe ends >=4” Core spray (N5)	Pressure boundary	Nickel-based alloy MS16 (Inconel)[5]	Treated water >270ºF (int)	Loss of material	Inservice Inspection Water chemistry control - BWR
				Cracking – fatigue	TLAA – metal fatigue
				Cracking	BWR stress corrosion cracking Water chemistry control - BWR
			Air-indoor (ext)	None	None

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs
Safe ends >=4" Jet pump instrument (N8)	Pressure boundary	Stainless steel A336 Cl F8[3]	Treated water >270ºF (int)	Loss of material	Inservice Inspection Water chemistry control – BWR
				Cracking – fatigue	TLAA – metal fatigue
				Cracking	BWR stress corrosion cracking Water chemistry control – BWR
			Air-indoor (ext)	None	None
Safe ends <4" Core DP / SLC (N10) Instrumentation (N11, N12)	Pressure boundary	Stainless steel A336 Cl F8[2] A479 TP316(SE)[2]	Treated water >270ºF (int)	Loss of material	Inservice Inspection Water chemistry control - BWR
				Cracking – fatigue	TLAA – metal fatigue
				Cracking	Inservice inspection Water chemistry control - BWR
			Air-indoor (ext)	None	None
Safe ends <=4" Drain (N15)	Pressure boundary	Low alloy steel ASTM A508-1[13]	Treated water >220ºF (int)	Loss of material	Inservice Inspection Water chemistry control - BWR
				Cracking – fatigue	TLAA - metal fatigue
			Air-indoor (ext)	None	None
Thermal sleeves Recirc inlet (N2) Core spray (N5)	Pressure boundary	Stainless steel Type 304[7,8]	Treated water >270ºF (int)	Loss of material	Water chemistry control - BWR
				Cracking - fatigue	TLAA – metal fatigue
				Cracking	BWR vessel internals Water chemistry control - BWR

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs
Thermal sleeves Feedwater (N4)	Pressure boundary	Stainless steel and Nickel-based alloy SA 312 Gr 304 Alloy 600[7]	Treated water >270ºF (int)	Loss of material	Water chemistry control - BWR
				Cracking – fatigue	TLAA – metal fatigue
				Cracking	Inservice inspection Water chemistry control - BWR
Weld N10 - SLC nozzle to safe end weld	Pressure boundary	Nickel-based alloy[3]	Treated water >270ºF (int)	Loss of material	BWR vessel internals Water chemistry control - BWR
				Cracking – fatigue	TLAA – metal fatigue
				Cracking	BWR Penetrations Water chemistry control - BWR
			Air-indoor (ext)	None	None

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs
Vessel Attachments and Supports					
Internal attachments	Pressure boundary	Stainless steel & Nickel-based alloy	Treated water >270ºF (int)	Loss of material	Inservice inspection Water chemistry control - BWR
Shroud support ring pad (1)		Alloy 182[1]		Cracking – fatigue	TLAA – metal fatigue
Shroud support feet (14)		Alloy 600[1]		Cracking	BWR vessel ID attachment welds Water chemistry control - BWR
Jet pump riser pads (20)		E308L[1]			
Core spray brackets (4)		E308L[1]			
Guide rod brackets (2)		E308L[1]			
Steam dryer brackets (4)		SA240 Type 304			
Dryer holddown brackets (4)		SS, Type 304			
Surveillance specimen holder brackets		SS, Type 304			
Feedwater sparger brackets (8)		SA240 Type 304			
Supports Stabilizer pads Support skirt	Support for Criterion (a)(1) equipment	Low alloy steel A533 Grade B[2] A533 Grade B[3]	Air-indoor (ext)	Loss of material	Inservice inspection
				Cracking - fatigue	TLAA – metal fatigue

Footnotes to Attachment 1:

1 NE 8067, App A, Sec. 19.1 (**Ref. 22)**
2 GE Spec 21A1115, (**Ref. 31**)
3 PP7015, Appendix C, table 5 (**Ref. 17**)
4 Certified Test Reports (**Ref. 32**)
5 Drawing 5920-624 (**Ref. 36)**
6 Drawing 5920-483 (**Ref. 40**)
7 UFSAR Table 4.2-1
8 BWRVIP-75 (**Ref. 5**)
9 GEK-9608 (**Ref. 32**)
10 VYNP-V1-**III**-P-1, (**Ref. 23**)
11 UFSAR, page 3.4-27
12 Customer Review.
13 Drawing 5920-244 (Ref. 43)