VERIFICATION OF VYNPS LICENSE RENEWAL PROJECT REPORT

Title of Report: TLAA and Exemption Evaluation Results

Report Number: LRPD-03

Revision: 0

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.

Other document(s) impacted by this revision: ___ Yes, See Attachment _X_ No

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REVISION DESCRIPTION SHEET

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0	Initial Issue	All

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1.0 Introduction

This report is part of the integrated plant assessment (IPA) performed to extend the operating license of Vermont Yankee Nuclear Power Station (VYNPS). This report reviews the time-limited aging analyses (TLAA), and exemptions to Part 10 of the Code of Federal Regulations (10 CFR), and evaluates them for the period of extended operation as required by 10 CFR 54. For additional information on the license renewal project and associated documentation, refer to the license renewal project plan. (Ref. 6.6.1)

The Code of Federal Regulations 10 CFR 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants" governs the issuance of renewed operating licenses for nuclear power plants and includes requirements for the review of time-limited aging analyses (TLAA). The definition of time-limited aging analyses (TLAA) is in 10 CFR 54.3.

> *Time-limited aging analyses*, for the purposes of this part, are those licensee calculations and analyses that:
>
> 1) involve systems, structures, and components within the scope of license renewal, as delineated in §54.4(a);
> 2) consider the effects of aging;
> 3) involve time-limited assumptions defined by the current term of operation, for example, 40 years;
> 4) were determined to be relevant in making a safety determination;
> 5) involve conclusions or provide the basis for conclusions related to the capability of the system, structure, and component to perform its intended functions, as delineated in §54.4(b); and
> 6) are contained or incorporated by reference in the current licensing basis.

An example of a TLAA is the reactor vessel neutron embrittlement analysis that is based on the neutron exposure for the current operating term and must be reevaluated for the period of extended operation.

Section 10 CFR 54.21(c) requires a list of time-limited aging analyses (TLAA) in the application for a renewed license. Section 10 CFR 54.21(c)(2) requires a list of current exemptions to 10 CFR 50 based on TLAA in the application for a renewed license.

> §54.21 Contents of application -- technical information.
>
> (c) An evaluation of time-limited aging analyses.
>
> > (1) A list of time-limited aging analyses, as defined in §54.3, must be provided. The applicant shall demonstrate that—
> >
> > i) the analyses remain valid for the period of extended operation; or
> >
> > ii) the analyses have been projected to the end of the period of extended operation; or

iii) the effects of aging on the intended function(s) will be adequately managed for the period of extended operation

In addition, 10 CFR 54 states that a list must be provided of plant-specific exemptions granted (and still in effect) pursuant to 10 CFR 50.12 that are based on time-limited aging analyses as defined in 10 CFR 54.3. An applicant must provide an evaluation that justifies continuation of these exemptions for the period of extended operation. TLAA and exemptions are discussed in this document and a reference is provided to supporting site documents.

The methods used for identification and evaluation of TLAA and exemptions are described in Section 2.0 with the identified TLAA listed in Attachment 1. Identified exemptions are listed in Attachment 2. TLAA search results from the Updated Final Safety Analysis Report (UFSAR) and recommended UFSAR text changes are included in Attachment 3. The potential TLAA are evaluated in Section 3.0 while the VYNPS exemptions based on TLAA are evaluated in Section 4.0. A summary description of the evaluation of TLAA for the period of extended operation will be provided in the UFSAR supplement.

2.0 Identification of TLAA and Exemptions

2.1 Identification of TLAA

The process used to identify the time-limited aging analyses is consistent with the guidance provided in NEI 95-10, *Industry Guidelines for Implementing the Requirements of 10 CFR 54 - The License Renewal Rule*, Revision 6, June 2005. Calculations and analyses that could potentially meet the definition of 10CFR 54.3 were identified by searching CLB documents including the following.

- Technical Specifications
- UFSAR
- docketed licensing correspondence
- fire protection program documents
- NRC safety evaluation reports
- BWRVIP documents

Industry documents that list generic time-limited aging analyses were also reviewed to provide additional assurance of the completeness of the plant-specific list. These documents included NEI 95-10; NUREG-1800, *Standard Review Plan (SRP) for Review for License Renewal Applications for Nuclear Power Plants*, Revision 1, September 2005; NUREG-1801, *Generic Aging Lessons Learned (GALL) Report*, Revision 1, September 2005; and NRC safety evaluation reports related to license renewal applications by other BWR licensees.

Industry documentation, owners group reports, vendor reports, and site searches were utilized to identify TLAA that are applicable to VYNPS. EPRI reports such as TR-105090 (Ref. 6.3.9) and other license renewal applications (Ref. 6.3.2, 6.3.3, 6.3.4) were used to identify generic TLAA. Site-specific evaluations ensured TLAA applicability. During preparation of the VYNPS Class 1 and non-Class 1 aging management review reports, TLAA were identified in individual reports. The TLAA identified in individual reports are evaluated in this report or in LRPD-04, TLAA – Mechanical Fatigue.

A computer database search was performed to identify TLAA from the UFSAR (Ref. 6.5.1), the Technical Specifications (Ref. 6.5.4), the QA program (Ref. 6.5.6), the ASME Section XI Inservice Inspection Program including all Relief Requests (Ref. 6.5.58), the Fire Hazards Analysis (Ref. 6.5.7), the Fire Protection Program Commitment Reference Manual (Ref. 6.5.8), and available NRC correspondence (Ref. 6.5.10). The search criteria utilized key words and phrases such as age, aging, crack growth, corrosion allowance, cycles, cyclic, embrittlement, EFPY, fatigue, 40 years, life (design life, service life), RT_{NDT}, time limit, usage factor. The Vermont Yankee Fire Protection and Appendix R Program (Ref. 6.5.9) was reviewed manually.

The key word search of the aforementioned CLB documentation resulted in a list of potential TLAA. Attachment 1 lists the resulting potential TLAA and the documents referencing the potential TLAA.

The TLAA identified by the various searches were consolidated. For example, the database search identified a number of reactor vessel neutron embrittlement analyses that were TLAA (e.g., RT_{NDT} and C_VUSE analyses). Section 3.1 of this report discusses the review of the reactor vessel neutron embrittlement TLAA. This is consistent with the license renewal application format and content guidelines presented in NEI 95-10.

2.2 Identification of Exemptions

A review of docketed correspondence identified VYNPS exemptions. No VYNPS exemptions depend on time-limited aging analyses.

To identify exemptions for VYNPS, a keyword search was conducted on the UFSAR, Technical Specifications, and NRC correspondence. This review involved a search of the database to identify exemptions that were granted pursuant to 10 CFR 50.12. The search criteria utilized key terms including “50.12”and “exemption.” Attachment 2 lists the identified exemptions and lists references for the exemptions. In accordance with 10 CFR 54.21(c)(2), exemptions that are not in effect (i.e., exemptions that were temporary or have been eliminated/withdrawn by later correspondence) are not discussed in the license renewal application and are not discussed further in this report.

3.0 Evaluation of TLAA

Attachment 1 of this document summarizes potential TLAA applicable to VYNPS. In the rest of Section 3, each potential TLAA identified in Attachment 1 was examined to determine if it meets the definition of a TLAA in accordance with 10 CFR 54.3. Analyses and calculations that meet the TLAA definition are evaluated in accordance with the options provided in 10 CFR 54.21 (c)(1).

3.1 Reactor Vessel Neutron Embrittlement

The regulations governing reactor vessel integrity are in 10 CFR 50. Section 50.60 requires that all light-water reactors meet the fracture toughness, pressure-temperature limits, and material surveillance program requirements for the reactor coolant boundary as set forth in Appendices G and H of 10 CFR 50.

The VYNPS current licensing basis analyses evaluating reduction of fracture toughness of reactor vessel for 40 years are TLAA. The reactor vessel neutron embrittlement time-limited aging analyses were projected to the end of the period of extended operation (54 EFPY) in accordance with 10 CFR 54.21 (c)(1)(ii) as summarized below.

The VYNPS current licensing basis contains calculations and analyses that address the effects of neutron irradiation embrittlement on the reactor vessel (Refs. 6.5.3, 6.5.11, 6.5.12, and 6.5.44). The analyses evaluating reduction of reactor vessel fracture toughness for 40-years are TLAA. The appropriate calculations have been updated based on a 60-year operating term assuming that licensed activities will continue to be conducted in accordance with the CLB. The Reactor Vessel Surveillance Program described in VYNPS Report LRPD-02, “Aging Management Program Evaluation Report” will ensure that the time-dependent parameters used in the TLAA described below remain valid through the period of extended operation. The reactor vessel neutron embrittlement TLAA was projected in Ref. 6.5.44 to approximately 51.6 EFPY. 51.6 EFPY was used in support of extended power uprate, based on actual EFPY before the uprate and an assumed capacity factor of 90% after the uprate. For license renewal, the fluence was extrapolated to 54 EFPY, as discussed in Section 3.1.1 below.

Upper shelf energy (C_VUSE) was calculated based on the 54 EFPY extrapolated fluence to demonstrate that 10 CFR 50 Appendix G requirements are satisfied. Section 3.1.2 below discusses the results.

Adjusted reference temperature has been calculated based on the 54 EFPY extrapolated fluence, and the results are presented in section 3.1.3 below.

The currently licensed P-T limit curves remain bounding for the period of extended operation, including the extended power uprate (Ref. 6.5.61). See section 3.1.4 below for more detail.

3.1.1 Reactor Vessel Fluence

GE’s Licensing Topical Report NEDC-32983P-A, which was approved by the NRC for licensing applications in Reference 6.2.31, documents the method used for the neutron flux calculation. The NRC found that, in general, this method adheres to the guidance in Regulatory Guide 1.190 for neutron flux evaluation. The calculated reactor vessel ID fluence for 51.6 EFPY is 5.16×10^{17} n/cm^2 (E>1 MeV), assuming a power uprate from 1593 MWt to 1912 MWt (Ref. 6.5.44). The neutron flux distribution was calculated based on the three-dimensional flux synthesis of two separate two-dimensional flux solution calculations performed in an (r,z) and an (r,Θ) model. These flux solution calculations use the two-dimensional discrete ordinates code DORTG01V, which is a controlled version of DORT in the GE Engineering Computation Program (ECP) library.

Extrapolated to 54 EFPY, the vessel surface (ID) fluence is 5.39 $x10^{17}$ n/cm^2(E>1 MeV). The fluence was extrapolated by simply extending the straight line between 33 EFPY and 51.6 EFPY to 54 EFPY. Using Regulatory Guide 1.99, Revision 2, Equation (3), results in a 54 EFPY ¼T fluence of $3.98x10^{17}$ n/cm^2.

The beltline is defined by 10 CFR 50 Appendix G, Fracture Toughness Requirements as 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 does not require material surveillance testing for ferritic materials unless the peak neutron fluence at the end of the design life exceeds $1.0x10^{17}$ n/cm^2. The beltline is thus considered the reactor pressure vessel ferritic materials with an end-of-life fluence that exceeds $1.0x10^{17}$ n/cm^2.

At VYNPS, the beltline for 40-years consists of four plates (1-14, 1-15, 1-16, 1-17) and their connecting welds, all adjacent to the active fuel zone. There are no nozzles in the beltline region (Ref. 6.2.1). The beltline has been re-evaluated for 60 years using the axial distribution of fast fluence at the RPV wall (Figure 3-7 of Ref. 6.5.44). Based on the additional fluence incurred during the period of extended operation, the vertical section of the reactor vessel ID that will receive greater than 1 x 10^{17} n/cm^2 extends from 3.5 inches below the bottom of the active fuel to 10 inches above the top of the active fuel. There are no nozzles in this region. Based on drawing 5920-3773 (Ref. 6.5.63**),** this is equivalent to a vessel height of 204 inches to 361.5 inches. This same drawing shows that the centerline of the recirculation inlet nozzles is at 186 inches. The top of the nozzle weld is not specifically shown on the drawing, but can be approximated as 202 inches. Above the core, the nearest nozzles are the instrumentation nozzles (N11A, N11B, N12A and N12B) at 422 inches. No nozzles will be added to the beltline region by additional fluence incurred during the period of extended operation at the uprated power. The limiting plate and weld material in the beltline for 40-years remain the limiting materials for the period of extended operation.

Fluence, calculated based on the operating term, is a time-limited assumption for the TLAA that evaluate reactor vessel embrittlement. The reactor vessel fluence calculation has been projected to the end of the period of extended operation and that result is used throughout the remainder of Section 3.1 of this report.

3.1.2 Pressure/Temperature Limits

Appendix G of 10 CFR 50 requires the reactor vessel to remain within established pressure-temperature (P-T) limits during reactor vessel boltup, hydrotest, pressure tests, normal operation, and anticipated operational occurrences. These limits are from calculations that use the materials and fluence data obtained through the reactor vessel surveillance program. Normally, the pressure-temperature limits are calculated for several years into the future.

In March 2003 (Ref. 6.5.16), VYNPS submitted a license amendment request to change the P-T limits to incorporate data from analysis of the first VYNPS surveillance capsule and to extend the curves to 32 EFPY. The NRC approved this submittal as Amendment 218 to the VYNPS license (Ref. 6.2.11). As stated in that SER, VYNPS used conservative values for determining the P-T limits. Those values were peak vessel fluence of 1.24×10^{18} n/cm^2, ¼ T ART of 89°F and a ¾ T ART of 73°F. Table 3-5 of this report compares the bases for the present curves with the projected fluence and ARTs for 54 EFPY and shows that the projected values at 54 EFPY (fluence of 5.39 $\times 10^{17}$ n/cm^2, ¼ T ART of 68.5°F and a ¾ T ART of 56.9°F) are still less that those used for the P-T curves. As such the TLAA for Pressure Temperature limits remains valid in accordance with 10CFR54.21(c)(1)(i).

VYNPS will submit a technical specification change request prior to 32 EFPY to officially update the curves in the Technical Specifications. Even though the curves may be the same, the applicable EFPY will be changed.

3.1.3 Charpy Upper Shelf Energy (C_VUSE)

Appendix G of 10 CFR 50 requires that reactor vessel beltline materials "have Charpy upper-shelf energy … of no less than 75 ft-lb initially and must maintain Charpy upper-shelf energy throughout the life of the vessel of no less than 50 ft-lb…."

Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," provides two methods for estimating Charpy upper-shelf energy (C_VUSE) at end of life. Position 1 applies for material that does not have surveillance data. Position 2 applies for material with surveillance data. Position 2 requires a minimum of two sets of credible surveillance data. Since VYNPS has data from only one material surveillance capsule, Position 2 does not apply. For Position 1, the percent drop in C_VUSE for a stated copper content and neutron fluence is determined by reference to Figure 2 of Regulatory Guide 1.99, Revision 2. This percentage drop is applied to the initial C_VUSE to obtain the adjusted C_VUSE. Table 3.1 calculates the end of life C_VUSE by this method.

Safety analysis report NEDC-33090P (Ref. 6.5.3) documents the most recent calculations of C_VUSE. NEDC-33090P was submitted to the NRC as part of the VYNPS power uprate request (Ref. 6.5.2). Analyses were done for 32 EFPY at the previously licensed power level of 1598 MWt, and for 33 and 51.6 EFPY with a power uprate to 1912 MWt at 25 EFPY. Results of NEDC-33090P are extrapolated to 54 EFPY in this report.

The VYNPS unirradiated surveillance specimens were from plate 1-14 with a C_VUSE of 89 ft-lb (137 ft-lb times 0.65) (Ref. 6.5.14). The 54 EFPY C_VUSE value for plate 1-14 was calculated using Regulatory Guide 1.99, Position 1, Figure 2. Specifically, the formulae for the lines were used to calculate the percent drop in C_VUSE (Ref. 6.2.9). The calculation used the fluence determined in Section 3.1.1 above. For 54 EFPY, Table 3-1 shows the minimum projected C_VUSE for plate1-14 remains above the 50 ft-lb requirement of Appendix G of 10 CFR 50. As such, this TLAA has been extrapolated for the period of extended operation in accordance with 10CFR54.21(c)(1)(ii).

Initial (un-irradiated) upper shelf energy data for the weld materials and for plates 1-15, 1-16, and 1-17 do not exist. (Ref. 6.5.14) The BWR Owners Group prepared an equivalent margins analysis for plants without this data. The analysis (NEDO-32205-A, -Ref. 6.5.13) used Code case N-512. The NRC reviewed and accepted the evaluation, as documented in the SER in

Ref. 6.2.32. Rather than calculating an end of life C_VUSE (impossible without an initial C_VUSE) a plant may calculate the percent drop in C_VUSE, and show that the percent drop is less than the percent drop in the equivalent margins analysis.

Appendix B of BWRVIP-74 provides a method to evaluate USE at 54 EFPY using plant-specific surveillance data. BWRVIP-74 gives allowable percent drops in C_VUSE of 23.5% for BWR 3-6 plate and from 39% for welds. The NRC approved the use of these new values in their SER (Ref. 6.2.24). Table 3-4 uses the BWRVIP-74 method to verify that the VYNPS reductions in USE remain less than the reduction calculated in the BWRVIP-74 equivalent margins analyses at 54 EFPY for beltline welds and plates 1-15, 1-16, 1-17. As such, this TLAA has been projected to the end of the period of extended operation in accordance with 10CFR54.21(c)(1)(ii).

3.1.4 Adjusted Reference Temperature

Irradiation by high-energy neutrons raises the value of adjusted reference temperature (ART) for the reactor vessel. The initial RT_{NDT} is determined through testing un-irradiated material specimens. The shift in reference temperature, ΔRT_{NDT}, is the difference in the 30 ft-lb index temperatures from the average Charpy curves measured before and after irradiation. (ART) = RT_{NDT} + ΔRT_{NDT} + margin. (Regulatory Guide 1.99, Revision 2)

Safety analysis report NEDC-33090P (Ref. 6.5.3) includes the most recent calculations of RT_{NDT}. NECD-33090P was submitted to the NRC as part of the VYNPS power uprate request (Ref. 6.5.2). The report calculated the adjusted RT_{NDT} for the welds and plates. Analyses were completed for both 32 EFPY at the previously licensed power level of 1598 MWt and for 33 EFPY with a power uprate to 1912 MWt at 25 EFPY. In addition to new fluence values, this report provided initial RT_{NDT} for each plate, rather than the maximum plate value found in the Reactor Vessel Integrity Database. Results of NEDC-33090P are extrapolated to 54 EFPY in this report. Regulatory Guide 1.99 Revision 2, Regulatory Position 1, defines the calculation methods used for ΔRT_{NDT} and ART.

VYNPS response to GL 92-01 (Ref. 6.5.15) included chemistry data. Chemistry factors (CF) were interpolated from Table 1 in RG 1.99. Initial RT_{NDT} values and standard deviations were taken from VYNPS NEDC-33090P, Table 3-2a. Standard deviations for ΔRT_{NDT}, σ_{Δ}, were calculated as one-half the ΔRT_{NDT} since in all instances 0.5 times the ΔRT_{NDT} was less than 28 °F for welds and 17 °F for plates. Margins were calculated as twice the square root of the sum of the squares of the two standard deviations. Note that adjusted reference temperatures use ¼T fluence.

Section 3.1.1 discussed calculation of fluence. Fluence factors (FF) were calculated using Equation 2 in Regulatory Guide 1.99, Revision 2.

Extrapolated ΔRT_{NDT} values were calculated by multiplying the CF and the FF for each plate and weld. The initial RT_{NDT}, the calculated ΔRT_{NDT} and the calculated margins were then added to get the new value of ART. Table 3-3 shows the 54 EFPY values of ART. As indicated in the table, the plates remain the limiting subcomponents rather than the welds; and Plate 1-14 remains the limiting plate. All calculated values are well below the 200 °F suggested in Section 3 of Regulatory Guide 1.99 and are thus acceptable for the period of extended operation. The TLAA for RT_{NDT} is thus projected through the period of extended operation in accordance with 10CFR54.21(c)(1)(ii).

3.1.5 Reactor Vessel Circumferential Welds

BWRVIP-74 reiterated the recommendation of BWRVIP-05 that RPV circumferential welds could be exempted from examination. The NRC SER for BWRVIP-74 agreed, but required that plants apply for this relief request individually. The relief request should demonstrate that at the expiration of the current license, the circumferential welds satisfy the limiting conditional failure probability for circumferential welds in the (BWRVIP-05) evaluation. VYNPS has applied for the relief request (Ref. 6.5.22), but has only evaluated the welds to the end of the current operating license. The changes in metallurgical conditions expected over the period of extended operation require additional analysis for 54 EFPY to extend the reactor vessel circumferential weld inspection relief request. The evaluations have been extended to 54 EFPY and the results are presented here.

VYNPS requested relief from the inspection of reactor vessel circumferential welds (Ref. 6.5.22). The VYNPS submittal included an analysis that showed that the reactor vessel parameters after 32 EFPY were within the NRC's 32 EFPY bounding Chicago Bridge & Iron (CBI) vessel parameters from the BWRVIP-05 SER. As such, there is a lower conditional probability of failure for circumferential welds at VYNPS than that stated in the NRC's Final Safety Evaluation Report of BWRVIP-05.

Table 3-6 reproduces the table from the submittal, with an added column providing the values for 54 EFPY. Consistent with earlier submittals, this table conservatively uses surface fluence rather than ¼T fluence, so the resulting change in RT_{NDT} is slightly higher than shown in Section 3.1.4 of this report.

The VYNPS reactor pressure vessel circumferential weld parameters at 54 EFPY will remain within the NRCs (64 EFPY) bounding CBI vessel parameters from the BWRVIP-05 SER. As such, the conditional probability of failure for circumferential welds remains below that stated in the NCR's Final Safety Evaluation of BWRVIP-05. Therefore, this analysis has been projected for the period of extended operation per 10 CFR 54.21 (c)(1)(ii). VYNPS will officially request this relief request for the period of extended operation.

3.1.6 Reactor Vessel Axial Welds

Applicants must evaluate axially oriented RPV welds to show that their failure frequency remains below the $5x10^{-6}$ calculated in the BWRVIP-74 SER. The SER states that an acceptable way to do this is to show that the mean RT_{NDT} of the limiting axial beltline weld at the end of the period of extended operation is less than the values specified in Table 1 of that SER. Table 3-7 of this report reproduces the 32 EFPY and 64 EFPY data from the SER, and adds the VYNPS data for 32 and 54 EFPY. The table shows that the VYNPS mean RT_{NDT} is well below that in the SER, and thus the VYNPS axial weld failure frequency is well below the acceptable limit of $5x10^{-6}$. Therefore, this analysis has been projected for the period of extended operation per 10 CFR 54.21(c)(1)(ii).

3.1.7 Surveillance Specimen Testing

10CFR50, Appendix H, requires a reactor vessel materials surveillance program that can verify the TLAAs for vessel embrittlement discussed above, and modify the projections if needed, based on measured embrittlement of actual material samples. The first VYNPS surveillance capsule was withdrawn from the vessel after approximately $4.3x10^{16}$ n/cm^2 and tested. The results are presented in Battelle Columbus Laboratories report BCL-585-84-3 (Ref. 6.5.18). The data agree with the data in the NRC RVID2 database (Ref. 6.2.1). The data in the capsule report showed the decrease in plate C_VUSE to be 2.5 times that predicted by RG 1.99.

VY re-evaluated the raw data points determined by Battelle using an EPRI hyperbolic tangent curve fitting routine. This resulted in revised unirradiated and irradiated C_VUSE results for both the plate and weld specimen. The new analyses still resulted in larger C_VUSE reductions than predicted by RG 1.99 for the plate, but not as large as predicted by the original Battelle report. The revised analyses resulted in a decrease in weld C_VUSE very close to the RG 1.99 predicted decrease opposed to the increase in the original analyses. The revised analyses were submitted to the NRC in VYNPS letter BVY 93-146. (Ref. 6.5.14) Table 3-2 summarizes both the Battelle report and BYV 93-146 for comparison.

In March 2003 (Ref. 6.5.16), VYNPS submitted a license amendment request to remove the plant-specific reactor vessel surveillance requirements from the Technical Specifications and replace them with the BWRVIP Integrated Surveillance Program (ISP). The NRC approved this submittal as Amendment 218 to the VYNPS license (Ref. 6.2.11). This amendment removed the plant-specific surveillance capsule requirements.

For the period of extended operation, VYNPS will continue to participate in the BWRVIP Integrated Surveillance Program (BWRVIP-74, 86 and 116). VYNPS will periodically adjust the projected values of fluence, C_VUSE and RT_{NDT} as additional surveillance capsule results are collected by the BWRVIP Integrated Surveillance Program (BWRVIP Reports 78, 86, and 116). See the Reactor Pressure Vessel Monitoring Program in LRPD-02, Aging Management Program Evaluation Report, for additional details. Surveillance capsule data is not a TLAA.

Table 3.1
VYNPS Charpy Upper Shelf Energy Data for 54 Effective Full-Power Years (EFPY)

Material Description						32 EFPY Projection			54 EFPY Projection		
Reactor Vessel Beltline Region Plates	Material Type	Plate ID	Heat #	%Cu	Initial USE [1]	1/4 T fluence (10^{19} n/cm^2)	% Drop in USE	USE (1/4 T)	1/4 T fluence (10^{19} n/cm^2)	% Drop in USE	USE (1/4 T)
Plate 1-17	A533B	330	C2640-1	0.12	EMA	0.017	8.00%	EMA	0.0398	9.79%	EMA
Plate 1-16	A533B	329	C2653-3	0.13	EMA	0.017	8.38%	EMA	0.0398	10.3%	EMA
Plate 1-15	A533B	328	C3116-2	0.14	EMA	0.017	8.76%	EMA	0.0398	10.7%	EMA
Plate 1-14	A533B	327	C3017-2	0.11	89	0.017	7.62%	82.2	0.0398	9.32%	67.7
Reactor Vessel Beltline Region Welds	Weld Type	Plate ID	Heat #	%Cu	Initial USE [2]	1/4 T fluence (10^{19} n/cm^2)	% Drop in USE	USE (1/4 T)	1/4 T fluence (10^{19} n/cm^2)	% Drop in USE	USE (1/4 T)
Welds	SMAW	955	NA/W-A	0.04	EMA	0.017	6.86%	EMA	0.0398	8.39%	EMA

References:

1 The material description and 32 EFPY projections came from the NRC Reactor Vessel Integrity Database (RVID2), Ref. 6.2.1.

2 The 54 EFPY projection uses the vessel ID fluence given in Ref. 6.5.44 converted to 1/4T fluence using the RG 1.99 formula. Vessel thickness = 5.064 inches (Ref. 6.5.3)

3 The 54 EFPY % drop in use is calculated from the fluence and the formulae for the curves in RG 1.99.

Table 3-2
VYNPS Surveillance Capsule #1 Test Data
(Discussed in Section 3.1.4)

Material Description	Plate, Longitudinal		Weld, NA	
	Capsule Report	BVY 93-146	Capsule Report	BVY 93-146
Heat Number[1]	C3017-2	C3017-2	3P4966	3P4966
Capsule No.	30 deg	30 deg[2]	30 deg	30 deg[2]
Lead Factor	0.83	0.83[2]	0.83	0.83[2]
Copper %	0.106	0.11	0.030	0.030
Neutron fluence (10^{19} n/cm^2)	0.0043	0.0043	0.0043	0.0043
fluence factor	0.06[3]	0.06[3]	0.06[3]	0.06[3]
Measured Initial USE	148	137	107	125
Measured Radiated USE	128	126	122	119
Drop in Use	20	11	-15	6
% Drop in USE	13.5%[4]	8.0%	-14.02%[4]	4.80%
RG 1.99 Predicted % drop in USE	5.39%[4]	5.50%	4.68%[4]	4.68%
USE correction factor	2.51[4]	1.45	1.00[4,5]	1.03

1 The heat number is from RVID2 (Ref. 6.2.1), it is not used in any calculation.
2 The capsule number and lead factor are from the capsule report, Ref. 6.5.18. They are not used in any calculation.
3 The fluence factor is not given in either report. It is calculated here using the fluence and the formula in RG 1.99.
4 The % drop in USE and the RG 1.99 predicted % drop in USE, and USE correction factor were not in the capsule report. They have been calculated here using the data above and the formulae for the curves in RG 1.99.
5 USE correction factor was set =1 as measured data showed an increase in USE.

Table 3-3
VYNPS RT_{NDT} for 32 and 54 Effective Full-Power Years (EFPY)

	Material Description								32 EFPY						54 EFPY					
Reactor Vessel Beltline Region Location (Beltline ID)	Base Metal	Plate ID	Heat #	%Cu	%Ni	Initial RT_{NDT} (Deg F)	σ_u	Chemistry Factor	1/4T fluence (10^{19} n/cm^2)	Fluence Factor	ΔRT_{NDT} (Deg F)	σ_Δ	Margin (Deg F)	Adjusted RT_{NDT} (Deg F)	1/4T fluence (10^{19} n/cm^2)	Fluence Factor	ΔRT_{NDT} (Deg F)	σ_Δ	Margin (Deg F)	Adjusted RT_{NDT} (Deg F)
Location Unknown 1-17	A533B	330	C2640-1	0.12	0.61	0.0	0	83.2	0.0170	0.155	12.9	6.5	12.9	25.8	0.0398	0.258	21.5	10.7	21.5	42.9
Location Unknown 1-16	A533B	329	C2653-3	0.13	0.59	0.0	0	90.7	0.0170	0.155	14.1	7.0	14.1	28.2	0.0398	0.258	23.4	11.7	22.8	46.8
Location Unknown 1-15	A533B	328	C3116-2	0.14	0.66	-10.0	0	101.5	0.0170	0.155	15.8	7.9	15.8	21.5	0.0398	0.258	26.2	13.1	25.6	42.4
Location Unknown 1-14	A533B	327	C3017-2	0.11	0.63	30.0	0	74.5	0.0170	0.155	11.6	5.8	11.6	53.1	0.0398	0.258	19.2	9.6	18.8	68.4
Reactor Vessel Beltline Region Location (Beltline ID)	Flux type	Weld ID	Heat #	%Cu	%Ni	Initial RT_{NDT} (Deg F)	σ_u	Chemistry Factor	1/4T fluence (10^{19} n/cm^2)	Fluence Factor	ΔRT_{NDT} (Deg F)	σ_Δ	Margin (Deg F)	Adjusted RT_{NDT} (Deg F)	1/4T fluence (10^{19} n/cm^2)	Fluence Factor	ΔRT_{NDT} (Deg F)	σ_Δ	Margin (Deg F)	Adjusted RT_{NDT} (Deg F)
Welds [1]	SMAW	955	NA/W-A	0.04	1.00	0	13	54	0.0170	0.155	8.4	10.0	27.3	35.7	0.0398	0.258	13.9	10.0	29.3	43.4
Welds [2]	SMAW	955	NA/W-A	0.04	1.00	0	0	54	0.0170	0.155	8.4	4.2	8.4	16.8	0.0398	0.258	13.9	7.0	13.6	27.9

1 This line mimics RVID2 and uses override values for σ_u and σ_Δ. Results in conservative margin.

2 This line mimics NEDC-33090P and uses 0 for σ_u and calculates σ_Δ per RG 1.99, consistent with the way RVID2 calculates the plates.

3 The 54 EFPY projection uses the vessel fluence as determined in Section 3.1.1 of this report.

4 The initial RT_{NDT} values are from the Structural Integrity Associates report attached to BVY-00-113. These values supersede RVID2, as agreed to by the NRC in their SER (Ref. 6.2.33).

Table 3-4
Equivalent Margin Analysis for VYNPS Plate Material USE

	32 EFPY CLTP[1]	33 EFPY CPPU[2]	54 EFPY CPPU[3]
Surveillance Plate % Cu	0.11%	0.11%	0.11%
Surveillance Plate Fluence (10^{19} n/cm^2)	4.49E+16	4.49E+16	4.49E+16
Surveillance Plate Measured Decrease	8.03%	8.03%	8.03%
RG 1.99 Predicted Decrease	5.55%	5.55%	5.55%
Ratio of Measured to Predicted	1.448	1.448	1.448
Beltline Plate % Cu	0.14%	0.14%	0.14%
32 EFPY 1/4T fluence (10^{19} n/cm^2)	0.0221	0.0235	0.0398[4]
RG 1.99 Predicted Decrease	9.4%	9.5%	10.7%[5]
Adjusted % Decrease	13.5%	13.8%	15.5%[6]
Limiting % Decrease	21.0%	21.0%	23.5%[7]
Plate Acceptable	Yes	Yes	Yes

All of the above decreases are less than the 23.5% decrease in the bounding equivalent margin analysis, so the analysis conclusions apply to the vessel plates.

1 The 32 EFPY, Current Licensed Thermal Power (CLTP) column is from NEDC-33090P, Table 3-1a. (Ref. 6.5.2) Note that as part of the power uprate, the capsule fluence and the 32 EFPY fluence were recalculated using neutron transport theory consistent with RG1.190. Hence the numbers vary slightly from those given in table 3-2.
2 The 33 EFPY, Constant Pressure Power Uprate (CPPU) column is from NEDC-33090P, Table 3-1c (Ref. 6.5.2)
3 The 54 EFPY, CPPU column is created here. The surveillance capsule data is the same as the first two columns, other values are discussed in the following footnotes.
4 The 54 EFPY 1/4T fluence was calculated in Section 3.1.1 of this report.
5 The RG 1.99 predicted value was calculated using the formula for the curves in RG 1.99.
6 The adjusted decrease equals the product of the RG 1.99 prediction (10.7%) and the surveillance capsule ratio of measured to predicted (1.448).
7 The limiting percent decrease for 54 EFPY is 23.5% per BWRVIP-74 (Ref. 6.4.11) as approved by the NRC in their SER (Ref. 6.2.24).

Table 3-4 (continued)
Equivalent Margin Analysis for VYNPS Weld Material USE

	32 EFPY CLTP[1]	33 EFPY CPPU[2]	54 EFPY CPPU[3]
Surveillance Weld % Cu	0.03%	0.03%	0.03%
Surveillance Weld Fluence (10^{19} n/cm^2)	4.49E+16	4.49E+16	4.49E+16
Surveillance Weld Measured Decrease	4.80%	4.80%	4.80%
RG 1.99 Predicted Decrease	4.77%	4.77%	4.77%
Ratio of Measured to Predicted	1.005	1.005	1.005
Beltline Weld % Cu	0.04%	0.04%	0.10%[4]
32 EFPY 1/4T fluence (10^{19} n/cm^2)	0.0221	0.0235	0.0398[5]
RG 1.99 Predicted Decrease	7.32%	7.43%	11.19%[6]
Adjusted % Decrease	7.36%	7.47%	11.24%[7]
Limiting % Decrease	34.0%	34.0%	39.0%[8]
Weld Acceptable	Yes	Yes	Yes

All of the above decreases are less than the decrease in the bounding equivalent margins analysis, so the analysis conclusions apply to the vessel welds.

1 The 32 EFPY, CLTP column is from NEDC-33090P, Table 3-1b. (Ref. 6.5.2). Note that as part of the power uprate, the capsule fluence and the 32 EFPY fluence were recalculated using neutron transport theory consistent with RG1.190. Hence the numbers vary slightly from those given in table 3-2.
2 The 33 EFPY, CCPU column is from NEDC-33090P, Table 3-1d (Ref. 6.5.2)
3 The 54 EFPY, CPPU column is created here. The surveillance capsule data is the same, other values are discussed in the following footnotes.
4 A maximum weld copper content of 0.1% was conservatively used, consistent with the original equivalent margin evaluations in BVY 93-146
5 The 1/4T fluence was calculated in Section 3.1.1 of this report.
6 The RG 1.99 predicted value was calculated using the formula for the curves in RB 1.99.
7 The adjusted decrease equals the product of the RG 1.99 prediction (11.19%) and the surveillance capsule ratio of measured to predicted (1.005).
8 The limiting percent decrease for 54 EFPY for welds is 39% per BWRVIP-74 (Ref. 6.4.11) as approved by the NRC in their SER (Ref. 6.2.24).

Table 3-5
VYNPS P-T Curve Bases, (Current 32 EFPY and 54 EFPY)

	Material Description					32 EFPY P-T Curve Bases						54 EFPY					
Reactor Vessel Beltline Region Location (Beltline ID)	Initial RT_{NDT} (Deg F)	σ_u	Chemistry Factor	Location	Thickness (inches)	fluence (10^{19} n/cm^2)	Fluence Factor	ΔRT_{NDT} (Deg F)	σ_Δ	Margin (Deg F)	Adjusted RT_{NDT} (Deg F)	fluence (10^{19} n/cm^2)	Fluence Factor	ΔRT_{NDT} (Deg F)	σ_Δ	Margin (Deg F)	Adjusted RT_{NDT} (Deg F)
Location Unknown 1-14	30.0	0	74.5	ID	0 (5.06)	1.24E+18	0.461	34.3	17.0	34.0	98.3	5.39E+17	0.305	22.7	11.3	22.7	75.4
Location Unknown 1-14	30.0	0	74.5	¼ T	1.3	9.15E+17	0.399	29.7	14.9	29.7	89.5	3.98E+17	0.258	19.2	9.6	19.2	68.5
Location Unknown 1-14	30.0	0	74.5	¾ T	3.8	4.99E+17	0.292	21.8	10.9	21.8	73.5	2.17E+17	0.181	13.5	6.7	13.5	56.9

1 Further information on Location 1-14 is found in Table 3-3.

2 The basis for the current P-T limits (32 EFPY) are found in Reference 6.2.11.

3 The 54 EFPY projection uses the vessel fluence as determined in Section 3.1.1 of this report.

Table 3-6
VYNPS RPV Circumferential Shell Welds

Parameter Description	USNRC 32 EFPY Bounding Parameters	VYNPS Beltline Circ Weld 32 EFPY	USNRC 64 EFPY Bounding Parameters	VYNPS Beltline Circ Weld 54 EFPY
Initial (unirradiated) reference temperature (RT_{NDT}), °F	-65	-70	-65	0
Neutron fluence at the end of the requested relief period (Peak Surface Fluence Entire Beltline)	5.1×10^{18} n/cm^2	2.99×10^{17} n/cm^2	1.02×10^{19} n/cm^2	5.39×10^{17} n/cm^2 [2]
Fluence Factor (calculated per RG 1.99 based on fluence in previous line.)	0.812	0.219	1.006	0.305
Weld Copper content, %	0.10	0.04	0.10	0.04[3]
Weld Nickel Content	0.99	1.00	0.99	1.00[3]
Weld Chemistry Factor (CF)	109.5	54	109.5	54[3]
Chemistry Factor times Fluence Factor	88.9	11.8	110.1	16.5
Margin (Implied) , °F	20.6	0.0	25.5	16.5
Increase in reference temperature (ΔRT_{NDT}), °F	109.5	11.8	135.6	32.9
Mean adjusted reference temperature (ART), °F = $RT_{NDT} + \Delta RT_{NDT}$	44.5	-58.2	70.6	32.9

1 The first column is from Table 2.6-4 of the NRC SER for BWRVIP-05 (Ref. 6.2.27).
2 The second column is from BVY 03-83 (Ref. 6.5.22)
3 The third column is from Table 2.6-5 of the NRC SER for BWRVIP-05.
4 The fourth column is new material for this report.

Table 3-7
VYNPS RPV Axial Shell Welds

Parameter Description	USNRC Limiting Plant-Specific Data	VYNPS Data for axial weld	USNRC Limiting Plant-Specific Data	VYNPS Data for axial weld
EFPY	32	32	64	54
Initial (unirradiated) reference temperature (RT_{NDT}), °F	-30	0	-30	0
Neutron Fluence	6.90E+18	2.99E+17	1.38E+18	5.39E+17
FF = Fluence Factor	0.896	0.219	1.089	0.305
Weld Copper content, %	0.10%	0.04%	0.10%	0.04%
Weld Nickel Content, %	1.08%	1.00%	1.08%	1.00%
CF = Chemistry Factor	135.0	54	135.0	54.00
Increase in reference temperature (ΔRT_{NDT}), °F = FF * CF	121.0	11.8	147.1	16.5
Mean adjusted reference temperature (ART), °F = $RT_{NDT} + \Delta RT_{NDT}$	91.0	11.8	117.1	16.5

Column 1 is from Table 2.6-4 of BWRVIP-05 SER.
Column 2 is generated from data in this report with some calculations using that data. No previously submitted data was located.
Column 3 is from Table 2.6-5 of BWRVIP-05 SER.
Column 4 is generated from data in this report with some calculations using that data

3.2 Metal Fatigue

LRPD-04 describes and evaluates fatigue evaluations that meet the definition of TLAA for Class 1 and non-Class 1 mechanical components. Cumulative usage factors have been documented and the actual numbers of design transient cycles have been projected to 60 years. An adequate program is in place to track cycles and to provide corrective actions if limits are approached. For details of fatigue TLAA evaluations, see LRPD-04, TLAA – Mechanical Fatigue.

3.3 Environmental Qualification of Electrical Equipment

This section provides the evaluation of TLAA for EQ components.

3.3.1 Background

For certain important-to-safety electrical components, operating plants must meet the requirements of 10 CFR 50.49 (Ref. 6.1.7) which defines the scope of components to be included, requires the preparation and maintenance of a list of in-scope components and requires the preparation and maintenance of a qualification file.

Environmental Qualification Program Manuals, Volume I & II (Ref. 6.5.47), document the VYNPS EQ program basis and history. Volumes I & II include the general engineering documentation that provides the environmental qualification specifications for each component. Environmental Qualification Program Manual, Volume I, identifies and summarizes the EQ program activities and processes for implementing EQ requirements, which exist to assure that EQ components and EQ-related activities satisfy applicable industry and regulatory requirements. Volume I of the program manual documents the philosophy and methodology for meeting the requirements outlined in 10 CFR 50.49.

Also, Volume I (Section 4.0) addresses how the industry and VYNPS have dealt with equipment qualification as it has evolved since licensing of the first plants. Volume I directs the user to appropriate documents for detailed processes and implementation requirements as necessary. In addition, the EQ Program includes enhanced Maintenance and Surveillance (M&S) Program requirements for electrical components that require environmental qualification. The EQ M&S Program is not included in the EQ Program Manual but utilizes input from the Program Manual. The EQ M&S Program records are controlled and maintained at the plant per plant procedure AP0305 (Ref. 6.5.50). The EQ M&S Program is to assure that the environmental qualification of specific plant equipment remains valid through the expected life of that equipment and that any change to the expected life is recognized and reanalyzed. The EQ M&S Program provides controls for scheduling, performing, and documenting maintenance performed on EQ equipment.

The VYNPS EQ Master Equipment List (EQMEL) (Ref. 6.5.45) is a hard copy of the EQ Program Volume I document. The EQMEL is based on plant procedure AP 0092 (Ref. 6.5.46). Volume I, Section 7.0 and calculation VYC-193 (Ref. 6.5.47 and 6.5.48) define environmental service parameters for the environmental qualification of equipment. EQ documentation for equipment is maintained in qualification documentation review (QDR) packages. The index in Volume I, Section 1 provides the complete list of QDRs and Section 6 identifies the applicable QDR number for each component, which includes the analysis that supports the equipment qualification (Ref. 6.5.45). The system component evaluation worksheet (SCEW) is a form, which summarizes the environmental qualification data and qualification status for components or component types in the format requested by the NRC in IEB 79-01B (Ref. 6.2.5). The SCEW is part of the QDR (Tab B). EQ-related maintenance requirements (if necessary to maintain qualification) are in the (QDR) for the specific EQMEL item (Ref. 6.5.47 and 6.5.50).

In order to identify EQ commitments and exemptions to 10 CFR 50.49 for VYNPS, a review was conducted of the searchable plant databases that include the UFSAR, the operating license, and NRC correspondence. This review involved a search of this database to identify exemptions based on EQ TLAA that were granted pursuant to 10 CFR 50.12. The search criteria included key terms including “50.12”, “exemption*”, “EQ”, “environmental qualification”, “40 years”, “forty years”, “plant life”, “design life”, “qualified life”, “life of the plant”, “service period” and “operating term”. In addition to the database searches, sections of the Technical Specifications, the original Safety Evaluation Report, and selected correspondence, such as NRC generic letters and bulletins were reviewed. The review at VYNPS identified no exemptions based on TLAAs for EQ electrical components.

3.3.2 Environmental Qualification

All operating plants must meet the requirements of 10 CFR 50.49 for certain important-to-safety electrical components. 10 CFR 50.49 defines the scope of components to be included, requires the preparation and maintenance of a list of in-scope components, and requires that preparation and maintenance of a qualification file that includes component performance specifications, electrical characteristics and environmental conditions.

The EQ master equipment list (EQMEL) meets the 10 CFR 50.49(d) and IE Bulletin 79-01B requirements for the list of EQ components. The listing of electrical equipment in the EQMEL (Ref. 6.5.45) is controlled under the M&S Program Manual in accordance with plant procedure AP 0092 (Ref. 6.5.46). 10 CFR 50.49(e)(5) requires component replacement or refurbishment prior to the end of the designated life, unless additional life is established through ongoing qualification. The equipment included in the scope of EQ is based on specific screening criteria in 10 CFR 50.49(b).

As part of the EQ program, when EQ equipment or parts thereof have a limited life, the preventive maintenance process ensures the equipment or parts are replaced prior to expiration of the qualified life. If excess conservatism exists in the original qualified life determination, then reanalysis, which meets the requirements of 10 CFR 50.49, can extend the qualified life. The reanalysis utilizes standard EQ techniques (such as the Arrhenius method), and becomes part of the EQ documentation. Conservatism may exist in the ambient temperature of the equipment, in unrealistically low activation energy, and in the application of the equipment. The primary method used for reanalysis is reducing excess conservatism in equipment service temperatures by using temperature values closer to actual temperature in the area around the applicable equipment. These reanalysis methods for EQ components are discussed in NUREG-1801, Section X.E1 (Ref. 6.1.3).

EQ equipment will be reevaluated for the environmental service conditions that are applicable to the equipment (i.e., 60 years of exposure versus 40 years). The environmental service conditions considered are normal and accident. For electrical equipment exposed to a harsh environment, 10 CFR 50.49 requires consideration of all significant aging effects from normal service conditions. This includes the expected thermal aging effects from the temperature to which the device is normally exposed, wear/cycle aging (applicable to limited types of EQ equipment), and radiation aging effects during normal plant operation. 10 CFR 50.49 also requires evaluation of the effects from harsh environments to which the equipment could be exposed under accident conditions. In general, the harsh environments analyzed as part of the EQ Program are those caused by loss of coolant accidents (LOCA), high energy line breaks (HELBs) inside the reactor building, and HELBs outside the reactor building.

For EQ equipment with a qualified life less than the design life of the plant, "ongoing qualification" is a method of long-term qualification involving additional testing. Ongoing qualification or retesting, as described in IEEE Std. 323-1974 (Ref. 6.1.4), Section 6.6(1) or (2), is not considered a viable option, and there are no immediate plans to implement such an option. If this option becomes viable, ongoing qualification or re-testing would be performed in accordance with accepted industry and regulatory standards.

The evaluation of the environmental service conditions for the license renewal period requires a reevaluation of only the normal aging effects**.** Therefore, the normal effects from operation for a 60-year period instead of a 40-year period will be evaluated. Radiation aging effects, thermal aging effects, and wear/cycle aging effects, as applicable, are analyzed for the period of extended operation. The following sections describe each of these considerations in more detail.

3.3.3 Radiation Considerations

Before entering the period of extended operation, the additional normal dose for the license renewal period will be evaluated. Typically a normal does that is 1.5 times (i.e., 60 years/40 years) the dose for the 40-year period is evaluated for the extended period. The dose values used are based on equipment locations, applying either an inside reactor building value, outside reactor building value or application-specific value. The total integrated dose for the 60-year period is determined by adding the established accident dose to the newly determined 60-year normal dose for the device. If the device is qualified for this total integrated dose, no additional review is required.

If the increased normal dose results in a total integrated dose above the qualified dose for the component, a location-specific review may be required to determine a lower dose for the specific component. Other options include requiring component or part replacement prior to exceeding the qualified total dose or performing radiation surveys to determine actual operating dose and evaluating against this value.

VYNPS EQ Manual Volume I, Section 7.0 and VYC-193 (Ref. 6.5.47 and 6.5.48) provide normal and accident radiation environmental data used for evaluating the environmental qualification of electrical equipment subject to 10 CFR 50.49. In addition VYC2339, which is referenced in the QDRs, includes more detailed analysis of specific components.

Some components have been installed under a plant modification, and will not experience 60 years of radiation aging by the end of the license renewal period. In these cases, credit may be taken for less than 60 years of aging. Also, plant modifications that affect the normal dose, such as power uprate, will be addressed for impact on the EQ TLAAs.

3.3.4 Thermal Considerations

The average (ambient) temperatures inside the reactor building in occupied areas are 84°F and non-occupied areas are100°F with peak allowable temperatures 106°F and 120°F respectively. The average temperature inside the drywell is dependent upon elevation with a range of 150°F to 270°F. The peak allowable temperatures are in a range of 160°F to 280°F. The average (ambient) temperatures for the auxiliary/turbine building in occupied areas is 100°F with peak allowable temperature 105°F, except where equipment walkdowns or temperature monitoring have identified localized elevated temperatures. Section 7.0 of the EQ Program Manual provides an extensive plant area listing of (ambient) temperatures (Ref. 6.5.47). For components exposed to (ambient) temperatures lower than the given averages, temperature monitoring can be utilized to confirm lower than design temperatures exist in these areas, and on that basis, extend qualified lives through the license renewal period.

If extension of the 40-year qualified life is chosen rather than component replacement, it would be based upon re-evaluation of an aging analysis as defined by 10 CFR 50.49 and discussed in NUREG-1801 Section X.E1 (Ref. 6.1.3). The aging analysis will be revised, as applicable, to identify the maximum service life based on traditional EQ techniques such as the Arrhenius method.

Some components have been installed under a plant modification, and will not experience 60 years of thermal aging by the end of the license renewal period. In these cases, credit may be taken for less than 60 years of aging.

3.3.5 Wear/Cycles Considerations

EQ evaluations for the license renewal period will address wear/cycle aging qualification prior to entering the period of extended operation. For most components, this aging does not apply; however, for electromechanical equipment like solenoid valves there would be an associated number of cycles over 40 years. The number of cycles assumed for the license renewal period is 1.5 times (i.e., 60 years/40 years) the number established for the 40-year period.

Some components have been installed under a plant modification, and will not experience 60 years of cycling by the end of the license renewal period. In these cases, credit may be taken for less than 60 years of aging. Credit may also be taken for lower actual frequency of cycles.

3.3.6 GSI-168, *EQ of Electrical Components*

As discussed in SECY-93-049 (Ref. 6.1.6), the staff reviewed significant license renewal issues and found that several were related to environmental qualification. A key aspect of these issues was whether the licensing basis should be reassessed or enhanced in connection with license renewal, and whether this reassessment should be extended to the current license term. In late 1993, the Commissioners instructed the Staff that the current EQ licensing basis must be used in the license renewal period and that any EQ concerns identified by the staff during the review of EQ for license renewal should be evaluated for the effect on current licenses, independent of license renewal.

The NRC Staff's EQ Task Action Plan (EQ-TAP) was initiated to address the adequacy of current EQ practices. (Ref. 6.2.4) Upon completion of the EQ-TAP review, the Staff concerns focused on issues related to the adequacy of accelerated aging practices in existing qualifications, and the lack of a “feedback mechanism” in EQ programs (i.e., programmatic requirements to determine the current condition of EQ equipment so that it can be evaluated against the assumptions and parameters for qualification). The EQ-TAP was subsequently closed and the remaining open issues were incorporated into GSI-168 for management tracking purposes. The EQ-TAP review did not identify any generic safety issues related to these open issues. NRC guidance for addressing GSI-168 for license renewal is contained in a June 1998 letter to NEI (Ref. 6.2.29). In this letter, the NRC states:

> “With respect to addressing GSI-168 for license renewal, until completion of an ongoing research program and staff evaluations, the potential issues associated with GSI-168 and their scope have not been defined to the point that a license renewal applicant can reasonably be expected to address them at this time. Therefore, an acceptable approach described in the SOC is to provide a technical rationale demonstrating that the current licensing basis for EQ pursuant to 10 CFR 50.49 will be maintained in the period of extended operation. Although the SOC also indicates that an applicant should provide a brief description of one or more reasonable options that would be available to adequately manage the effects of aging, the staff does not expect an applicant to provide the options at this time.”

Consistent with previous NRC guidance, and RIS-2003-09 (Ref. 6.2.2) no additional information is required to address GSI-168 in a license renewal application (Ref. 6.1.2).

3.3.7 Summary

The VYNPS environmental qualification (EQ) of electrical components program **(Ref.** 6.5.46**)** manages component thermal, radiation and cyclical aging, as applicable, through the use of aging evaluations based on 10 CFR 50.49(f) qualification methods. As required by 10 CFR 50.49, EQ components not qualified for the current license term will be refurbished, replaced, or have their qualification extended prior to reaching the aging limits established in the evaluation. Aging evaluations for EQ components that specify a qualification of at least 40 years are considered TLAA for license renewal. The EQ program ensures that these analyses for EQ components are maintained in accordance with their qualification bases.

The VYNPS program is an existing program established to meet VYNPS commitments for 10 CFR 50.49. It is consistent with NUREG-1801, Section X.E1, "Environmental Qualification (EQ) of Electric Components."

The VYNPS program includes consideration of operating experience to modify qualification bases and conclusions, including qualified life. Compliance with 10 CFR 50.49 provides reasonable assurance that components can perform their intended function(s) during accident conditions after experiencing the effects of inservice aging. Consistent with NRC guidance provided in RIS 2003-09, no additional information is required to address GSI-168, "EQ of Electrical Components."

Based upon a review of the existing program and associated operating experience, continued implementation of the VYNPS environmental qualification of electrical components program provides reasonable assurance that the aging effects will be managed and that the in-scope EQ components will continue to perform their intended function(s) for the period of extended operation. The effects of aging will be managed by the VYNPS program in accordance with the requirements of 10 CFR 54.21(c)(1)(iii). (Ref. 6.1.5 and 6.5.47)

3.4 Concrete Containment Tendon Prestress

This section is not applicable as VYNPS does not have pre-stressed tendons in the containment building. Containment (Torus) and Torus Attached Piping

The torus and torus attached piping systems were analyzed for fatigue due to mechanical loadings as well as thermal and anchor motion. LRPD-04, Mechanical Fatigue addresses these analyses.

3.5 Metal Corrosion Allowance

Most pressure retaining components are constructed with a wall thickness in excess of minimum required wall thickness for that component. This excess wall thickness provides a corrosion allowance over the life of the component to assure that minimum wall thickness requirements are still met at end of life. If these corrosion allowances were meant to cover the original 40 year design life of the component, they could be considered TLAA. Individual corrosion allowances are discussed below. The results show that there are no analyses for corrosion allowances based on time-limited assumptions and hence no TLAA. Loss of material caused by corrosion of metal components will be managed for the period of extended operation.

3.5.1 Reactor Pressure Vessel

UFSAR Section 4.2.4.1 states “Although little corrosion of plain carbon or low alloy steels occurs at temperatures of 500°F to 600°F, higher corrosion rates occur at temperatures around 140°F. The 0.125-inch minimum thickness stainless steel cladding provides the necessary corrosion resistance during reactor shutdown and also helps maintain water clarity during refueling operations. Exterior exposed ferritic surfaces of pressure-containing parts have a minimum corrosion allowance of 1/32-inch. All carbon and low alloy steel nozzles exposed to the reactor coolant have a corrosion allowance of 1/16-inch. The vessel shape is designed to limit coolant retention pockets and crevices”.

Although the original reactor vessel corrosion allowances were conservative values intended to encompass 40 years of operation, no specific corrosion rate, and no specific analysis associated with these values has been identified. As such, there are no TLAA associated with these corrosion allowances. Loss of material from carbon steel and low alloy steel is an aging effect requiring management as identified in VYNPS Report AMRM-31, Aging Management Review of the Reactor Pressure Vessel. The Inservice Inspection Program and the Water Chemistry Control Program manage this effect as detailed in AMRM-31, Aging Management Review of the Reactor Pressure Vessel.

3.5.2 Recirculation Pump Casing

Section 4.3.4 of the UFSAR states: “The design objective for the recirculation pump casing is a useful life of 40 years, accounting for corrosion, erosion, and material fatigue.” Although the original corrosion allowances were conservative values intended to encompass 40 years of operation, no specific corrosion rate, and no specific analysis associated with these values has been identified. As such, there are no TLAA associated with these corrosion allowances.

Corrosion is an aging mechanism leading to loss of material. Loss of material is an aging effect evaluated in AMRM-33, Aging Management Review of the Reactor Coolant System. The recirculation pump casing is cast of austenitic stainless steel, which is inherently resistant to corrosion; consequently significant corrosion of this pump casing is not expected. The Inservice Inspection Program and the Water Chemistry Control Program manage this effect.

3.5.3 Main Steam Isolation Valve

UFSAR section 4.6.3, Description (of the MSIVs), states "The design objective for the valve is a minimum of 40 years' service at the specified operating conditions. The estimated operating cycles per year is 100 cycles during the first year and 50 cycles per year thereafter. In addition to minimum wall thickness required by applicable codes, a corrosion allowance of 0.120 inch minimum is added to provide for 40 years' services." Although the original corrosion allowances were conservative values intended to encompass 40 years of operation, no specific corrosion rate, and no specific analysis associated with these values has been identified. As such, there are no TLAA associated with these corrosion allowances.

Corrosion is an aging mechanism leading to loss of material. Loss of material is an aging effect evaluated in AMRM-33, "Aging Management Review of the Reactor Coolant System." The Water Chemistry Control - BWR program, the ASME Section XI Inservice Inspection Program and the System Walkdown program manage loss of material from these valves. .

3.5.4 HPCI System

Section 6.4.1 of the UFSAR states "The system is designed for a service life of 40 years, accounting for corrosion, erosion, and material fatigue". Although the original corrosion allowances were conservative values intended to encompass 40 years of operation, no specific corrosion rate, and no specific analysis associated with these values has been identified. As such, there are no TLAA associated with these corrosion allowances.

Corrosion and erosion are aging mechanisms that lead to loss of material. Loss of material for the HPCI system is identified as an aging effect requiring management in AMRM-05, "Aging Management Review of the High Pressure Coolant Injection System". The Water Chemistry Control – BWR program and the Flow Accelerated Corrosion Program manage loss of material due to corrosion for the HPCI system.

3.5.5 RCIC and HPCI Turbine Casings

The table on page C.2-52 of the UFSAR states

"2. The minimum wall thickness of the turbine casing shall be based on that to limit stress to the allowable working stress when subjected to design pressure plus corrosion allowance. Allowable stresses shall be in accordance with ASME B&PV Code, Section VIII."

Although the original corrosion allowances were conservative values intended to encompass 40 years of operation, no specific corrosion rate, and no specific analysis associated with these values has been identified. As such, there is no TLAA associated with these corrosion allowances.

Corrosion is an aging mechanism leading to loss of material. Loss of material from the RCIC turbine casing is addressed in AMRM-06, Aging Management Review of Reactor Core Isolation Cooling System. The Water Chemistry Control - BWR Program, the Periodic Surveillance and

Preventive Maintenance Program manage loss of material for the RCIC and HPCI turbine casings.

3.5.6 RCIC and HPCI Pumps

The table on page C.2-49 of the UFSAR states

"2. The minimum wall thickness of the pump shall be based on that to limit stress to the allowable working stress when subjected to design pressure plus corrosion allowance. Allowable stresses shall be in accordance with ASME B&PV Code, Section III."

Although the original corrosion allowances were conservative values intended to encompass 40 years of operation, no specific corrosion rate, and no specific analysis associated with these values has been identified. As such, there are no TLAA associated with these corrosion allowances.

Corrosion is an aging mechanism leading to loss of material. Loss of material from the RCIC pump casing is addressed in AMRM-06, Aging Management Review of Reactor Core Isolation Cooling System. Loss of material from the HPCI pump casing is addressed in AMRM-05, Aging Management Review of the High Pressure Coolant Injection System. The Water Chemistry Control – BWR Program, the Flow Accelerated Corrosion Program, and the Periodic Surveillance and Preventive Maintenance Program manage this aging effect.

3.5.7 RHR Heat Exchanger

The table on page C.2-55 of the UFSAR states "The minimum thickness of the following components shall be designed to contain the design pressure plus corrosion allowance.

A. Shell
B. Shell Cover
C. Channel Ring
D. Tubes."

Although the original corrosion allowances were conservative values intended to encompass 40 years of operation, no specific corrosion rate, and no specific analysis associated with these values has been identified. As such, there is no TLAA associated with these corrosion allowances.

Corrosion is an aging mechanism leading to loss of material. Loss of material from the RHR heat exchanger components is addressed in AMRM-02, Aging Management Review of Residual Heat Removal System. The Water Chemistry Control – Closed Cooling Water Program, the Systems Walkdown Program (external surfaces), and the Service Water Integrity Program(all surfaces) manage this aging effect.

3.6 <u>ASME Section XI Inservice Inspection</u>

All currently active Section XI Code relief requests submitted to the NRC by VYNPS are specific to the third ISI inspection interval, and thus by definition do not involve TLAA. Descriptions of each of these relief requests may be found in the ISI Program Procedure. (Ref. 6.5.58)

3.7 TLAA in BWRVIP Documents

BWR Vessel and Internals Project (BWRVIP) documents identify various potential TLAA. The TLAA applicable to VYNPS are described below.

3.7.1 BWRVIP-05 Reactor Vessel Axial Welds

BWRVIP-05 justified the elimination of reactor vessel circumferential welds from examination. BWRVIP-74 extended this justification to cover the period of license renewal. See BWRVIP-74 below and Section 3.1.6 above for review of the TLAA associated with this issue.

3.7.2 BWRVIP-18 Core Spray Internals

There are no TLAA identified in BWRVIP-18.

3.7.3 BWRVIP-25 Core Plate

This document concerns two aging effects for core plate rim hold-down bolts: loss of preload and cracking.

The calculation of loss of preload on the core plate rim hold-down bolts is a TLAA (Ref. 6.2.25). BWRVIP-25 calculated the loss of preload for these bolts for forty years. Appendix B to BWRVIP-25 projected this calculation to 60 years, showing that the VYNPS bolts would experience only 5 to 19 percent loss of preload. This TLAA is thus projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

There is no TLAA associated with cracking of the core plate bolts. The inspection recommendations of BWRVIP-25 are intended to manage cracking of the core plate bolts for the period of extended operation. VYNPS has implemented (Ref. 6.5.43) the inspection requirements of BWRVIP-25 in the VYNPS BWR Vessel Internals Program, which will adequately manage cracking of the core plate rim hold down bolts for the period of extended operation.

3.7.4 BWRVIP-26 Top Guide

BWRVIP-26 calculated the minimum top guide fluence for 32 EFPY (40 years) as 4 $\times 10^{21}$ n/cm^2. Appendix C to BWRVIP-26 projected the calculation of the top guide fluence to 6 $\times 10^{21}$ n/cm^2 for 48 EFPY (60 years). (Ref. 6.2.26) BWRVIP-26 and the NRC SER for BWRVIP-26 consider this a TLAA.

This calculation confirms that every BWR exceeded the IASCC threshold after approximately 4 EFPY and must therefore inspect for IASCC. This analysis does not meet the criteria for a TLAA as there is no safety determination based on this analysis (the analysis does not justify performing less inspections).

The threshold for IASCC is 5$\times 10^{20}$ n/cm^2 (Refs. 6.4.4 and 6.2.26). The VYNPS top guide fluence will exceed this threshold. Therefore VYNPS must manage IASCC of the top guide assembly. VYNPS has implemented the inspection recommendations in BWRVIP-26 through the BWR Vessel Internals Program (Ref. 6.5.43). The BWR Vessel Internals Program will adequately manage the effects of aging on the top guide for the period of extended operation.

3.7.5 BWRVIP-27 SLC/Core ΔP

The BWRVIP-27 fatigue analysis of the SLC/core ΔP line for 60 years of operation is a TLAA. The NRC SER (Ref. 6.2.30) states that fatigue and the projected cumulative usage factors (CUF) should be addressed by each applicant who applies for license renewal. The VYNPS SLC/ΔP nozzle is a low alloy steel (A508 Cl2) nozzle. VYNPS reviewed the CLB fatigue analyses and CUFs of all components in VYNPS Report LRPD-04, "TLAA – Mechanical Fatigue". No fatigue analysis for the SLC/ΔP nozzle was found and no CUF for the nozzle was identified. As such, VYNPS has no TLAA associated with the SLC/ΔP nozzle. Cracking is an aging effect for this nozzle that is managed for the period of extended operation per VYNPS Aging Management Review Report MARM-32, Aging Management Review of the Reactor Vessel Internals.

3.7.6 BWRVIP-38 Shroud Support

The BWRVIP-38 fatigue analysis of the shroud support is a TLAA. Fatigue of the reactor vessel internals, including the shroud, are discussed in VYNPS Report LRPD-04, TLAA – Mechanical Fatigue. The CUFs for the shroud are based on the design basis transients and remain valid for the period of extended operation in accordance with 10CFR54.21(c)(1)(i).

BWRVIP-38 also identifies that the original stress analysis contains a bounding crack growth rate. It further points out that individual plants may seek approval to use a lower crack growth rate based on operating experience. If a plant has received approval to use a different crack growth rate, it may involve a TLAA. VYNPS has not sought nor received such approval and therefore has no TLAA relating to crack growth for the shroud support.

3.7.7 BWRVIP-41 Jet Pump Assembly

The NRC SER for BWRVIP-41 requires plant-specific evaluation of jet pump fatigue by each applicant. VYNPS addresses fatigue in Report LRPD-04, TLAA – Mechanical Fatigue. LRPD-04 found no fatigue analysis and no CUF for the VYNPS jet pumps. As such, VYNPS has no TLAA associated with the jet pumps. Cracking is an aging effect for these pumps that is managed for the period of extended operation per VYNPS Aging Management Review Report MARM-32, Aging Management Review of the Reactor Vessel Internals.

The SER for BWRVIP-41 identifies evaluation of thermal/radiation embrittlement of the jet pump cast austenitic stainless steel components as a TLAA if cracks exist in the components. If the applicant can show that cracks do not exist, "…loss of fracture toughness resulting from thermal and/or neutron embrittlement will not be a significant aging effect."

VYNPS has observed no cracking in the cast components of the jet pump assemblies. The BWRVIP has not reported cracks in these components. In fact, BWRVIP-41 states that cracks are not expected in these components. Therefore, this is not a TLAA. If cracking appears, VYNPS will follow the recommendations of the BWRVIP to manage that cracking.

3.7.8 BWRVIP-47 Lower Plenum

BWRVIP-47 identified fatigue analyses, especially of lower plenum pressure boundary components, as a TLAA. (Some plants have components whose CUF will exceed 1.0 during the period of extended operation). VYNPS addresses fatigue in Report LRPD-04, "TLAA –

Mechanical Fatigue". The only lower plenum CUF identified by LRPD-04 for VYNPS was a CUF for the CRD penetrations equal to 0.13. This CUF is determined by the allowed number of transients and as such remains valid for the period of extended operation per 10CFR54.21(c)(1)(i).

3.7.9 BWRVIP-48 Vessel ID Attachment Welds

The BWRVIP-48 fatigue analyses for various configurations of different vessel ID bracket attachments are TLAA. The analyses addressed VYNPS bracket configurations. (VYNPS has no unique bracket configurations.) Analysis of fatigue for 60 years showed that no CUFs are above 0.4. This analysis remains valid for the period of extended operation in accordance with 10CFR54.21(c)(1)(i).

3.7.10 BWRVIP-49, Instrument Penetrations

The BWRVIP-49 fatigue analysis for several configurations of instrumentation penetrations, including the VYNPS configuration, is a TLAA. Analysis of fatigue for 60 years showed that all CUFs are below 0.4. This analysis remains valid for the period of extended operation in accordance with 10CFR54.21(c)(1)(i).

3.7.11 BWRVIP-74, Reactor Pressure Vessel

BWRVIP-74 and the NRC SER for BWRVIP-74 (Ref. 6.2.24) discuss the following four TLAA.

1. Pressure/Temperature Curves

   The SER concludes "a set of P-T curves should be developed for the heatup and cooldown operating conditions in the plant at a given EFPY in the LR period." Section 3.1.2 and Table 3-5 address the VYNPS P-T curves.

2. Fatigue

   The SER states that the license renewal applicant should not rely solely on the analysis in BWRVIP-74, but should also verify that the number of cycles assumed in the original fatigue design is conservative. VYNPS Report LRPD-04, TLAA – Mechanical Fatigue addresses fatigue of the reactor pressure vessel.

   The SER also states that NRC staff concerns on environmental fatigue were not resolved and that each applicant should address environmental fatigue for the components covered by BWRVIP-74. VYNPS Report LRPD-04, TLAA – Mechanical Fatigue addresses environmentally assisted fatigue.

3. Equivalent Margins Analysis for RPV Materials with Charpy USE Less than 50 ft-lbs

   BWRVIP-74, addresses the percent reductions in Charpy USE for limiting BWR/3-6 plates and BWR non-Linde 80 submerged arc welds (23.5 percent and 39 percent, respectively). The NRC SER for BWRVIP-74 (Ref. 6.2.24) states that the applicant shall demonstrate that the percent reduction in USE for their beltline materials is less than the BWRVIP-74 values. Further, the SER states that the applicant shall demonstrate that the percent reduction of their surveillance weld and plate material is less than or equal to the values predicted by RG 1.99, Revision 2.

Section 3.1.3, Table 3-1 and Table 3-4 address Charpy USE for reactor pressure vessel materials. For VYNPS, the percent reduction in C_VUSE for the beltline materials remains well below the percent reduction in the equivalent margins analysis.

4. Material Evaluation for Exempting RPV Circumferential Welds from Inspection

   See Section 3.1.5, Section 3.1.6, Table 3-6 and Table 3-7 for a discussion of the reactor vessel welds.

3.7.12 BWRVIP-76 Core Shroud

BWRVIP-76, Appendix K, states that plant-specific analyses for shroud fatigue will be reviewed to determine if there is a TLAA. A review of the VYNPS plant-specific shroud analyses (Refs. 6.5.55, 6.5.56, and 6.5.57) identified one TLAA.

VYC-1362 (Ref. 6.5.55) contains GE report GE-NE-523-A005-0195, “Vermont Yankee Core Shroud Primary Stresses”. This report is a stress calculation and does not involve a TLAA.

VYC-1363 (Ref. 6.5.56) contains GE proprietary report GE-NE-523-A194-1294, which develops the seismic model and does the seismic analysis for the core shroud. This report is structural model calculation and does not involve a TLAA.

The VYNPS calculation (VYC-1364, Ref. 6.5.57) of the allowable interval between inspections for various core shroud welds (2 cycles for some and 1 cycle for others) uses the limit load analysis techniques described in ASME Code, Section XI to calculate crack growth, which is valid as long as total neutron fluence remains below $3x10^{20}$ n/cm^2. (Ref. 6.5.57) Extrapolation of neutron fluence data from References 6.5.11 and 6.5.44 show that shroud fluence will be approximately $1.5x10^{20}$ n/cm^2 at the end of the period of extended operation (54 EFPY). The extrapolation was based on $3.518x10^8$ MWH (Ref. 6.5.44) at 1583 MWt with a peak flux of $8.17x10^{10}$ n/cm^2/sec (Ref. 6.5.11) and 7.943E8 MWH (Ref. 6.5.44) at 1912 MWt with a peak flux of $9.67x10^{10}$ n/cm^2/sec (Ref. 6.5.44). This compares reasonably to $1.39x10^{20}$ n/cm^2 for 54 EFPY at 1583 MWt as calculated in Ref 6.5.11. Therefore, this calculation remains valid for the period of extended operation per 10 CFR 54.21(c)(1)(i).

3.8 Other Plant-Specific TLAA

3.8.1 Crane Load Cycles

In the late 1970s, the NRC requested all licensees of operating reactors to review their controls for handling heavy loads to determine the extent to which the guidelines of NUREG-0612 were satisfied, and to identify the changes and modifications that would be required to fully satisfy these requirements (Ref. 6.2.22). Licensee responses required verification that crane designs complied with the guidelines of Crane Manufacturer's Association of America (CMAA) Specification 70 and Chapters 2-1 and 2-2 of ANSI B30.2-1976, including the demonstration of equivalency of actual design requirements for instances where specific compliance with these standards is not provided.

VYNPS's response (Ref. 6.5.42) identified that NUREG-0612 applied only to the reactor building crane at VYNPS. The reactor building crane was designed and built by Whiting Corporation and has a main hook load rating of 110 tons, and an auxiliary hook load rating of 7.73 tons. The reactor building crane was modified in 1976 by replacing the original trolley with one that has a dual load path on the main hoist when used for shipping cask operation. The modification satisfies the intent of APCSB BTP 9-1 which called for the crane to be designed and fabricated to a number of industry standards, including ANSI B30.2 and CMAA-70. The staff's safety evaluation of this modification, as transmitted by letter from R. Reid (NRC) to R. Groce (Yankee Atomic) on January 28, 1977, approved the modifications, implicitly confirming compliance with CMAA-70. A subsequent review was deemed unnecessary.

CMAA-70 calculates allowable stress range based on joint category and service class, which in turn assumes a number of cycles. However, this is not a TLAA as the calculation does not specifically use the number of cycles. The minimum number of load cycles in CMAA-70 is 20,000, for Class A cranes, with a mean effective load factor range of 0.35-0.53. The reactor building crane at VYNPS has been reviewed in accordance with CMAA-70, and is conservatively classified as a Class A crane for this review. The total load cycles and mean effective load factors for this crane have been estimated for the period of extended operation. Even using conservative estimates, total load cycles are well below 20,000 and effective load factors are well below 0.53 (Table 3.9-1). Therefore, the crane allowable stress range remains valid through the period of extended operation.

Table 3.8-1 Summary of Crane Cycles and Mean Effective Load Factor (k)

	Best Estimate		**Conservative**	
	Cycles	**k**	**Cycles**	**k**
Reactor Building Crane	3007	0.40	7590	0.40

Details of how this estimate was derived are provided in Attachment 4.

3.8.2 Reflood Thermal Shock of the Reactor Vessel Internals

UFSAR Section 3.3.5.4 addresses reflood thermal shock of the reactor vessel internals (core shroud). This evaluation of thermal shock is a TLAA as it is based on the shroud receiving a maximum integrated neutron fluence of 2.7×10^{20} n/cm^2 (greater than 1 MeV) by the end of plant life. The value of 2.7×10^{20} n/cm^2 is a generic value that bounds all BWRs. To show that

VYNPS remains bounded for the period of extended operation, it is adequate to show that shroud fluence for 54 EFPY remains below $2.7x10^{20}$ n/cm^2.

The peak shroud flux was calculated (Ref. 6.5.44) for the extended power uprate at $9.67x10^{10}$ n/cm^2-sec. Integrating this and the pre-uprate flux from Ref. 6.5.11 gives an end of life shroud fluence of $1.5x10^{20}$ n/cm^2. This value remains below the 2.7 $x10^{20}$ n/cm^2 value used in the evaluation discussed in the UFSAR. As such, this TLAA remains valid for the period of extended operation in accordance with 10CFR54.21(c)(1)(i).

3.8.3 Reflood Thermal Shock of the Reactor Vessel

Section B.4.9 of the VYNPS UFSAR refers to a thermal shock analysis performed on a representative GE BWR reactor vessel. This analysis is in GE topical report NEDO-10029, An Analytical Study in Brittle Fracture of GE-BWR Vessel subject to the Design Basis Accident (LOCA). NEDO-10029 also appears in Table 1.10 of the VYNPS UFSAR.

VYNPS has reviewed this analysis and determined that it is not a TLAA. It does not satisfy 10 CFR 54.3 (4) in that this analysis was not used by the licensee in making any safety determination. Rather, this analysis was prepared by GE in 1969 to answer concerns of the ACRS.

In the 1969 time frame, thermal shock was a concern for all commercial reactor vessels. Regulatory Guide 1.2, Thermal Shock to Reactor Pressure Vessels, addressed these concerns. As more information was developed, it became clear that this was a concern for Pressurized Water Reactors and not for Boiling Water Reactors. Regulatory Guide 1.2 was withdrawn. The withdrawal notice says Regulatory Guide 1.2 was superceded by 10CFR50.61, Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events, and by Regulatory Guide 1.54, Format and Content of Plant-Specific Pressurized Thermal Shock Safety Analysis Reports for Pressurized Water Reactors. These two replacement documents apply only to PWRs. There are currently no requirements for analysis of thermal shock in BWRs.

Brittle fracture of BWR vessels is addressed in BWRVIP-05. These analyses demonstrate that the probability of brittle fracture is acceptably low for BWR vessels, low enough to justify reduced inspection of the circumferential welds. This analysis included Level C and Level D events (including LOCA) as well as cold overpressure events. The analyses demonstrated that LOCAs were not limiting events because conditions that cause rapid vessel cooling also cause rapid depressurization. The NRC SER and SER supplement for BWRVIP-05 agrees that the probability of failure of a BWR vessel is sufficiently low that 100% inspection of the axial and circumferential welds is not required.

Reactor vessel neutron embrittlement (C_VUSE, RT_{NDT}, P-T limits, circumferential and axial welds) have been addressed as TLAA based on accumulated fluence over the life of the plant. These TLAA address all required aspects of the BWR vessel. As such there is no additional significant information in NEDO-10029 requiring evaluation and it is not a TLAA.

3.8.4 Feedwater Nozzle Crack Growth

Cracks were identified in the reactor vessel main feedwater nozzles in the mid 1970s. Modifications to the plant, including removing the cracks by grinding and installing re-designed feedwater nozzle thermal sleeves, were completed in 1976 (Ref. 6.5.51). The maximum growth rate of feedwater nozzle cracks was evaluated by calculation VYC-1005 (Ref. 6.5.51). This calculation is *not* a TLAA as it does not justify the use of the feedwater nozzles for the life of the plant. Rather, it calculates an acceptable interval between ultrasonic inspections to measure actual crack growth. As committed to the NRC in BVY 01-02 (Ref. 6.5.52), the fatigue monitoring program as implemented in OP-4172 (Ref. 6.5.53) and AP-0145 (Ref. 6.5.54). counts the thermal cycles on the feedwater nozzles and requires re-inspection of the nozzles before the allowed number of cycles is reached.. Mitigation efforts (via chemistry control) to date have been very successful with little or no flaw growth noted (Ref. 6.5.51). VY will continue to mitigate cracking, monitoring cycles, and re-inspect as necessary throughout the period of extended operation.

3.8.5 Upset, Emergency and Faulted Conditions

Section C2.2.2 of Appendix C to the UFSAR, discusses the probability of upset, emergency, and faulted conditions occurring in 40 years. The definitions support the quantitative event classifications, and were never meant to be precise quantitative values. The probability of an event occurring in 40 years, P_{40}, was approximated only within an order of magnitude rather than calculated. Given the lack of any specific analysis, these probabilities are not TLAA.

3.8.6 Probability of a Steam Line Break

Section I.4 of Appendix I of the UFSAR calculates the steam line break probability for a 40 year plant life. The UFSAR (Section I.1) says these probabilities are being calculated to “point out the more critical components and locations within the reactor piping system, so that attention can be directed to those areas.” The small changes in system reliability due to the period of extended operation will affect all systems and will not change the relative system reliabilities. This is not a TLAA as no safety related decisions are made based on the results of this analysis.

4.0 Identification and Evaluation of Exemptions

Pursuant to 10 CFR 54.21(c)(2), an applicant for license renewal must provide (1) a listing of plant-specific exemptions granted pursuant to 10 CFR 50.12 that are in effect and based on TLAA, and (2) an evaluation of these exemptions to justify their continuation for the period of extended operation. This section identifies exemptions for VYNPS and concludes that no exemptions that remain in effect are based on TLAA.

As discussed in Section 2.0, the searchable computer database that includes the UFSAR and NRC correspondence was reviewed. Attachment 2 provides a listing of the exemptions that were identified and lists the identified references for the exemptions. In accordance with 10 CFR 54.21(c)(2), exemptions that are not in effect are not required to be discussed in the license renewal application. For evaluation purposes, exemptions that were found to remain in effect were grouped into the following categories:

- Fire Protection Requirement Exemptions
- Other 10 CFR Requirement Exemptions

The following sections discuss the exemptions in each category and identify whether the exemptions are based on TLAA.

4.1 Fire Protection Requirement Exemptions

The VYNPS implementation of Appendix R resulted in eleven exemption requests to Sections III.G, III.J and III.L of 10 CFR 50 Appendix R. These exemptions are discussed in the VYNPS Safe Shutdown Capabilities Analysis (Ref. 6.5.19) with additional references provided therein.

None of these exemptions are based on calculations or analyses that consider the effects of aging or involve time-limited assumptions defined by the current operating term and therefore are **not** based on TLAA.

4.1.1 Exemption from Section III.G.3.b, relief from installation of automatic fire suppression through the control room, (Section 6.1 of Ref. 6.5.19)

VYNPS requested an exemption from the 10 CFR 50, Appendix R, Section III.G.1 requirements to have a fixed fire suppression system in the control room on the basis that the existing fire protection features in the control room are equal in effectiveness to a fixed fire suppression system.

This exemption was granted because the control room is a unique area of the plant that is required to be continually occupied by the operators. In the event of a fire, manual fire suppression would be effective and prompt. Because the operators provide a continuous fire watch in the control room, a fixed fire suppression system is not necessary to achieve adequate fire protection.

This exemption is not based on calculations or analyses that consider the effects of aging or involve time-limited assumptions defined by the current operating term and therefore is **not** based on a TLAA.

4.1.2 Exemption from Section III.G.2.a, installation of 3-hour rated fire barriers for the RCIC room, (Section 6.2 of Ref. 6.5.19)

VYNPS requested an exemption from the 10 CFR 50, Appendix R, Section III.G.2.a because the door, stairwell and equipment hatch that provide RCIC fire separation are not 3-hour fire rated fire barriers.

This exemption was granted because the majority of the room is 3-hour fire rated, and the un-rated door, hatch, and stairs are heavy steel designed to withstand a high energy line break. The fire load is low, and the fire detection system is alarmed in the control room, allowing early dispatch of the fire brigade. The staff concluded that completing the 3-hour fire rating of all room components would not significantly increase the level of fire protection in this zone.

This exemption is not based on calculations or analyses that consider the effects of aging or involve time-limited assumptions defined by the current operating term and therefore is **not** based on a TLAA.

4.1.3 Exemption from Section III.G.2.b, various reactor building area fire suppression (Section 6.3 of Ref. 6.5.19)

VYNPS requested an exemption from the 10 CFR 50, Appendix R, Section III.G.2.b to the extent that the automatic fire suppression systems required by the code were not installed in several areas of the reactor building.

The exemption was granted because the staff concluded that the existing fire protection, combined with the proposed fire protection measures in the subject zones, would provide a level of fire protection equivalent to the technical requirements of Section III.G.2.b of Appendix R.

This exemption is not based on calculations or analyses that consider the effects of aging or involve time-limited assumptions defined by the current operating term and therefore is **not** based on a TLAA.

4.1.4 Exemption from Section III.G.2.b, 20 feet of separation in Reactor Building elevation 252 East and West (RB3 and RB4), (Section 6.4 of Ref. 6.5.19)

VYNPS requested an exemption from the 10 CFR 50, Appendix R, Section III.G.2.b to the extent that it requires the installation of an automatic fire suppression system in the area and to the extent that it requires 20 feet of separation free of intervening combustibles.

This exemption was granted because VY committed to install an early fire detection system, and several fire barriers. The NRC concluded that the fire load in the area was low and that a fire would develop slowly. Given the early warning detection system, and the new barriers, the fire could be extinguished manually before it damaged redundant equipment.

This exemption is not based on calculations or analyses that consider the effects of aging or involve time-limited assumptions defined by the current operating term and therefore is **not** based on a TLAA.

4.1.5 Exemption from Section III.G.2.b, 20 feet of separation in Reactor Building elevation 252, NW corner, (Section 6.5 of Ref. 6.5.19)

VYNPS requested an exemption for the northwest corner of the reactor building, RB-3 and RB-4, from the 10 CFR 50, Appendix R, Section III.G.2.b. to the extent that it requires 20 feet of separation free of intervening combustibles.

This exemption was granted because the low fire loading, early warning detection system, manual fire suppression systems, other fire barriers and the 18 feet of existing separation provided adequate fire protection.

This exemption is not based on calculations or analyses that consider the effects of aging or involve time-limited assumptions defined by the current operating term and therefore is **not** based on a TLAA.

4.1.6 Exemption from Section III.G.1.a, electrical repair (Section 6.6 of Ref. 6.5.19)

VYNPS requested an exemption from the 10 CFR 50, Appendix R, Section III.G.1.a to allow connection of a batter charger, and replacement of fuses following a fire in the cable vault/cable spreading area.

This exemption was granted based on the fact that the repairs were simple and quick, involving equipment staged in appropriate locations, with approved procedures in place. The repairs can be completed well before the systems are needed following a reactor scram.

This exemption is not based on calculations or analyses that consider the effects of aging or involve time-limited assumptions defined by the current operating term and therefore is **not** based on a TLAA.

4.1.7 Exemption from Section III.G.2.a, fire sealing on main steam and main feedwater line penetrations, (Section 6.7 of Ref. 6.5.19)

VYNPS requested an exemption from the 10 CFR 50, Appendix R, Section III.G.2 because the penetration seals on the main steam and feedwater lines were un-rated versus the required 3-hour rated seals.

This exemption was granted based on the subject penetration area being a high radiation area that is inaccessible to personnel during plant operation and hence free of transient combustibles. There are no other possible sources of fire. The staff concluded that the subject penetration provides adequate fire protection even with the unqualified seal, and that no particular enhancement would be gained if the existing seal were replaced with a qualified seal.

This exemption is not based on calculations or analyses that consider the effects of aging or involve time-limited assumptions defined by the current operating term and therefore is **not** based on a TLAA.

4.1.8 Exemption from Section III.L.3, use of Vernon Tie for alternative shutdown equipment, (Section 6.8 of Ref. 6.5.19)

VYNPS requested an exemption from the 10 CFR 50, Appendix R, Section III.L.3 to allow use of the Vernon tie-line as an alternative to the onsite emergency diesel generator for fire events involving the control room, the cable spreading room, and fire zones RB-1, RB-2, RB-3 and RB-4 when offsite power is not available.

This exemption was granted based on the staff's conclusion that the Vernon tie line provides an acceptable alternative to power from an onsite emergency diesel generator when normal sources of offsite power are not available for a fire in the control room, cable spreading room, or reactor building fire zones RB-1/2/3/4.

This exemption is not based on calculations or analyses that consider the effects of aging or involve time-limited assumptions defined by the current operating term and therefore is **not** based on a TLAA.

4.1.9 Exemption from Section III.L.2 and G.1.a, ADS and Low Pressure Injection use in fire areas where a offsite power is not available, (Section 6.9 of Ref. 6.5.19)

VYNPS requested an exemption from the 10 CFR 50, Appendix R, Section III.L.2 and III.G.1 to allow use of the automatic depressurization system (ADS) and low pressure injection (either core spray of low pressure coolant injection) as a means of achieving post-fire safe-shutdown conditions in fire zones RB-1, RB-2, RB-3, and RB-4 when offsite power is not available; i.e. high pressure injection is not available.

This exemption was granted based on the staff's conclusion that the detection and suppression capabilities in fire zones RB-1/2/3/4 would be adequate to protect against fire hazards in the zones contingent on VYNPS installing additional fire detection capability as committed. The staff further concluded that the revised shutdown strategy for fire zones RB-1/2/3/4 (use of ADS with either CS or LPCI) and the re-designation of these fire zones as areas requiring an alternative shutdown capability provide and acceptable level of safe-shutdown protection.

This exemption is not based on calculations or analyses that consider the effects of aging or involve time-limited assumptions defined by the current operating term and therefore is **not** based on a TLAA.

4.1.10 Exemption from Section III.J, emergency yard lighting, (Section 6.10 of Ref. 6.5.19)

VYNPS requested an exemption from the 10 CFR 50, Appendix R, Section III.J requirement to have 8-hour battery backup for lighting in the general yard and nitrogen storage area and instead use the security lighting as access lighting for these areas.

This exemption was granted based on the security lighting being powered from a separate power source and therefore not being subject to fire loss.

This exemption is not based on calculations or analyses that consider the effects of aging or involve time-limited assumptions defined by the current operating term and therefore is **not** based on a TLAA.

4.1.11 Exemption from Section III.G.2.c, Rockbestos cable in fire zone R, (Section 6.11 of Ref. 6.5.19)

VYNPS requested an exemption from the 10 CFR 50, Appendix R, SectionIII.G.2.c to use fire resistant cables with Rockbestos insulation instead of the code requirement to enclose the cables in a 1-hour rated fire barrier.

This exemption was granted when the NRC staff concluded that the fire resistant cables provided essentially the same protection that a 1-hour fire barrier would provide.

This exemption is not based on calculations or analyses that consider the effects of aging or involve time-limited assumptions defined by the current operating term and therefore is **not** based on a TLAA.

4.2 Other 10 CFR Requirement Exemptions

Two additional exemptions to the Code of Federal Regulations were identified. These exemptions involve the calculation of P-T limits using Code Case N-640 and the use of the alternate source term (AST) in plant analyses.

None of these exemptions are based on calculations or analyses that consider the effects of aging or involve time-limited assumptions defined by the current operating term and therefore none of these exemptions are based on TLAA.

4.2.1 Exemption from 10 CFR 50 Appendix G, Use of Code Cases N-588 and N-640 for calculating P/T limits

In December, 2000, VYNPS requested an exemption to use Code Case N-588 and N-640 to calculate the pressure/temperature curves for the technical specifications. In February of 2001, VYNPS amended the request to use just code case N-640.

In April, 2001 the NRC (Ref. 6.2.20) acknowledged the withdrawal of Code Case N-588 and approved the use of code case N-640 for determining the P/T limits. Case N-640 allows use of the K_{Ic} equation in place of the K_{Ia} equation to calculate the P/T curves. The NRC affirmed that knowledge gained since issuance of the code demonstrates the margin of safety to protect the public health and safety from potential reactor vessel failure is conservative using the k_{Ia} equation and still sufficient to ensure the structural integrity of the reactor vessel using the k_{Ic} equation.

This exemption is not based on calculations or analyses that consider the effects of aging or involve time-limited assumptions defined by the current operating term and therefore is **not** based on a TLAA.

4.2.2 Alternative Source Term (AST) Methodology

VY requested (Ref. 6.5.40) an exemption from 10CFR50.54(o) and 10CFR50, Appendix J, Option B, Sections III.A and Sections III.B. The request was to use an alternative source term as discussed in 10CFR50.67. Since that submittal, eleven supplemental submittals have been made and approval of this exemption is still anticipated by VYNPS.

The alternative source term calculation anticipated the requested power uprate to 1912 megawatts thermal, and operation at the maximum extended load line limit (MELLA) power-flow condition thus ensuring a bounding core isotopic inventory. This AST calculation, and associate accident re-analyses do not consider the effects of aging or involve time-limited assumptions defined by the current operating term and therefore the associated exemption is **not** based on TLAA.

5.0 Summary and Conclusions

This report identified and listed TLAA and exemptions that were potentially applicable to the VYNPS. The TLAA were evaluated using one or more of the three options in 10 CFR 54.21 (c)(1).

No exemptions that will remain in effect for the period of extended operation are based on TLAA.

6.0 References

6.1 Codes and Standards

6.1.1 10 CFR Part 54, *Requirements for Renewal of Operating Licenses for Nuclear Power Plants*

6.1.2 NUREG-1800, *Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants*, July 2001.

6.1.3 NUREG-1801, Rev. 0, Volumes 1 and 2, *Generic Aging Lessons Learned (GALL) Report*

6.1.4 IEEE Std. 323-1974, *Qualifying Class 1E Equipment for Nuclear Power Generating Stations*, The Institute of Electrical and Electronics Engineers, Inc., 1974.

6.1.5 *Requirements For Renewal of Operating Licenses For Nuclear Power Plants*, 10 CFR Part 54, Federal Register, Vol. 60 No. 88, Monday, May 8, 1995, (60 FR 22461), Final Rules, Includes the Statement of Considerations (SOC) for the Final Rule.

6.1.6 SECY-93-049, *Implementation of 10 CFR Part 54, ' Requirements for Renewal of Operating Licenses for Nuclear Power Plants*,' March 1, 1993

6.1.7 *Environmental Qualification of Electrical Equipment Important to Safety*, 10 CFR 50.49, Federal Register, Volume 48, No. 15, January 21, 1983.

6.2 NRC Documents

6.2.1 US NRC, Reactor Vessel Integrity Database (RVID), Version 2.0.1, July, 2000

6.2.2 NRC Regulatory Issue Summary 2003-09 Environmental Qualification of Low-Voltage Instrumentation and Control Cables, May 2, 2003

6.2.3 Reg. Guide 1.89, Rev. 1, Environmental Qualification of Certain Electrical Equipment Important to Safety for Nuclear Power Plants

6.2.4 Memo from L. Joseph Callan, Executive Director for Operations, NRC, to Chairman and Commissioners, USNRC, Subject: Report on the Status of the Environmental Qualification Task Action Plan, November 15, 1996

6.2.5 NRC IE Bulletin 79-01B, Environmental Qualification of Class IE Electrical Equipment, January 14, 1980, Office of Inspection and Enforcement, U.S. Nuclear Regulatory Commission

6.2.6 NUREG 0588, Rev. 1, Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment

6.2.7 Guidelines for Evaluating Environmental Qualification of Class IE Electrical Equipment in Operating Reactors (DOR Guidelines), Enclosure 4 to IE Bulletin 79-01B, U.S. Nuclear Regulatory Commission, Washington, DC, January 14, 1980

6.2.8 NRC Regulatory Issue Summary (RIS) 2003-09, May 2, 2003, Environmental Qualification of Low-Voltage Instrumentation and Control Cables

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6.2.13 NVY 82-66, Exemption pertaining to requirement for fixed fire suppression in the Control Room, 10 May 1982

6.2.14 NVY 86-240, Exemption from Appendix R to 10CFR50 Concerning Automatic Fire Suppression, Separation, and Repairs, 1 December 1986

6.2.15 NVY 89-137, Issuance of Exemptions to 10CFR50, Appendix R, Section III.J. Emergency Lighting, and Section III.G.2.a, Separation, 26 June 1989

6.2.16 NVY 97-128, Vermont Yankee Nuclear Power Station (TAC NOS. M95442 and M95149), 12 August 1997

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6.2.18 NVY 97-81, Vermont Yankee Nuclear Power Station (TAC NO. M95482), 9 June 1997

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6.2.22 Letter, USNRC to Vermont Yankee, Control of Heavy Loads at Nuclear Power Plants, 22 December 1980

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6.2.24 NRC letter from C.I. Grimes (NRC) to C. Terry, (BWRVIP Chairman), Acceptance for referencing of EPRI Proprietary Report TR-113596, “BWR Vessel and Internals Project, BWR Reactor Vessel Inspection and Flaw Evaluation Guidelines (BWRVIP-74) and Appendix A, ‘Demonstration of Compliance with the Technical Information requirements of the License Renewal Rule (10CRF54.21)’, 18 October 2001

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6.2.31 USNRC letter S.A. Richards to J.F. Klapproth, Safety Evaluation for NEDC-32983P, General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluation (TAC No. MA9891), MFN 01-050, September 14, 2001

6.2.32 USNRC letter, J. T. Wiggins to L. A. England, Acceptance for referencing of Topical Report NEDO-32205, Revision 1, '10CFR50 Appendix G Equivalent Margin Analysis for Low Upper Shelf Energy in BWR/2 through BWR/6 Vessels, 8 Dec 1993

6.2.33 NVY 01-46, NRC Letter with SER, R.M. Pulsifer to M.A. Balduzzi, Vermont Yankee Power Station – Issuance of Amendment RE: P/T Limit Curves (TAC NO. MB0764), May 4, 2001

6.3 Industry Documents

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6.4.3 BWRVIP-25, EPRI Report TR-107284, BWR Core Plate Inspection and Flaw Evaluation Guidelines (BWRVIP-25), December 1996

6.4.4 BWRVIP-26, EPRI Report TR-107285, BWR Top Guide Inspection and Flaw Evaluation Guidelines (BWRVIP-26), December 1996

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6.4.14 BWRVIP-116, EPRI Report 1007824, Integrated Surveillance Program (ISP) Implementation for License Renewal, July 2003

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6.5.1 VYNPS Updated Final Safety Analysis Report, Revision 17.

6.5.2 BVY 03-80, 9/10/2003, J.K.Thayer to USNRC Document Control Desk, Technical Specification Proposed Change No. 263, Extended Power Uprate

6.5.3 NEDC-33090P Rev 0, (Attachment 4 to Reference 6.5.2), September 2003, Safety Analysis Report for Vermont Yankee Nuclear Power Station Constant Pressure Power Uprate

6.5.4 Appendix A to Operating License DPR-28, Technical Specifications and Bases for Vermont Yankee Nuclear Power Station Vernon, Vermont Docket No. 50-271, Amendment # 208

6.5.5 Vermont Yankee Technical Requirements Manual (TRM), Revision 19

6.5.6 QAPM, Entergy Quality Assurance Program Manual, Effective Oct. 27, 2003

6.5.7 Vermont Yankee Nuclear Power Corporation Fire Hazards Analysis, Revision 5, 8/20/02

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6.5.15 BVY 93-107, L. A. Tremblay, Jr. to NRC Document Control Desk, Response to Request for Additional Information, GL 92-01 - Reactor Vessel Structural Integrity, September 24, 1993

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6.5.17 BVY 03-28, 1 April 2003, Fourth-Interval Inservice Inspection Program Plan, Fourth-Interval Inservice Inspection Pressure Test Program, and Request for Approval of ISI Relief Requests

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6.5.46 VYNPS Document, AP 0092, Environmental Qualification (EQ) Document Change Notification

6.5.47 VYNPS Document, Environmental Qualification Program Manuals, Volume I (Rev. 47) & II (Rev. 13)

6.5.48 VYC-193, Main (VY Design Basis Radiation Dose Calculation Specifications)

6.5.49 VYNPS EQ Database (information tool only at this time)

6.5.50 VYNPS Document, AP 0305 Rev 11, EQ Maintenance and Surveillance (M/S) Program

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6.5.55 VYNPS Calculation VYC-1362, Vermont Yankee Core Shroud Primary Stresses, Revision 1, 3/16/95

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6.5.58 VYNPS Procedure PP-7015, Vermont Yankee Inservice Inspection Program, Revision 3, 09/01/2003

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6.5.60 Technical Report TR-5319-2 (Teledyne Engineering Services), Plant Unique Analysis Report of the Torus Attached Piping for Vermont Yankee Nuclear Power Station, 30 September 1983

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6.5.62 MPR-751, Mark 1 Containment Program Augmented Class 2/3 Fatigue Evaluation Methods and Results for Typical Torus Attached and SRV Piping Systems, November, 1982

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6.5.64 Minor modification MM 2003-040, Reactor Building Auxiliary Hoist Upgrade

6.5.65 VYNPS Procedure, OP-2200, Operation of the Reactor and Turbine Bridge Cranes, Revision 17, LPC #11, 12/16/2003

6.6 VYNPS License Renewal Documents

6.6.1 LRPG-01, License Renewal Project Plan

6.6.2 VYNPS Report LRPD-02, Aging Management Program Evaluation Results

6.6.3 VYNPS Report LRPD-04, TLAA – Mechanical Fatigue

6.6.4 LRPG-08, TLAA and Exemption Evaluations

Attachment 1 - List of Potential TLAA and References

TLAA Description	Resolution Option	Section	Reference
Reactor Vessel Neutron Embrittlement Analyses		3.1	
Reactor vessel fluence	Not a TLAA. Fluence is projected to the end of the period of extended operation	3.1.1	6.5.11 6.5.44
Pressure-Temperature limits	Analysis remains valid 10CRR54.21(c)(1)(i).	3.1.2	6.5.61
Charpy Upper Shelf Energy	Analysis projected per 10CFR54.21(c)(1)(ii)	3.1.3	6.5.3, 6.5.13, 6.5.14
Adjusted Reference Temperature	Analysis projected per 10CFR54.21(c)(1)(ii)	3.1.4	6.5.3
Reactor vessel circumferential welds	Analysis projected per 10CFR54.21(c)(1)(ii)	3.1.5	6.4.11
Reactor vessel axial welds	Analysis projected per 10CFR54.21(c)(1)(ii)	3.1.6	6.4.11
Surveillance Specimen testing	Not a TLAA	3.1.7	6.5.18
Metal fatigue		NA	LRPD-04
Class 1 fatigue	Analysis (CUFs) remain valid for the period of extended operation per 10CFR54.21(c)(1)(i) OR Aging effect managed per 10CFR54.21(c)(1)(iii)	NA	LRPD-04, Section 2.1.3
Non-Class 1 fatigue	Analyses remains valid through the period of extended operation per 10CFR54.21(c)(1)(i).	NA	LRPD-04, Section 2.2.1
Non-Class 1 Pressure Vessels, Heat exchangers,	Analyses are not TLAA	NA	LRPD-04, Section 2.2.2
Jet Pump Risers	Analysis is not a TLAA.	NA	LRPD-04 Section 2.4.1
Core Spray Piping in the reactor vessel	Analysis is not a TLAA.	NA	LRPD-04 Section 2.4.2
Reactor Vessel Plate 1-15 flaw	Analysis remains valid through the period of extended operation per 10CFR54.21(c)(1)(i).	NA	LRPD-04 Section 2.4.3
Core spray nozzle to safe end weld overlay	Analysis is not a TLAA.	NA	LRPD-04 Section 2.4.4
Containment Corrosion	Analysis is not a TLAA.	NA	LRPD-04 Section 2.4.5

TLAA Description	Resolution Option	Section	Reference
Primary Containment Localized Thinning	Analysis remains valid through the period of extended operation per 10CFR54.21(c)(1)(i).	NA	LRPD-05 Section 2.4.6
VYNPS Response to Bulletin 88-08	There is no TLAA.	NA	LRPD-04 Section 2.5.1
Effects of Reactor Water Environment on Fatigue Life	Analysis remains valid per 10CFR54.21(c)(1)(i) OR Analysis projected per 10CFR54.21(c)(1)(ii) OR Aging effect managed per 10CFR54.21(c)(1)(iii).	NA	LRPD-04 Section 2.5.2
Environmental Qualification Analyses for Electrical Components	Aging effect managed by EQ program per 10CFR54.21(c)(1)(iii)	3.3	6.2.3 6.2.5 6.5.45 6.5.46
Concrete Containment Tendon Prestress Analysis	Not applicable for BWRs	3.4	
Containment Liner Plate, Metal Containment, and Penetrations Fatigue Analyses			LRPD-04
Fatigue analysis of the torus	Analyses are projected through the period of extended operation per 10CFR54.21(c)(1)(ii).		LRPD-04, Section 2.3.1
Fatigue analysis of the SRV discharge piping	Analysis remains valid per 10CFR54.21(c)(1)(i) AND Analysis projected per 10CFR54.21(c)(1)(ii)		LRPD-04, Section 2.3.2
Fatigue analysis of other torus attached piping	Analysis is projected through the period of extended operation per 10CFR54.21(c)(1)(ii)		LRPD-04, Section 2.3.3
Metal corrosion allowance		3.6	UFSAR
Reactor vessel	Not a TLAA.	3.6.1	UFSAR 4.2.5.1
Recirculation pump casing	Not a TLAA.	3.6.2	UFSAR 4.3.4
Main steam isolation valve	Not a TLAA.	3.6.3	UFSAR 4.6.3
HPCI system	Not a TLAA.	3.6.4	UFSAR 4.7.5
HPCI & RCIC turbine casings	Not a TLAA.	3.6.5	UFSAR 6.5.2.3
HPCI & RCIC pump casings	Not a TLAA.	3.6.6	UFSAR 6.4.1
RHR heat exchanger	Not a TLAA.	3.6.7	UFSAR, page C.2-55
Inservice Inspection Program Relief requests	Not a TLAA; applicable only to current ten-year interval.	3.7	6. 5.58
TLAA in BWRVIPs			
BWRVIP-05, Reactor vessel welds	TLAA updated by BWRVIP-74.	3.8.1	BWRVIP-05

TLAA Description	Resolution Option	Section	Reference
BWRVIP-18, Core spray internals	No TLAA identified in BWRVIP-18.	3.8.2	BWRVIP-18
BWRVIP-25, Core plate	TLAA for loss of preload for core plate bolts is projected through the period of extended operation by BWRVIP-25 Appendix B per 10CFR54.21(c)(1)(ii).	3.8.3	BWRVIP-25
BWRVIP-26, Top guide	Not a TLAA	3.8.4	BWRVIP-26
BWRVIP-27, SLC/DP	No TLAA for VYNPS related to SLC/ΔP nozzle.	3.8.5	BWRVIP-27
BWRVIP-38, Shroud support	Analysis remains valid through the period of extended operation per 10CFR54.21(c)(1)(i).	3.8.6	BWRVIP-38
BWRVIP-41, Jet pump assemblies	No TLAA for VYNPS related to the jet pumps.	3.8.7	BWRVIP-41
BWRVIP-47, Lower plenum	Analysis remains valid for the period of extended operation in accordance with 10CFR54.21(c)(1)(i)	3.8.8	BWRVIP-47
BWRVIP-48, Vessel ID attachment welds	The analysis remains valid for the period of extended operation in accordance with 10CFR54.21(c)(1)(i).	3.8.9	BWRVIP-48
BWRVIP-49, Instrument penetrations	The analysis remains valid for the period of extended operation in accordance with 10CFR54.21(c)(1)(i).	3.8.10	BWRVIP-49
BWRVIP-74, Reactor pressure vessel P-T Curves Fatigue Equivalent Margins for USE Reactor Vessel Welds	 P-T curves are addressed in Section 3.1.2. Fatigue is addressed in LRPD-04. EMA for USE are addressed in Section 3.1.3. RPV welds are addressed in Sections 3.1.5 and 3.1.6.	3.8.11	BWRVIP-74
BWRVIP-76, Core shroud	The analysis remains valid for the period of extended operation in accordance with 10CFR54.21(c)(1)(i).	3.8.12	BWRVIP-76

TLAA Description	Resolution Option	Section	Reference
Other TLAA			
Crane Load cycles	Not a TLAA. Attachment 4 shows allowable cycles will not be exceeded for the period of extended operation.	3.9.1	6.2.21 6.5.42
Reflood thermal shock of the core shroud.	Analysis remains valid for the period of extended operation in accordance with 10CFR54.21(c)(1)(i).	3.9.2	UFSAR Section 3.3.5.4
Reflood thermal shock of the reactor vessel	Not a TLAA	3.9.3	UFSAR Section J.
Feedwater nozzle crack growth	Not a TLAA, not based on the life of the plant.	3.9.4	6.5.51
Upset, Emergency and Faulted Conditions	Not a TLAA, no analysis.	3.9.5	Section C2.2.2 of the UFSAR
Probability of a steam line break	Not a TLAA, no safety related decisions based on this probability.	3.9.6	Section I.4 of the UFSAR
Fire protection exemptions			
Exemptions from 10 CFR 50 Appendix R	Not a TLAA	4.1	6.5.19
Other 10 CFR exemptions			
Exemption to use code case N-640 for calculation of P-T limits	Not a TLAA, an application of updated technology	4.2.1	6.2.20
Alternative Source Term (AST)	Not a TLAA.	4.2.2	6.5.40

Attachment 2 – List of Exemptions and References

Exemption Description	In Effect?	References
Fire protection program		
SSCA 6.1	Yes – discussed in Section 4.1.1	6.5.30 (submitted) 6.2.13(approved)
SSCA 6.2	Yes – discussed in Section 4.1.2	6.5.31 (submitted) 6.2.14 (approved)
SSCA 6.3	Yes – discussed in Section 4.1.3	6.5.31 (submitted) 6.2.14 (approved)
SSCA 6.4	Yes – discussed in Section 4.1.4	6.5.31 (submitted) 6.2.14 (approved)
SSCA 6.5	Yes – discussed in Section 4.1.5	6.5.31 (submitted) 6.2.14 (approved)
SSCA 6.6	Yes – discussed in Section 4.1.6	6.5.31 (submitted) 6.2.14 (approved)
SSCA 6.7	Yes – discussed in Section 4.1.7	6.5.32 (submitted) 6.2.15 (approved)
SSCA 6.8	Yes – discussed in Section 4.1.8	6.5.33 (submitted) 6.2.16 (approved)
SSCA 6.9	Yes – discussed in Section 4.1.9	6.5.34 (submitted) 6.2.16 (approved)
SSCA 6.10	Yes – discussed in Section 4.1.10	6.5.35 (submitted) 6.2.17 (approved)
SSCA 6.11	Yes – discussed in Section 4.1.11	6.5.36 (submitted) 6.2.18 (approved)
Other exemptions		
Exemption from 10 CFR 20.1003, definition of total effective dose equivalent	Yes (Not in scope for license renewal - only Part 50 exemptions are in scope. Exemption was reviewed and it is not based on any time-limited assumptions.)	6.5.37 (submitted) 6.2.19 (approved)
Code Cases N-588 and N-640 for P/T curves	Yes – discussed in section 4.2.1	6.5.38 (submitted) 6.5.39 (withdrew N-588) 6.2.20 (approved)
Alternative source term (AST) methodology	No – awaiting approval discussed in section 4.2.2 **(OI #**Error! Reference source not found.**)**	6.5.40 (submitted) 6.5.41 (supp #11)

Attachment 3 – UFSAR TLAA Search Results

UPDATED FINAL SAFETY ANALYSIS REPORT		
FSAR Section	**FSAR Text**	**Recommended Change**
Key word is in blue.		
1.2 Definitions 36 Shutdown	Shutdown - The reactor is shutdown when the effective multiplication factor (**k**eff) is sufficiently less than 1.0 that the full withdrawal of any one control rod could not produce criticality under the most restrictive potential conditions of temperature, pressure, core age, and fission product concentration.	No change required.
1.5.2 Power Generation Design Criteria (Planned Operation) Nuclear Systems	3. The fuel cladding shall be designed to accommodate without loss of integrity the pressures generated by the fission gases released from the fuel material throughout the design life of the fuel.	No change required.
1.6.5.9 Makeup Water Treatment System	The Makeup Water Treatment System processes raw river water from the Connecticut River to maintain a supply of high quality water which may be used as a makeup for the station and reactor cycles.	No change required.
2.4.4.3 Public Use	Scale samples were taken from selected species for age-growth studies.	No change required.
2.5.2.1 Introduction	These borings show that the area is overlaid by glacial deposits from the Pleistocene Age, with an average 30 feet of glacial overburden above the local bedrock, which consists of a hard biotite gneiss.	No change required.
2.5.2.3 Regional Geology	Foliated igneous rocks of middle- and late-Devonian age underlie a large portion of the region.	No change required.

UPDATED FINAL SAFETY ANALYSIS REPORT

FSAR Section	FSAR Text	Recommended Change
2.5.2.3.2 Geological History	In all cases, the faults dip steeply, and appear to be Triassic or younger in age. . . . All minor faults in the region appear to be high-angle and Triassic or younger in age.	No change required.
3.2.3 Description	Sufficient plenum volume is provided to prevent excessive internal pressure from these fission gases or other gases liberated over the design life of the fuel.	No change required. Fuel design life isn't changing.
3.3.5.1 Evaluation Methods	The ASME Boiler and Pressure Code, Section III for Class A vessels, is used as a single guide to determine limiting stress intensities and cyclic loadings for the reactor vessel internals.	None. ASME section III is still the single guide.
3.3.5.4 Thermal Shock	The peak strain resulting in the shroud support plate is about 6.5%. This strain is higher than the 5.0% strain permitted by the ASME Code, Section III, for ten cycles. However, if the ASME Code curve is extended below ten cycles, the peak strain of 6.5% corresponds to about six allowable cycles. Figure 3.3-10 illustrates both the ASME Code curve and the basic material curves from which it was established (with the safety factor of two on strain or twenty on cycles, whichever is more conservative). The extension of the ASME Code curve represents a similar criteria to that used in the ASME Code, Section III, but applied to fewer than ten cycles of loading. For this Type 304 stainless steel material, a 10% peak strain corresponds to one allowable cycle of loading. Even a 10% strain for a single cycle loading represents a very conservative suggested limit because this has a	None. The shroud strain doesn't change. This is not a 40 year TLAA. None. Peak strain isn't changing. Conditions which lead to calculated peak strain still aren't expected to occur during plant lifetime.

UPDATED FINAL SAFETY ANALYSIS REPORT		
FSAR Section	**FSAR Text**	**Recommended Change**
	large safety margin below the point at which even minor cracking is expected to begin. Because the conditions which lead to the calculated peak strain of 6.5% are not expected to occur even once during the entire reactor lifetime, the peak strain is considered tolerable. The ASME Code, Section III, allows 220 cycles of this loading, thus no significant deformations result. The most irradiated point on the inner surface of the shroud is subjected to a total integrated neutron flux of 2.7 x 10^{20} nvt (greater than 1 MeV) by the end of plant life. The peak thermal shock stress is 155,700 psi, corresponding to a peak strain of 0.57%. The shroud material is Type 304 stainless steel, which is not significantly affected by the expected level of irradiation. The material does experience some hardening and an apparent loss in uniform elongation, but it does not experience a loss in reduction of area. Because reduction of area is the property which determines tolerable local strain, irradiation effects can be neglected.	No Change. This is generic GE material. The VY shroud fluence for 54 EFPY at 1912 MWE, based on the shroud flux values in GE-NE-0000-0014-0292-01, is still below the 2.7x10^{20} generic value in the UFSAR.

UPDATED FINAL SAFETY ANALYSIS REPORT		
FSAR Section	**FSAR Text**	**Recommended Change**
3.3.6 Inspection and Testing	A vibration analysis of reactor vessel internals was performed in the design to reduce failures due to vibration. When necessary, vibration measurements were made during startup tests to determine the vibration characteristics of the reactor vessel internals and the recirculation loops under forced recirculation flow. Vibratory responses were recorded at various recirculation flow rates using strain gages on fuel channels and control rod guide tubes, accelerometers on the shroud support plate and recirculation loops, and linear differential transducers on the upper shroud and shroud head-steam separator assembly. The vibration analyses and tests were designed to determine any potential, hydraulically-induced equipment vibrations and to check that the structures should not fail due to fatigue. The structures were analyzed for natural frequencies, mode shapes, and vibrational magnitudes that could lead to fatigue at these frequencies. With this analysis as a guide, the reactor internals were instrumented and tested to ascertain that there are no gross instabilities. The cyclic loadings were evaluated using as a guide the cyclic stress criteria of the ASME Code, Section III. These field tests were only performed on reactor vessel internals that represent a significant departure from design configurations previously tested and found to be acceptable. Field test data were correlated with the analyses to ensure validity of the analytical techniques on a continuing basis.	No change required.

Attachment 3 – VYNPS TLAA Search Results

UPDATED FINAL SAFETY ANALYSIS REPORT		
FSAR Section	**FSAR Text**	**Recommended Change**
3.4.5.2.2 Materials of Construction	3. Inconel 750 is used for the collet fingers, which must function as leaf springs when cammed open to the unlocked position. Colmonoy 6 hard facing is applied to the area contacting the index tube and unlocking cam surface of the guide cap to provide a long-wearing surface adequate for design life.	No change needed. This is an active component with no guarantee of a 40 year life.
3.4.5.3.1 CRD Hydraulic Supply and Discharge Subsystems	Although the drives can function without cooling water, the life of their seals is shortened by exposure to reactor temperatures.	No change needed, this is still true.
3.4.7.1 (CRD) Development Tests	The development drive (one prototype) testing prior to 1970 included over 5,000 scrams and approximately 100,000 latching cycles during 5,000 hours of exposure to simulated operating conditions. That usable seal lifetimes greater than 1,000 scram cycles may be expected.	No change needed, this is still true. No change needed—not based on life of the plant.
3.7.2 Power Generation Design Bases	1. The ability to achieve rated core power output throughout the design lifetime of the fuel without sustaining fuel damage.	No change needed, fuel design lifetime isn't changing.
4.2.2 Power Generation Design Bases	1. The reactor vessel design lifetime shall be 40 years.	The original reactor vessel design lifetime was ~~shall be~~ 40 years. The vessel is acceptable for 60 years of operation.

UPDATED FINAL SAFETY ANALYSIS REPORT		
FSAR Section	**FSAR Text**	**Recommended Change**
4.2.3 (RV) Safety Design Basis	2.To minimize the possibility of brittle fracture failure of the nuclear system process barrier, the following shall be required: (1) the initial ductile-brittle transition temperature of materials used in the reactor vessel shall be known by reference or established empirically; (2) expected shifts in transition temperature during design service life due to environmental conditions, such as neutron flux, shall be determined and employed in the reactor vessel design; (3) operation margins to be observed with regard to the transition temperature shall be designated for each mode of operation.	No Change.

UPDATED FINAL SAFETY ANALYSIS REPORT		
FSAR Section	**FSAR Text**	**Recommended Change**
4.2.4.1 Reactor Vessel 1st paragraph	The reactor vessel is a welded vertical cylindrical pressure vessel with hemispherical heads. The reactor vessel is designed and fabricated for a useful life of 40 years based upon the specified design and operating conditions. The vessel is designed, fabricated, inspected, tested, and stamped in accordance with the ASME Boiler and Pressure Vessel Code, Section III, its interpretations, and applicable requirements for Class A vessels as defined therein. The reactor vessel and its supports are designed in accordance with the loading criteria of Appendix C, "Structural Loading Criteria." The materials used in the design and fabrication of the reactor pressure vessel are shown in Table 4.2.1. Although little corrosion of plain carbon or low alloy steels occurs at temperatures of 500°F to 600°F, higher corrosion rates occur at temperatures around 140°F. The 0.125-inch minimum thickness stainless steel cladding provides the necessary corrosion resistance during reactor shutdown and also helps maintain water clarity during refueling operations. Exterior exposed ferritic surfaces of pressure-containing parts have a minimum corrosion allowance of 1/32-inch. All carbon and low alloy steel nozzles exposed to the reactor coolant have a corrosion allowance of 1/16-inch.	The reactor vessel is a welded vertical cylindrical pressure vessel with hemispherical heads. The reactor vessel was originally ~~is~~ designed and fabricated for a useful life of 40 years based upon the specified design and operating conditions. The vessel is acceptable for 60 years of operation. The vessel is designed, fabricated, inspected, tested, and stamped in accordance with the ASME Boiler and Pressure Vessel Code, Section III, its interpretations, and applicable requirements for Class A vessels as defined therein. The reactor vessel and its supports are designed in accordance with the loading criteria of Appendix C, "Structural Loading Criteria." The materials used in the design and fabrication of the reactor pressure vessel are shown in Table 4.2.1. No change required to this paragraph.

UPDATED FINAL SAFETY ANALYSIS REPORT		
FSAR Section	**FSAR Text**	**Recommended Change**
4.2.4.1 Reactor Vessel 5th paragraph	Another way of minimizing the NDT temperature is by reducing the integrated neutron exposure at the inner surface of the reactor vessel. The coolant annulus between the vessel and core shroud and the core location in the vessel limit the integrated neutron exposure of reactor vessel material to less than 1 x 10^{19} nvt from neutrons with energy levels greater than 1 MeV, within the 40-year design lifetime of the vessel. This is not the expected exposure, nor is it the absolute limit of safe exposure; it is an exposure value that can be demonstrated to be safe and is Practical to maintain. The estimated exposure for the 40-year life is less than 2.3 x 10^{17} nvt for neutron energies greater than 1 MeV at the vessel inner surface. (Reference 17).	Another way of minimizing the NDT temperature is by reducing the integrated neutron exposure at the inner surface of the reactor vessel. The coolant annulus between the vessel and core shroud and the core location in the vessel limit the integrated neutron exposure of reactor vessel material to less than 1 x 10^{19} nvt from neutrons with energy levels greater than 1 MeV, within the original 40-year design lifetime of the vessel. This is not the expected exposure, nor is it the absolute limit of safe exposure; it is an exposure value that can be demonstrated to be safe and is practical to maintain. The estimated exposure for the 60~~40~~-year life is less than 5.4~~2.3~~ x 10^{17} nvt for neutron energies greater than 1 MeV at the vessel inner surface.~~ (Reference 17).~~.

Attachment 3 – VYNPS TLAA Search Results

UPDATED FINAL SAFETY ANALYSIS REPORT		
FSAR Section	**FSAR Text**	**Recommended Change**
4.2.4.9 Reactor Vessel Insulation	The reactor vessel insulation has an average maximum heat transfer rate of approximately 80 Btu/hr-ft2 at the operating conditions of 550°F for the vessel and 135°F for the outside air. The maximum insulation thicknesses are 4 inches for the upper head, 3-1/2 inches for the cylindrical shell and nozzles, and 3 inches for the bottom head. The upper head insulation is designed to permit complete submersion in water during shutdown without loss of insulating material, contamination of the water, or adverse effect on the insulation efficiency of the insulation assembly after draining and drying. The lower head and cylindrical shell insulation is permanently installed for the 40-year design life of the vessel. 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.	The reactor vessel insulation has an average maximum heat transfer rate of approximately 80 Btu/hr-ft2 at the operating conditions of 550°F for the vessel and 135°F for the outside air. The maximum insulation thicknesses are 4 inches for the upper head, 3-1/2 inches for the cylindrical shell and nozzles, and 3 inches for the bottom head. The upper head insulation is designed to permit complete submersion in water during shutdown without loss of insulating material, contamination of the water, or adverse effect on the insulation efficiency of the insulation assembly after draining and drying. The lower head and cylindrical shell insulation is permanently installed. ~~for the 40-year design life of the vessel.~~ 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

UPDATED FINAL SAFETY ANALYSIS REPORT		
FSAR Section	**FSAR Text**	**Recommended Change**
4.2.5 Safety Evaluation 3rd paragraph	The reactor vessel is designed for a 40-year life and will not be exposed to more than 1 x 10^{19} nvt of neutrons with energies exceeding 1 MeV. The reactor vessel is also designed for the transients which could occur during the 40-year life.	The reactor vessel was originally ~~is~~ designed for a 40-year life and ~~will~~would not be exposed to more than 1 x 10^{19} nvt of neutrons with energies exceeding 1 MeV. The reactor vessel was ~~is~~ also designed for the transients which could occur during the 40-year life. Vessel operation up to 60 years was reviewed and the maximum fluence to the vessel inner wall is 5.39 x 10^{17}n/cm^2, still well below the original design value.
	The design transients used in the original ASME III design of the Vermont Yankee reactor vessel are specified in Section 5 and Attachment C of Reference 3. Reference 4 provides an up-to-date list of design transients, vessel cyclic limits, references to current design specifications, and stress reports for reactor components.	No change.
4.2.5 Safety Evaluation	Following fabrication of the reactor vessel the NRC and nuclear power industry established updated methods to determine initial transition temperature and neutron shift. Fracture toughness requirements (Reference 16), material drop weight, and Charpy impact test results were used to determine a reference nil-ductility temperature (RTndt) for all pressure boundary components. The guidance of Regulatory Guide 1.99, Revision 2 (Reference 5) was employed to conservatively establish adjusted RTndt (ARTndt) for the plates and welds adjacent to the core.	No change.

UPDATED FINAL SAFETY ANALYSIS REPORT		
FSAR Section	**FSAR Text**	**Recommended Change**
4.2.6 Inspection and Testing	Vermont Yankee's approach is to monitor startup/shutdown and feedwater on/off flow cycles and perform UT exams on a frequency that will assure potential crack growth is smaller in relation to ASME XI limits.	No change.
4.3.4 (RR) Desciption	Since the removal of Reactor Recirculation System valve internals requires unloading of the nuclear fuel, the valves are provided with high quality back seats and trim to facilitate stem packing renewal without draining the vessel and to provide adequate leak-tightness during normal operation. The design objective of the back seats and trim is to provide a minimum 20-year service life.	No change. Note this is a “short lived” item.
4.3.4 Description page 4.3-7	The pump casing is designed in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Class C, as far as this code can be applied. This class is used because the pump casing does not experience temperature transients as severe as those that portions of the reactor vessel and certain piping connections experience; therefore, it is not necessary to make the cyclic analysis required for Class A equipment. The design objective for the recirculation pump casing is a useful life of 40 years, accounting for corrosion, erosion, and material fatigue. The pump drive motor, impeller, wear rings, and seals are designed for as long a life as is practical.	No change. The original design objective for the recirculation pump casing was ~~is~~ a useful life of 40 years, accounting for corrosion, erosion, and material fatigue. The pump drive motor, impeller, wear rings, and seals are designed for as long a life as is practical. The pump casing was reviewed for license renewal loss of material due to corrosion, erosion, and cracking due to material fatigue are managed for the period of extended operation.

UPDATED FINAL SAFETY ANALYSIS REPORT		
FSAR Section	**FSAR Text**	**Recommended Change**
4.6.3 Description (MSIV) 24th paragraph, page 4.6-7	The isolation valve is designed to pass saturated steam at 1250 psig and 575°F with a moisture content of approximately 0.23%, oxygen content of 30 ppm, and a hydrogen content of 4 ppm. The design objective for the valve is a minimum of 40 years service at the specified operating conditions. The estimated operating cycles per year is 100 cycles during the first year and 50 cycles per year thereafter. In addition to minimum wall thickness required by applicable codes, a corrosion allowance of 0.120 inches minimum is added to provide for 40 years services.	The isolation valve is designed to pass saturated steam at 1250 psig and 575°F with a moisture content of approximately 0.23%, oxygen content of 30 ppm, and a hydrogen content of 4 ppm. The design objective for the valve was ~~is~~ a minimum of 40 years service. The estimated operating cycles per year is 100 cycles during the first year and 50 cycles per year thereafter. In addition to minimum wall thickness required by applicable codes, a corrosion allowance of 0.120 inches minimum is added to provide for 40 years services. Allowable operating cycles and corrosion allowance were reviewed and remain valid for the period of extended operation (60-years).
4.6.3 Description (MSIVs)	Design specification ambient operating conditions are 135°F normal, 150°F maximum, at 100% relative humidity, in a radiation field of 15 R/hr gamma and 25 Rad/hr neutron plus gamma, continuous for design life. The inboard valves are not exposed to these maximum conditions continuously, and the outboard valves are in much less severe ambient conditions.	No change.
4.6.4 Test Program	A full-size, 20 inch valve produced for actual use in a BWR was tested in a range of steam/water blowdown conditions simulating postulated accident conditions. The test valve was opened and closed more than 400 times (200 cycles) during the test program, . . .	No change.
4.10.3.2 Unidentified Leakage Rage	An analysis was undertaken to estimate, on the basis of information available when the plant was built from the Pipe Study and other sources, the probability of a line break occurring in a reactor piping system as a result of progressive crack growth.	No change.

<table>
<tr><th colspan="3">UPDATED FINAL SAFETY ANALYSIS REPORT</th></tr>
<tr><th>FSAR Section</th><th>FSAR Text</th><th>Recommended Change</th></tr>
<tr><td rowspan="2">4.10.3.2 Unidentified Leakage Rate</td><td colspan="2">As Found:
5. For particular reactor piping system, the leak detection capability is approximately known and, so, the largest leak which might escape detection can be estimated. The probability of a line break in the system is then:

Leaks / Component- Year X No. of components
X Design life
X Probability of break for given leak rate

Results of this study (see Appendix "I", Figure I-3) demonstrate the importance of adequate leak detection capability in maintaining a high piping system reliability. A relatively higher risk of a steam line break, as compared to a waterline break, is also indicated, as a result of the lower leak rate and, hence, more difficult detection of a steam line crack of a given length.</td></tr>
<tr><td colspan="2">Suggested Change: No change is required to this description. The actual calculation in Appendix I is changed below.</td></tr>
<tr><td>4.10.3.3 Total Leakage Rate</td><td>A flow recorder continually plots time versus discharge flow rate from each sump; an increase in leakage rate is detectable by an increase in sump discharge flow time and an increased frequency in discharge flow cycles.</td><td>No change.</td></tr>
<tr><td>5.2.3.7 Primary Containment Normal Heating, Ventilation, and Air

Conditioning Systems</td><td>Maintaining the drywell ambient temperature in the range of 135°F to 165°F except for the upper drywell regions during normal plant operation assures that the insulation on motors, isolation valves, operators and sensors, instrument cable, electrical cable and gasket materials or sealants used at the penetrations have a sustained life without deterioration.</td><td>No change.</td></tr>
</table>

UPDATED FINAL SAFETY ANALYSIS REPORT

FSAR Section	FSAR Text	Recommended Change
6.4.1 High Pressure Coolant Injection System 14th paragraph, page 6.4-5	The system is designed for a service life of 40 years, accounting for corrosion, erosion, and material fatigue.	The system was ~~is~~ designed for an original service life of 40 years, accounting for corrosion, erosion, and material fatigue. The HPCI system was reviewed for license renewal, and corrosion, erosion, and material fatigue were evaluated for 60 years.
6.5.2.1 LOCA Analysis Methods and Results	LOCA analysis methods developed by General Electric Company (Reference 2) which conform to 10CFR50.46 requirements were used to demonstrate 10CFR50.46 conformance for the Vermont Yankee Nuclear Power Station (VYNPS) for the first 16 cycles of plant operation.	No change.
6.6 Inspection and Testing	The portions of the Core Standby Cooling Systems requiring pressure integrity are designed to specifications for inservice inspection to detect defects which might affect the cooling performance. The reactor vessel nozzles and the core spray and feedwater spargers receive particular attention. Records are kept of the number of design basis thermal cycles these components receive.	No change.
7.2.3.10 Wiring	Wiring and cables for Reactor Protection System instrumentation are selected to avoid excessive deterioration due to temperature and humidity during the design life of the plant.	No change.

UPDATED FINAL SAFETY ANALYSIS REPORT		
FSAR Section	**FSAR Text**	**Recommended Change**
7.3.4.9 Environmental Capabilities	Verification that the isolation equipment has been designed, built, and installed in conformance to the specified criteria is accomplished through quality control and performance tests in the vendor's shop or after installation at the plant before startup, during startup, and thereafter during the service life of the equipment.	No change.
7.5.6.2.3 Physical Arrangement	Each LPRM detector assembly contains four miniature fission chambers with an associated solid sheath cable. Each fission chamber produces a current which when coupled with the LPRM signal-conditioning equipment, provides the desired scale deflection throughout the design lifetime of the chamber. Each individual chamber of the assembly is a moisture-proof, pressure-sealed unit. Each assembly also contains a calibration tube for a Traversing Incore Probe (TIP). The enclosing tube around the entire assembly contains holes along its length. These holes allow circulation of the reactor coolant water to cool the fission chambers. Numerous tests have been performed on the chamber assemblies including tests of linearity, lifetime, gamma sensitivity, and cable effects.(1) These tests and experience in operating reactors, including Vermont Yankee, provide confidence in the ability of the LPRM Subsystem to monitor neutron flux to the design accuracy throughout the design lifetime.	No change. NOTE: AMRM-32, Section 2.0, states "The LPRM have limited lifetimes and are replaced as determined by OP-4407 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."

UPDATED FINAL SAFETY ANALYSIS REPORT		
FSAR Section	**FSAR Text**	**Recommended Change**
7.5.9.3 Power Generation Evaluation	An adequate number of TIP machines is supplied to assure that each instrument location assembly can be probed by a TIP and the central one can be probed by every TIP to allow intercalibration. The system has been field tested in an operating reactor to assure reproducibility for repetitive measurements, and the mechanical equipment has undergone life testing under simulated operating conditions to assure that all specifications can be met.	No change.
9.2.4.2 Low Purity Wastes	For the purpose of analyzing future radiological impacts during the plant's life, it is assumed that 1% of the combined processed stream treated each year would be discharged from the station.	No change.
10.13.1 Power Generation Objectives	The power generation objective of the Station Makeup Water System is to maintain a supply of treated water that may be used as makeup for the station and reactor cycles.	No change.
14.5.4.1 Pressure Regulator Failure	An analysis of the impact of thermal stress from this event on the RPV fatigue life has been made (NEDO-22243-1, "Safety Evaluation of MSIV Low Turbine Inlet Pressure Isolation Setpoint Change for Vermont Yankee Nuclear Power Station," May 1983). The analysis concluded that the additional usage factor associated with the transient is insignificant.	No change.
14.5.6.1 Recirculation Pump Seizure	For Cycle 22 and future cycles, Vermont Yankee is expected to have rated power Operating Limit MCPRs at least 0.20 higher than the Safety Limit MCPR.	No change.

UPDATED FINAL SAFETY ANALYSIS REPORT		
FSAR Section	**FSAR Text**	**Recommended Change**
A.9.2.1 General	Pipe support design methodology has been consistent throughout the life of the plant and is described in Section 12.2 and Section C.3.9 of Appendix C of this FSAR.	No change.
B.2.2.4 Quality Assurance and Inspection of the Reactor Primary System	"The Committee continues to emphasize the importance of quality assurance in fabrication of the primary system and of inspection during service life.	No change.
B.2.2.4 Quality Assurance and Inspection of the Reactor Primary System	Provisions are being made to the maximum extent considered feasible for inspection of primary system components during service life, consistent with the requirements of "Draft ASME Code for Inservice Inspection of Nuclear Reactor Coolant Systems."	No change.
B.2.3.2 Addition of Cooling Tower Complex	"At the time of the previous review by the Committee, the applicant planned to use the Connecticut River as a heat sink by drawing cooling water for the main condenser from the river, heating it in the condenser, and returning the heated water to the river. Since that time, limitation by state agencies on the allowable temperature rise and maximum temperature of water returned to the river has led the applicant to propose the use of cooling towers to reject a portion of the waste heat from the plant to the atmosphere."	No change. Does not involve a TLAA.
B 4.10 Resolution	Where deflection is not the limiting factor, the ASME Boiler and Pressure Vessel Code, Section III, was used as a guide to determine limiting stress intensities and cyclic loadings for the core internal structure.	No change.

UPDATED FINAL SAFETY ANALYSIS REPORT		
FSAR Section	**FSAR Text**	**Recommended Change**
B.5.3 Fuel Orientation	Experience has shown that the distinguishing features will be visible during the design lifetime of the fuel. In all cases, fueling procedures require that the fuel assembly number be verified.	No change. Fuel lifetime isn't changing.
C.2.2 Loading Conditions and Allowable Limits	Certain of the limits described in these criteria, i.e., deformation limit and fatigue limit, are included for completeness, but do not necessarily require application to all components. Where it is clear that fatigue or excess deformation are not of concern for a particular structure or component, a formal analysis with respect to that limit is not required.	No change. NOTE: Appendix C is identified as "historical" and not maintained current in Section C.1.1. All changes for Appendix C may be ignored and the "historical" record maintained.

UPDATED FINAL SAFETY ANALYSIS REPORT		
FSAR Section	**FSAR Text**	**Recommended Change**
C.2.2.2 Allowable Limits (definition of allowable limits)	Current FSAR Text: Generic Definition (Current FSAR) P_{40} = 40 year event encounter probability Upset (likely) $1.0 > P_{40} > 10^{-1}$ Emergency (low probability) $10^{-1} > P_{40} > 10^{-3}$ Faulted (extremely low probability) $10^{-3} > P_{40} > 10^{-6}$ (sixth paragraph) SF_{min} is related to the event probability by the following equation: $SF_{min} = \frac{9}{3 - \log_{10} P_{40}}$ (Equation A) where: $10^{-1} > P_{40} > 10^{-5}$ (Equation A applies) $10^{-1} > P_{40} > 10^{-6}$ (SF_{min} = 1.125) $1.0 > P_{40} > 10^{-1}$ (SF_{min} = 2.25) (Page C.2-6 of 65, first paragraph) These expressions show the probabilistic significance of the classical safety factor concept as applied to reactor safety. The SF_{min} values corresponding to the current governing accident event probabilities are summarized as follows: Item Governing Loading Conditions P_{40} SF_{min} (Last paragraph of section C.2.2) The minimum safety factor decreases as the event probability diminishes and if the event is too improbable (incredible: $P_{40} < 10^{-6}$) then no safety factor is appropriate or required.	

<table>
<tr><th colspan="3">UPDATED FINAL SAFETY ANALYSIS REPORT</th></tr>
<tr><th>FSAR Section</th><th>FSAR Text</th><th>Recommended Change</th></tr>
<tr><td></td><td colspan="2">
Recommended Changes:

Generic Definition (FSAR Changes)
$P_{60\cancel{40}}$ = 60~~40~~ year event encounter probability
Upset (likely) $1.0 > P_{60\cancel{40}} > 10^{-1}$
Emergency (low probability) $10^{-1} > P_{60\cancel{40}} > 10^{-3}$
Faulted (extremely low probability) $10^{-3} > P_{60\cancel{40}} > 10^{-6}$
(sixth paragraph)
SF_{min} is related to the event probability by the following equation:
$$SF_{min} = \frac{9}{3 - \log_{10} P_{60\cancel{40}}} \qquad \text{(Equation A)}$$
where:
$10^{-1} > P_{60\cancel{40}} > 10^{-5}$ (Equation A applies)
$10^{-1} > P_{60\cancel{40}} > 10^{-6}$ (SF_{min} = 1.125)
$1.0 > P_{60\cancel{40}} > 10^{-1}$ (SF_{min} = 2.25)

(Page C.2-6 of 65, first paragraph)
These expressions show the probabilistic significance of the classical safety factor concept as applied to reactor safety. The SF_{min} values corresponding to the current governing accident event probabilities are summarized as follows:

Item Governing Loading Conditions $P_{60\cancel{40}}$ SF_{min}

(Last paragraph of section C.2.2)
The minimum safety factor decreases as the event probability diminishes and if the event is too improbable (incredible: $P_{60\cancel{40}} < 10^{-6}$) then no safety factor is appropriate or required.
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UPDATED FINAL SAFETY ANALYSIS REPORT		
FSAR Section	**FSAR Text**	**Recommended Change**
C.2.4.1 Criteria	Stress analysis requirements and load combinations for the reactor vessel have been evaluated for the cyclic conditions expected throughout the 40 year life, with the conclusion that ASME code limits are satisfied. The vessel design report contains the results of the detailed design stress analyses performed for the reactor vessel to meet the code requirements. Selected components, considered to possibly have higher than code design primary stresses as a result of rare events or a combination of rare events, have been analyzed in accordance with the requirements of the loading criteria in this appendix. Results of the most critical of those analyses are included in a following section. The conclusion is that the limits in the criteria have been met.	Stress analysis requirements and load combinations for the reactor vessel were originally ~~have been~~ evaluated for the cyclic conditions expected throughout ~~the~~ a 40 year life, with the conclusion that ASME code limits are satisfied. The vessel is acceptable for 60 years of operation. The vessel design report contains the results of the detailed design stress analyses performed for the reactor vessel to meet the code requirements. Selected components, considered to possibly have higher than code design primary stresses as a result of rare events or a combination of rare events, have been analyzed in accordance with the requirements of the loading criteria in this appendix. Results of the most critical of those analyses are included in a following section. The conclusion is that the limits in the criteria have been met.
C.2.4.2 Vessel Fatigue Analysis	An analysis of the reactor vessel shows that all components are adequate for cyclic operation by the rules of Section III of the ASME Code. Exhibit 4 of the Reactor Pressure Vessel Design Report gives a summary of the analysis. The analysis indicates that for the more critical components on the vessel that the primary plus secondary stress intensity range is less than 3 SM and that a plastic analysis is not required. Also, the usage factors for the conservatively specified operating cycles is substantially less than the code allowed 1.0.	No change.
C.2.5.3 Fatigue Analysis	A fatigue analysis was performed using as a guide the	No change. Appendix C is identified as “historical” in

UPDATED FINAL SAFETY ANALYSIS REPORT		
FSAR Section	**FSAR Text**	**Recommended Change**
	ASME Boiler and Pressure Vessel Code, Section III. The method of analysis used to determine the cumulative fatigue usage as described in APED 5460, "Design and Performance of GE-BWR Jet Pumps," September 1968. The most significant fatigue loading occurs in the jet pump - shroud - shroud support area of the internals. The analysis was performed for the Dresden plant, where the configuration (stilt type shroud support) was similar to the Vermont Yankee plant. Therefore, the calculated fatigue usage is expected to be a good approximation for this plant. Loading Combinations and Transients Considered 1. Normal Startup and Shutdown Vessel fluid temperature goes from 70°F to 545°F at 100°F/hr rate. 120 cycles. 2. Operating Basis and Design Basis Earthquake Considered using a conservative axi-symmetric load. Resulting stress negligible. 3. Ten-Minute Blowdown Vessel fluid initially at 545°F. Stuck-open relief valve causes fluid temperature to drop to 370°F in ten minutes. 1 cycle. 4. HPCI Operation Produced by loss of feedwater pumps. Fluid	Section C.1.1; it should be left as historical and not updated.

UPDATED FINAL SAFETY ANALYSIS REPORT		
FSAR Section	**FSAR Text**	**Recommended Change**
	temperature between vessel and shroud drops to 300°F in about five minutes. Vessel lower plenum may reach 100°F. 30 cycles. 5. LPCI Operation (DBA) Vessel flooded with 140°F cooling water subsequent to complete vessel blowdown in 30 seconds. 1 cycle. 6. Improper Start of Recirculation Loop 130°F water flows in reverse through one recirculation outlet nozzle for 55 seconds. Bulk water temperature between vessel and shroud steps from 545°F to 480°F and returns to 545°F. 1 cycle. Cumulative Fatigue Usage Umax ≈ 0.33 (Uallowable = 1.0) Remarks The location of maximum fatigue usage is on the bottom side of the baffle plate at the point where the baffle plate attaches to the shroud in the vicinity of the minimum ligament.	
Recirculation Loop Piping, pages C.2-32 and C.2-33	<u>Statement of Criteria</u> B. For load combinations that have a very low probability of occurrence, maintain primary stresses below the following limit:	<u>Statement of Criteria</u> B. For load combinations that have a very low probability of occurrence, maintain primary stresses below the following limit:

<table>
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<tr><th>FSAR Section</th><th>FSAR Text</th><th>Recommended Change</th></tr>
<tr><td></td><td>
$\frac{225}{SF}$ times B31.1.0 allowable stresses, where

$SF = \frac{9}{3 - \log_{10} P_{40}}$

and

P_{40} = Probability of load combination occurrence in 40-year plant life.

Method of Analysis

B. Effects from the following loading combinations determined in accordance with rules of B31.1.0:

1. The sum of the longitudinal stresses due to pressure, dead weight, and inertia effects of maximum hypothetical earthquake must be less than 1.8 times the hot allowable stress. The probability of this load occurrence during the 40-year plant life is 10^{-3} and SF = 1.5.

2. The sum of the longitudinal stresses due to maximum pressure, dead weight and inertia effects of design basis earthquake must be less than 1.5 times the hot allowable stress. The probability of this load occurring during the 40-year plant life is 10^{-2} and SF = 1.8.

3. The sum of the longitudinal stresses due to maximum pressure, dead weight and inertia effects of maximum hypothetical earthquake must be less
</td><td>
$\frac{225}{SF}$ times B31.1.0 allowable stresses, where

$SF = \frac{9}{3 - \log_{10} P_{60\cancel{40}}}$

and

$P_{60\cancel{40}}$ = Probability of load combination occurrence in 60~~40~~-year plant life.

Method of Analysis

B. Effects from the following loading combinations determined in accordance with rules of B31.1.0:

1. The sum of the longitudinal stresses due to pressure, dead weight, and inertia effects of maximum hypothetical earthquake must be less than 1.8 times the hot allowable stress. The probability of this load occurrence during the 60~~40~~-year plant life is 10^{-3} and SF = 1.5.

2. The sum of the longitudinal stresses due to maximum pressure, dead weight and inertia effects of design basis earthquake must be less than 1.5 times the hot allowable stress. The probability of this load occurring during the 60~~40~~-year plant life is 10^{-2} and SF = 1.8.

3. The sum of the longitudinal stresses due to maximum pressure, dead weight and inertia effects of maximum hypothetical earthquake must
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<tr><th colspan="3">UPDATED FINAL SAFETY ANALYSIS REPORT</th></tr>
<tr><th>FSAR Section</th><th>FSAR Text</th><th>Recommended Change</th></tr>
<tr><td></td><td>than 2.0 times the hot allowable stress. The probability of this load combination occurring during the 40 year plant life is .25 x 10^{-3} and SF = 1.36.</td><td>be less than 2.0 times the hot allowable stress. The probability of this load combination occurring during the 60~~40~~ year plant life is .25 x 10^{-3} and SF = 1.36.</td></tr>
<tr><td>Main Steam Piping
pages C.2-35 and C.2-36</td><td>Statement of Criteria

B. For load combinations that have a very low probability of occurrence, maintain primary stresses below the following limit:

$\frac{225}{SF}$ times B31.1.0 allowable stresses, where

$SF = \frac{9}{3 - \log_{10} P_{40}}$

and

P_{40} = Probability of load combination occurrence in 40-year plant life.

Method of Analysis

B. Effects from the following loading combinations determined in accordance with rules of B31.1.0:

1. The sum of the longitudinal stresses due to pressure, dead weight, and inertia effects of maximum hypothetical earthquake must be less than 1.8 times the hot allowable stress. The probability of this load occurrence during the 40-year plant life is 10^{-3} and SF = 1.5.</td><td>Statement of Criteria

B. For load combinations that have a very low probability of occurrence, maintain primary stresses below the following limit:

$\frac{225}{SF}$ times B31.1.0 allowable stresses, where

$SF = \frac{9}{3 - \log_{10} P_{60\cancel{40}}}$

and

$P_{60\cancel{40}}$ = Probability of load combination occurrence in 60~~40~~-year plant life.

Method of Analysis

B. Effects from the following loading combinations determined in accordance with rules of B31.1.0:

1. The sum of the longitudinal stresses due to pressure, dead weight, and inertia effects of maximum hypothetical earthquake must be less than 1.8 times the hot allowable stress. The probability of this load occurrence during the 60~~40~~-year plant life is 10^{-3} and SF = 1.5.</td></tr>
</table>

UPDATED FINAL SAFETY ANALYSIS REPORT		
FSAR Section	**FSAR Text**	**Recommended Change**
	2. The sum of the longitudinal stresses due to maximum pressure, dead weight and inertia effects of design basis earthquake must be less than 1.5 times the hot allowable stress. The probability of this load occurring during the 40-year plant life is 10^{-2} and SF = 1.8. 3. The sum of the longitudinal stresses due to maximum pressure, dead weight and inertia effects of maximum hypothetical earthquake must be less than 2.0 times the hot allowable stress. The probability of this load combination occurring during the 40 year plant life is .25 x 10^{-3} and SF = 1.36.	2. The sum of the longitudinal stresses due to maximum pressure, dead weight and inertia effects of design basis earthquake must be less than 1.5 times the hot allowable stress. The probability of this load occurring during the 60~~40-~~year plant life is 10^{-2} and SF = 1.8. 3. The sum of the longitudinal stresses due to maximum pressure, dead weight and inertia effects of maximum hypothetical earthquake must be less than 2.0 times the hot allowable stress. The probability of this load combination occurring during the 60~~40~~ year plant life is .25 x 10^{-3} and SF = 1.36.
RCIC and HPCI pump casings page C.2-49	2. The minimum wall thickness of the pump shall be based on that to limit stress to the allowable working stress when subjected to design pressure plus corrosion allowance.	No change.
RCIC and HPCI turbines page C.2-52	2. The minimum wall thickness of the turbine casing shall be based on that to limit stress to the allowable working stress when subjected to design pressure plus corrosion allowance.	No change.
Table C.2.2 Primary Stress Limit	The methods of linear elastic stress analysis may be used in the fracture analysis where its use is clearly conservative or supported by experimental evidence. Examples where "fracture mechanics" may be applied are for fillet welds or end of fatigue life crack propagation.	No change.

Attachment 3 – VYNPS TLAA Search Results

<table>
<tr><th colspan="3">UPDATED FINAL SAFETY ANALYSIS REPORT</th></tr>
<tr><th>FSAR Section</th><th>FSAR Text</th><th>Recommended Change</th></tr>
<tr><td rowspan="2">TABLE C.2.4

Fatigue Limit</td><td colspan="2">General Limit (2)

Summation of mean fatigue(1) damage usage including emergency or fault events with design and operation loads following Miner hypotheses ... either one (not both)
a. Fatigue cycle usage from analysis — <.05
b. Fatigue cycle usage from test — <0.33

(1) Fatigue failure is defined here as a 25% area reduction for a load carrying member which is required to function or excess leakage causing loss of function, whichever is more limiting. In the fatigue evaluation, the methods of linear elastic stress analysis may be used when the 3Sm range limit of ASME III has been met. If 3Sm is not met, account will be taken of (a) increases in local strain concentration, (b) strain ratcheting, (c) redistribution of strain due to elastic-plastic effects. The January, 1969 draft of the USAS B31.7 Piping Code may be used. With elastic-plastic methods, strain hardening may be used not to exceed in stress for the same strain, the steady state cyclic strain hardening measured in a smooth low cycle fatigue specimen at the

average temperature of interest.

(2) It is acceptable to use the ASME Section III Design Fatigue Curves in conjunction with a cumulative usage factor of 1.0 (using Miner's hypothesis) in lieu of using the mean fatigue data curves with a limit on fatigue usage of 0.05, since the two methods are approximately equivalent.</td></tr>
<tr><td colspan="2">Recommended Change: No change. This is all theory, and the theory isn't changing.</td></tr>
<tr><td>F.1 Summary Description</td><td>Vermont Yankee has made changes to the facility over the life of the plant that may have invoked the final General Design Criteria as design criteria.</td><td>No change.</td></tr>
<tr><td>F.2.1 Group I Overall Plant Requirements</td><td>Complete records of the as-built design of the station, changes during operation and quality assurance records will be maintained throughout the life of the station.</td><td>No change.</td></tr>
</table>

UPDATED FINAL SAFETY ANALYSIS REPORT		
FSAR Section	**FSAR Text**	**Recommended Change**
H.1 Summary Description	It is the purpose of Section H.2 to review initial reactor core design criteria and, by presentation of analytical data, show the existence of adequate thermal margins. The thermal operating limits for the Vermont Yankee Nuclear Power Station are evaluated for each cycle of operation. These are presented in the current cycles Core Performance Analysis Report and in the Technical Specifications.	No change.
H3.3 Recirculation Flow	However, this accuracy was only predictable at the beginning of life, in the recently calibrated condition, and with some subjective engineering estimates introduced into some of the component uncertainty contributors.	No change.
I.1 Probability of Leaking Failures	One of the objectives of the Pipe Rupture Study is to predict, based on our knowledge of stress levels and crack propagation rates in a BWR piping system, the rate of occurrence of through-wall cracks during the life of the plant for each piping system and component.	No change.

UPDATED FINAL SAFETY ANALYSIS REPORT		
FSAR Section	**FSAR Text**	**Recommended Change**
I.2 Critical Crack Size	Fracture data for a wider range of materials and temperatures may be obtained from the terminal fracture behavior of wide-plate crack growth test specimens, as recorded by Brothers. In another task under the pipe study, E. Kiss9 has conducted reversed-bending fatigue tests of 6-inch pipe at room temperature, with internal pressure in some cases. Stress intensity factors as high as 111 ksi inch have been recorded for circumferential through-wall cracks, with no crack instability occurring. The data for elevated temperature falls in approximately the same range of K_c as the room temperature data. These points represent terminal fracture following several thousand cycles of plastic strain.	No change.

UPDATED FINAL SAFETY ANALYSIS REPORT		
FSAR Section	**FSAR Text**	**Recommended Change**
I.4 PROBABILITY OF LINE BREAK	A BWR typically has about 250 piping components of size 4 inches or larger which are located between the vessel and the first shutoff valve. Of this total, 100 components are associated with steam and 150 with water. If a detectable leak rate is 5 gpm, then (from Figure I-3) a crack in a steamline has a 3.7×10^{-4} probability leading to line break, and a crack in a waterline has 6×10^{-6} probability of line break. The probability of a steam line break in a 40-year plant design life is: $5.3 \times 10^{-4} \frac{\text{Leaks}}{\text{Component-Year}} \times$ X 100 components X 40 years X 3.7×10^{-4} = 7.8×10^{-4} This is equivalent to a system reliability of 0.9992. For waterlines, the probability of a break is: $5.3 \times 10^{-4} \times 150 \times 40 \times 6 \times 10^{-6} = 1.9 \times 10^{-5}$, or a reliability of 0.99998.	

UPDATED FINAL SAFETY ANALYSIS REPORT		
FSAR Section	**FSAR Text**	**Recommended Change**
	Recommended Changes: A BWR typically has about 250 piping components of size 4 inches or larger which are located between the vessel and the first shutoff valve. Of this total, 100 components are associated with steam and 150 with water. If a detectable leak rate is 5 gpm, then (from Figure I-3) a crack in a steamline has a 3.7×10^{-4} probability leading to line break, and a crack in a waterline has 6×10^{-6} probability of line break. The probability of a steam line break in a 60~~40~~-year plant design life is: $5.3 \times 10^{-4} \dfrac{\text{Leaks}}{\text{Component-Year}}$ X X 100 components X 60~~40~~ years X 3.7×10^{-4} = $1.2\cancel{7.8} \times 10^{-3\cancel{4}}$ This is equivalent to a system reliability of 0.9988~~92~~. For waterlines, the probability of a break is: $5.3 \times 10^{-4} \times 150 \times 60\cancel{40} \times 6 \times 10^{-6} = 2\cancel{1}.9 \times 10^{-5}$, or a reliability of 0.99997~~8~~.	
K.1.2.2 Structural and Design Evaluations	Although the repair is not considered an ASME Boiler & Pressure Vessel (B&PV) Code repair, the repair satisfies the Design By Analysis stress and fatigue criteria of the ASME Boiler & Pressure Vessel Code, Section III, Subsection NG (Reference 5).	No change.
K.3.1 Design Objectives	The function of the core shroud repair is to structurally replace all potentially sensitized 304 stainless steel circumferential core shroud welds, i.e., H1 through H7 (See Figure K.1-1). In addition, the repair can accommodate a complete failure of the H8 shroud weld with the shroud support legs intact. The design life of the repair is 40 years.	No change required, the 40 year life of the repair is greater than the period of extended operation. Repair was made in 1995.

UPDATED FINAL SAFETY ANALYSIS REPORT		
FSAR Section	**FSAR Text**	**Recommended Change**
K.4.3.1 Repair Hardware Structural Evaluation	• The design by analysis stress and fatigue criteria of the ASME Boiler & Pressure Vessel Code, Section III, Subsection NG, are satisfied. • The maximum fatigue usage in the tie rod assembly due to OBE and thermal expansion (including open and shutdown) loads occurs in the threaded section of the spring rods. The fatigue usage at this location is less than 12%. • The fatigue usage from shroud and flow induced vibration is negligible.	No change.
K.4.3.2 Flow Induced Vibration	As discussed above, the evaluations show that stresses resulting from flow-induced vibration are small and pose no fatigue concern.	No change.

Technical Specifications		
Tech Spec Section	**Tech Spec Text**	**Recommended Change**
Key word is in blue.		
B 3.6.A Pressure Temperature Limits	All components in the Reactor Coolant System are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in Section 4.2 of the FSAR. During startup and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.	OK as is.
B 3.6.A Pressure Temperature Limits	The guidance of Branch Technical Position – MTEB 5-2, material drop weight, and Charpy impact test results were used to determine a reference nil-ductility temperature (RT_{NDT}) for all pressure boundary components. For the plates and welds adjacent to the core, fast neutron (E > 1 Mev) irradiation will cause an increase in the RT_{NDT}. For these plates and welds an adjusted RT_{NDT} (ART_{NDT}) of 89°F and 73°F (¼ and ¾ thickness locations) was conservatively used in development of these curves for core region components. Based upon plate and weld chemistry, initial RT_{NDT} values, predicted peak fluence ($2.3x10^{17}$ n/cm2) for a gross power generation of $4.46x10^{8}$ MWH(t) (Battelle Columbus Laboratory Report BCL 585-84-3, dated May 15, 1984) these core region ART_{NDT} values conservatively bound the guidance of Regulatory Guide 1.99, Revision 2.	No change till revised P/T curves are developed and submitted.

Technical Specifications		
Tech Spec Section	**Tech Spec Text**	**Recommended Change**
	Due to convection cooling, stratification, and cool CRD flow, the bottom head area is subject to lower temperatures than the balance of the pressure vessel. The RT_{NDT} of the lower head is lower than the ART_{NDT} used for the beltline. The lower head area is also not subject to the same high level of stress as the flange and feedwater nozzle regions. The dashed Bottom Head Curve is less restrictive than the enveloping curve used for the upper regions of the vessel and provides Operator's with a conservative, but less restrictive P/T limit for the cooler bottom head region. The actual shift in RT_{NDT} of the critical plate and weld material in the core region will be established periodically during operation by removing and evaluating, in accordance with ASTM E185, reactor vessel material irradiation surveillance specimens installed near the inside wall of the reactor vessel in the core area.	No change required for the current P/T curves, re-evaluate when curves are developed for the period of extended operation. OK as is.
3.6. E. Structural Integrity and Operability Testing	The structural integrity and the operability of the safety-related systems and components shall be maintained at the level required by the original acceptance standards throughout the life of the plant.	No change.
B 3.6. E. Structural Integrity and Operability Testing	Prior to operation, the reactor primary system was free of gross defects. In addition, the facility has been designed such that gross defects should not occur throughout plant life.	No change.

Attachment 3 – VYNPS TLAA Search Results

Technical Specifications		
Tech Spec Section	**Tech Spec Text**	**Recommended Change**
B 4.7.A Primary Containment System	Experience with this type of coating during plant operating cycles between 1972 and the present indicates that this inspection methodology and interval are adequate. Since valve internals are designed for a 40-year lifetime, an inspection program which cycles through all valves in one-eighth of the design lifetime is extremely conservative.	No change NOTE: Valve internals will still be designed for 40 years. Valve bodies will be in aging management programs to assure acceptability for 60 years. Therefore this doesn't change.
B 4.7.D Primary Contianment Isolation Valves	The test closure time limit of five seconds for these main steam isolation valves provides sufficient margin to assure that cladding perforations are avoided and 10CFR100 limits are not exceeded.	No change.

VYNPS Correspondence		
Correspondence	**Content**	**TLAA**
BVY 98-138	Implementation of ASME Code Case N560 Risk Informed Inspection of Class 1 piping.	NONE No TLAA in this memo. Follow up to verify implementation. If this includes small bore piping, a small bore piping program isn't needed.
ERC 2000-029	Revision 3 to RPV Pressure/Temperature Limits	YES. RPV Pressure/Temperature limits are addressed in this document. Review this reference for possible input to RPV Pressure/Temperature Limits section

Attachment 3 – VYNPS TLAA Search Results

ERC 2001-019	Fatigue Pro study	NONE. Letter is to a vendor about study being conducted.
ERC 2002-035	Fatigue Pro study, Phase 2	NONE. Letter is to a vendor about study being conducted.
ERC-2004-2005	Update HELB for MELLA.	NONE. (Hit because it was signed by B. Slifer.
DM 21	Design Memo #21: CRD Thermal Cycle design notes.	YES. Cycle management is covered in the Metal Fatigue TLAA. Review this reference for input to material fatigue TLAA.
INF 92-007	FAC of Feedwater Piping	NONE Information Notice to PWRs only.
INF 92-035	Higher than predicted erosion/corrosion of RCPB in containment at BWRs.	NONE Information only, no response required.
INF 93-020	Thermal Fatigue Cracking of Feedwater Piping to Steam Generators	NONE Information only, PWRs only.
INF 93-021	Erosion/Corrosion Program Generic Comments	NONE Information only, no response required.
INF 98-045	Cavitation Erosion of letdown line orifices resulting in cracking of pipe welds	NONE Information only.
SIL 0243	Mitigation of SCC is Austenitic Stainless Steel small bore piping in BWRs.	NONE Information only.
SIL 0318	BWR Reactor Vessel Cyclic Duty Monitoring	YES Cyclic duty monitoring is covered in the Metal Fatigue TLAA. Review this reference for input to metal fatigue TLAA.

SIL 0409 Rev 2	Incore Dry Tube Cracks have been found.	NONE. Cracks have not been found and no pressure boundary leakage has resulted. No inspection of tubes is required till they reach their 20 year life (not 40 year life). Dry tubes are subject to aging management review in AMRM-32.
SIL 0426	Automatic Depressurization System Cycling	NONE Applicable to BWR-5/6 only.
SIL 0638-01	Cracking of control blades.	NONE. Control blades are short lived.

VYNPS QA Program (QAPM)		
QAPM Section	**QAPM text**	**Recommended change**
Table 1 K.5 ANSI N45.2.5 Section 4.9	The words "splicing crew" are interpreted to refer to all project members that are actively engaged in preparing and assembling cadweld mechanical splices at the final splice location. Separate test cycles will be established for each bar size and each splice position.	None – Not a TLAA
Table 1, N.11 ANSI N45.2.12 Section 4.5.1	The QAPM Section A.6 corrective action program may be used instead of these requirements as long as the appropriate time limits are applied to significant conditions adverse to quality.	None – Not a TLAA

VYNPS Fire Protection Commitment Reference Manual		
FPCRM Section	**FPCRM text**	**TLAA**

Attachment 3 – VYNPS TLAA Search Results

Page 29, (h)	Service or operating life should be a minimum of one half hour for the self-contained units.	NONE. Short lived component.
Page 69, H.	Control Room personnel may be furnished breathing air by a manifold system piped from a storage reservoir, if practical. Service or rated operating life shall be a minimum of one-half hour for the self-contained units.	NONE. Short lived component.
Page A-25, Appendix A, (h)	Service or operating life should be a minimum of one half-hour for the self-contained units.	NONE. Short lived component.
Page B-23, Appendix B, 4.4	The licensee has proposed to provide a recharging capability or new apparatus which has a greater service life to insure a supply of emergency breathing air for a period of six hours.	NONE. Short lived component.

VYNPS Fire Protection and Appendix R Program		
PP 7011 Section	**PP 7011 text**	**TLAA**
No hits	NA	NA

Attachment 4 – Estimate of Crane Cycles

Estimate of outages for VYNPS:

Beginning of life =	1972	Actual date is 12/1/72 per PP7015
End of extended operation =	2032	Actual date is 3/21/2012
Current outage frequency =	1.5	years

Outages	Refuel outages to date	Till EOL (2032)	Total	
Best estimate	24	19	**43**	Assumes 18 month cycles for rest of plant life.
Conservative:	24	29	**53**	Assumes one outage per year for the rest of plant life.

Reactor building crane:

	Main hook	Aux hook
RB bridge crane rated load (tons):	110	7.73

Main hook rating from NRC SER (NVY 84-139) still current per OP 2200, Appendix B
Aux hook rating from MM2003-040 and OP 2200, Appendix B
Loads and weights taken from FVY 81-134, updated per OP 2200, Appendix B

Results for VYNPS RB crane:

	Cycles	**K**
Best estimate	**3007**	**0.40**
Conservative:	**7590**	**0.40**

Attachment 4 – Estimate of Crane Cycles

Best estimate

	Heavy loads	#	Weight	# Lifts per outage	# Lifts in 60 years	Load magnitude W	Load probability	Mean effective load factor K	Notes:
A	Reactor vessel head	1	54.00	2	86	0.491	0.029	0.0034	Off and on each outage.
B	Drywell head	1	44.00	2	86	0.400	0.029	0.0018	Off and on each outage.
C	Steam dryer	1	22.00	2	86	0.200	0.029	0.0002	Off and on each outage.
D	Steam separator (shroud head)	1	33.00	2	86	0.300	0.029	0.0008	Off and on each outage.
E	Shield blocks	2	64.00	4	172	0.582	0.057	0.0113	Off and on each outage.
		2	67.00	4	172	0.609	0.057	0.0129	Off and on each outage.
		2	71.50	4	172	0.650	0.057	0.0157	Off and on each outage.
F	New fuel storage vault plugs	3	3.00	6	258	0.027	0.086	0.0000	Off and on each outage.
G	Spent fuel pool gate	1	0.60	2	86	0.005	0.029	0.0000	Off and on each outage.
		1	0.45	2	86	0.004	0.029	0.0000	
H	Refueling slot plugs	3	6.00	6	258	0.055	0.086	0.0000	Off and on each outage.
		1	7.25	2	86	0.066	0.029	0.0000	
I	Vessel head insulation	1	4.50	2	86	0.041	0.029	0.0000	Off and on each outage.
J	Spent fuel shipping cask	1	110.00	1	43	1.000	0.014	0.0143	Empty in once each outage, full out once each outage.
		1	35.00	1	43	0.318	0.014	0.0005	Empty in once each outage.
K	Filter-demineralizer hatch	2	8.00	4	172	0.073	0.057	0.0000	Off and on each outage.
L	Contaminated equipment storage area hatches	2	2.50	4	172	0.023	0.057	0.0000	Off and on each outage.

Attachment 4 – Estimate of Crane Cycles

Best estimate

	Heavy loads	#	Weight	# Lifts per outage	# Lifts in 60 years	Load magnitude W	Load probability	Mean effective load factor K	Notes:
M	Reactor head strongback	1	4.00	2	86	0.036	0.029	0.0000	Off and on each outage.
N	Stud tensioner monorail	1	3.50	2	86	0.032	0.029	0.0000	Off and on each outage.
O	Cattle chute	1	14.00	2	86	0.127	0.029	0.0001	Off and on each outage.
P	Dryer/separator storage pit shield plug	1	43.50	2	86	0.395	0.029	0.0018	Off and on each outage.
		3	28.00	6	258	0.255	0.086	0.0014	Off and on each outage.
Q	Crane load block	1	6.00	3	129	0.055	0.043	0.0000	Assumed 3 empty cycles of main hook each outage.
R	HP water blaster	1	2.50	2	86	0.023	0.029	0.0000	In and out each outage.
S	Vessel service platform	1	5.00	0	40	0.045	0.013	0.0000	No longer used, 40 cycles.

Best estimate total cycles: 3007

Best estimate mean effective load factor K: 0.4004

Attachment 4 – Estimate of Crane Cycles

Conservative estimate

	Heavy Loads	#	Weight	# Lifts per outage	# Lifts in 60 years	Load magnitude W	Load probability	Mean effective load factor K	Notes:
a	Reactor vessel head	1	54.00	4	212	0.491	0.028	0.0033	Off and on, twice each outage
b	Drywell head	1	44.00	4	212	0.400	0.028	0.0018	Off and on, twice each outage
c	Steam dryer	1	22.00	4	212	0.200	0.028	0.0002	Off and on, twice each outage
d	Steam separator (shroud head)	1	33.00	4	212	0.300	0.028	0.0008	Off and on, twice each outage
e	Shield blocks	2	64.00	8	424	0.582	0.056	0.0110	Off and on, twice each outage
		2	67.00	8	424	0.609	0.056	0.0126	Off and on, twice each outage
		2	71.50	8	424	0.650	0.056	0.0153	Off and on, twice each outage
f	New fuel storage vault plugs	3	3.00	12	636	0.027	0.084	0.0000	Off and on, twice each outage
g	Spent fuel pool gate	1 1	0.60 0.45	4 4	212 212	0.005 0.004	0.028 0.028	0.0000 0.0000	Off and on, twice each outage
h	Refueling slot plugs	3 1	6.00 7.25	12 4	636 212	0.055 0.066	0.084 0.028	0.0000 0.0000	Off and on, twice each outage
i	Vessel head insulation	1	4.50	4	212	0.041	0.028	0.0000	Off and on, twice each outage
j	Spent fuel shipping cask	1	110.00	2	118	1.000	0.016	0.0155	Full cask out, twice per outage, plus 12 to empty SFP.
		1	35.00	2	118	0.318	0.016	0.0005	Empty cask in, twice per outage, plus 12 to empty SFP.
k	Filter-demineralizer hatch	2	8.00	8	424	0.073	0.056	0.0000	Off and on, twice each outage

Attachment 4 – Estimate of Crane Cycles

Conservative estimate

	Heavy Loads	#	Weight	# Lifts per outage	# Lifts in 60 years	Load magnitude W	Load probability	Mean effective load factor K	Notes:
l	Contaminated equipment storage area hatches	2	2.50	8	424	0.023	0.056	0.0000	Off and on, twice each outage
m	Reactor head strongback	1	4.00	4	212	0.036	0.028	0.0000	Off and on, twice each outage
n	Stud tensioner monorail	1	3.50	4	212	0.032	0.028	0.0000	Off and on, twice each outage
o	Cattle chute	1	14.00	4	212	0.127	0.028	0.0001	Off and on, twice each outage
p	Dryer/separator storage pit shield plug	1	43.50	4	212	0.395	0.028	0.0017	Off and on, twice each outage
		3	28.00	12	636	0.255	0.084	0.0014	Off and on, twice each outage
q	Crane load block	1	6.00	10	530	0.055	0.070	0.0000	10 empty cycles
r	HP water blaster	1	2.50	4	212	0.023	0.028	0.0000	Off and on, twice each outage
s	Vessel service platform	1	5.00	0	40	0.045	0.005	0.0000	No longer used, assumed 40 cycles.
	Conservative total cycles:				**7590**				
			Conservative mean effective load factor K:					**0.4007**	