APS A subsidiary of Pinnacle West Capital Corporation

Palo Verde Nuclear Generating Station

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102-06108-DCM/GAM
December 18, 2009

ATTN: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Dear Sirs:

Subject: Palo Verde Nuclear Generating Station (PVNGS) Units 1, 2, and 3 Docket Nos. STN 50-528, 50-529 and 50-530 Response to November 3, and December 4, 2009, Request for Additional Information Regarding Reactor Vessel and Reactor Coolant Pumps for the Review of the PVNGS License Renewal Application, and License Renewal Application Amendment No. 5

By letter dated November 3, 2009, as supplemented by letter dated December 4, 2009, the NRC issued a request for additional information (RAI) related to the PVNGS license renewal application (LRA). Enclosure 1 contains APS's response to the November 3, and December 4, 2009, RAI. Enclosure 2 contains PVNGS LRA updates to reflect changes made as a result of the RAI responses.

APS makes no commitments in this letter. Should you need further information regarding this submittal, please contact Russell A. Stroud, Licensing Section Leader, at (623) 393-5111.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on 12/18/09
(date)

Sincerely,

D. C. Mims

DCM/RAS/GAM

A138
NRR

Enclosures:

1. Response to November 3, and December 4, 2009, Request for Additional Information Regarding Reactor Vessel and Reactor Coolant Pumps for the Review of the PVNGS License Renewal Application

2. Palo Verde Nuclear Generating Station License Renewal Application Amendment No. 5

cc:	E. E. Collins Jr.	NRC Region IV Regional Administrator
	J. R. Hall	NRC NRR Project Manager
	R. I. Treadway	NRC Senior Resident Inspector for PVNGS
	L. M. Regner	NRC License Renewal Project Manager

ENCLOSURE 1

Response to November 3, and December 4, 2009, Request for Additional Information Regarding Reactor Vessel and Reactor Coolant Pumps for the Review of the PVNGS License Renewal Application

Part I. License Renewal Application (LRA) Time-Limited Aging Analyses (TLAAs) 4.2, "Reactor Vessel Neutron Embrittlement Analysis," and 4.7.7, "Absence of a TLAA for a Reactor Coolant Pump Flywheel Fatigue Crack Growth Analysis."

NRC RAI 4.2.1-1

LRA Tables 4.2-3 to 4.2-5 documented the 54 effective full power year (EFPY) upper-shelf energy (USE) values for PVNGS, Units 1, 2, and 3. LRA Tables 4.2-6 to 4.2-8 documented their corresponding 54 EFPY RT_{PTS} values. The staff found mistakes in these tables and requests corrections:

1. In Table 4.2-3 for Unit 1, the 54 EFPY USE values for the intermediate shell plates M-6701-2 and M-6701-3 are incorrect.
2. In Table 4.2-4 for Unit 2, all unirradiated USE values are incorrect.
3. In Table 4.2-6 for Unit 1, the nickel content for the lower shell axial welds is incorrect.

APS Response to RAI 4.2.1-1

The following corrections to Tables 4.2-3, 4.2-4, 4.2-6 are provided in LRA Amendment No. 5 in Enclosure 2 to this letter:

1. In Table 4.2-3 for PVNGS Unit 1, the 54 EFPY USE value for the intermediate shell plate M-6701-2 is being revised from 78.6 ft-lb to 75.4 ft-lb and the 54 EFPY USE value for the intermediate shell plate M-6701-3 is being revised from 75.4 ft-lb to 78.6 ft-lb.
2. In Table 4.2-4 for PVNGS Unit 2, the unirradiated USE values are being revised to reflect the correct Unit 2 values.
3. In Table 4.2-6 for PVNGS Unit 1, the nickel content for the lower shell axial welds is being revised from 0.035 percent to 0.079 percent.

In addition to the corrections described above, LRA Amendment No. 5 in Enclosure 2 contains the following corrections to Tables 4.2-4, 4.2-7, and 4.2-8:

4. In Tables 4.2-4 and 4.2-7, the heat numbers in the last lines are being corrected from 3P7869 to 4P7869.
5. In Table 4.2-4, the % Drop in USE @ EOL for Unit 2 lower shell axial welds 101-142A, B, C is being corrected from 21.44 to 23.94.

6. In Table 4.2-8, the 54 EFPY RT_{PTS} values for the first three lines were transposed with the next three. From the top down, they are being corrected to read 28.1, 68.1, 8.1, 38.1, 54.7, and 48.1 degrees Fahrenheit, instead of 38.1, 54.7, 48.1, 28.1, 68.1, and 8.1.

These errors have been entered into the PVNGS corrective action program as Palo Verde Action Request (PVAR) 3412350.

NRC RAI 4.2.1-2

LRA Section 4.2.1, "Neutron Fluence, Upper Shelf Energy and Adjusted Reference Temperature (Fluence, USE, and ART)," Tables 4.2-3 to 4.2-5 listed 54 EPPY USE values for the beltline materials of PVNGS, Units 1, 2, and 3. They are all derived in accordance with Regulatory Guide (RG) 1.99, Revision (Rev.) 2, "Radiation Embrittlement of Reactor Vessel Materials," without using surveillance data. The applicant also stated, "[t]he most recent coupon examination results also show that the decline in USE and increase in RT_{NDT}[reference temperature] in plate and weld materials are less than originally predicted by RG 1.99, Revision 2." One can conservatively estimate the ART and USE values for the beltline materials without using credible multiple surveillance data only when the estimates based on all measured surveillance data in accordance with RG 1.99, Rev. 2 are lower. Please provide your basis for not considering all surveillance data. Add references (i.e., surveillance reports) to support your clarification.

APS Response to RAI 4.2.1-2

1. The following statement is from LRA Section 4.2.1 on page 4.2-6:

   The most recent coupon examination results also show that the decline in USE and increase in RT_{NDT} [reference temperature] in plate and weld materials are less than originally predicted by Regulatory Guide 1.99, Revision 2, with only one exception. The exception is the Unit 3 weld coupon set....

   This statement is a paraphrase from the executive summaries of the most-recent coupon examination reports for each of the three Palo Verde units (Enclosures in References 1, 2, and 3, Executive Summary of each). This statement pertains to LRA Table 4.2-2, which summarizes these surveillance results. It is not in support of the embrittlement parameter projections of Tables 4.2-3 and following, but applies only to the results of these most-recent, 230° capsule examination results. The statement means only that (with one exception) the most-recently-withdrawn coupons (and, therefore, those with the highest exposure) exhibit less-severe reductions in toughness parameters than would be predicted using chemistry data and Regulatory Guide 1.99, Revision 2 methods, at the actual fluence measured by the internal coupon dosimetry.

The results of the earlier coupon examinations are consistent with the most-recent results, and are summarized in the credibility evaluation of these three most-recent examination reports (Enclosures in References 1, 2, and 3, Appendix D of each).

2. The predictions of 54 EFPY values without surveillance data for the summary tables of this section are necessary in order to provide a common basis for comparison of predicted values at the end of the period of extended operation. These predictions demonstrate that the surveillance program should be able to continue to demonstrate adequate margins of embrittlement parameters to analytic and regulatory limits for the period of extended operation. However the disposition of these TLAAs does not depend on a 10 CFR 54.21(c)(1)(i) validation of these predictions "without surveillance data" alone. Validation of the existing lifetime embrittlement projections also depends on the fact that with low-leakage cores the present predictions of a 54 EFPY lifetime neutron fluence are less than the design basis fluence used for these embrittlement projections. The disposition also depends on 10 CFR 54.21(c)(1)(iii) aging management (the surveillance program). See "Disposition...," LRA pages 4.2-6 and -7.

References for RAI 4.2.1-2:

1. Letter no. 102-05242 from APS to the NRC, "Palo Verde Nuclear Generating Station Unit 1 Reactor Vessel Material Surveillance Capsule at 230°," April 5, 2005 (transmittal of WCAP-16374-NP, "Analysis of Capsule 230° from Arizona Public Service Company Palo Verde Unit 1 Reactor Vessel Radiation Surveillance Program," February 2005) (ADAMS Accession No. not publicly available).

2. Letter no. 102-05457 from APS to the NRC, "Palo Verde Nuclear Generating Station Unit 2 Reactor Vessel Material Surveillance Capsule at 230°," April 4, 2006 (transmittal of WCAP-16524-NP, "Analysis of Capsule 230° from Arizona Public Service Company Palo Verde Unit 2 Reactor Vessel Radiation Surveillance Program," February 2006) (ADAMS Accession No. ML061040586).

3. Letter no. 102-05348 from APS to the NRC, "Palo Verde Nuclear Generating Station Unit 3 Analysis of Reactor Vessel Material Surveillance Capsule at 230°," September 26, 2005 (transmittal of WCAP-16449-NP, "Analysis of Capsule 230° from Arizona Public Service Company Palo Verde Unit 3 Reactor Vessel Radiation Surveillance Program," August 2005) (ADAMS Accession No. ML053390139).

NRC RAI 4.2.1-3

The staff found that the WCAP-15589 report (ML003764418), "Analysis of Capsule 38° from the Arizona Public Service Company Palo Verde Unit 1 Reactor Vessel Radiation Surveillance Program," labeled two surveillance data from Capsule 38° as intermediate shell plate M-6701-2 and used them in the chemistry factor calculation for this limiting plate. However, the WCAP-16374 report (ML051020500), "Analysis of Capsule 230° from the Arizona Public Service Company Palo Verde Unit 1 Reactor Vessel Radiation Surveillance Program," dropped the two Capsule 38° surveillance data from the chemistry factor calculation for intermediate shell plate M-6701-2. A statement on page D-3 of the WCAP-16374 report indicates a possible misidentification of specimens: "[t]he lower shell plate M4311-1 also has surveillance data but only one set up to this point (from Capsule 38°), thus it will not be evaluated." Please confirm that you misidentified surveillance specimens for PVNGS, Unit 1 in the WCAP-15589 report and explain how it happened. Or, explain the basis of discarding surveillance data from Capsule 38° in the WCAP-16374 report in calculating the chemistry factor for intermediate shell plate M-6701-2. Please confirm that Units 2 and 3, did not experience similar misidentifications as Unit 1.

APS Response to RAI 4.2.1-3

Two PVNGS Unit 1 Capsule 38° data points were inadvertently labeled as intermediate shell plate M-6701-2 in Revision 0 of WCAP-15589, October 2000, which was submitted to the NRC in APS letter no. 102-04500, October 20, 2000. This labeling error was corrected in Revision 1 to WCAP-15589, March 2003. All PVNGS Unit 1 base metal data from Capsule 38° are correctly labeled as lower shell plate M-4311-1 in Revision 1 to WCAP-15589. APS submitted Revision 1 to WCAP-15589 to the NRC in letter no. 102-06094, dated November 13, 2009 (Agencywide Document Access and Management System [ADAMS] Accession No. ML093290263).

WCAP-16374-NP, Revision 0, "Analysis of Capsule 230° from Arizona Public Service Company Palo Verde Unit 1 Reactor Vessel Radiation Surveillance Program," excluded the Capsule 38° data from the chemistry factor calculation for intermediate shell plate M-6701-2 because those data were from lower shell plate M-4311-1. Data from plate M-4311-1 were not evaluated in WCAP-16374 because results were available from only a single surveillance capsule, and thus were insufficient to evaluate the chemistry factor.

A possible contributor to the initial mislabeling of Unit 1 Capsule 38° data points in WCAP-15589 as intermediate shell plate M-6701-2 is that three of the Unit 1 capsules are lower shell plate M-4311-1 base metal material, and the other three are intermediate shell plate M-6701-2 base metal material (UFSAR Table 5.3-13). The 38° capsule report inadvertently labeled the capsule with the material in one of the other Unit 1 capsules.

PVNGS Units 2 and 3 each use a single surveillance plate (lower shell) base metal material in each surveillance program as shown in UFSAR Tables 5.3-14 and 5.3-15. The current reactor vessel surveillance program reports and the current UFSAR, Revision 15, correctly identify the surveillance program plate for Units 2 and 3. Therefore, it is confirmed that PVNGS Units 2 and 3 did not experience similar misidentifications as did PVNGS Unit 1 with respect to the surveillance plate material.

NRC RAI 4.7.7-1

The title of LRA Section 4.7.7 is "Absence of a TLAA for a Reactor Coolant Pump Flywheel Fatigue Crack Growth Analysis." To demonstrate that this is true, please confirm that the underlying stress and fracture mechanics analysis for the flywheels is not a part of the current licensing basis for PVNGS, Units 1, 2, and 3, or the underlying stress and fracture mechanics analysis does not contain fatigue and fatigue crack evaluations.

APS Response to RAI 4.7.7-1

The underlying stress and fracture mechanics analysis for the reactor coolant pump (RCP) flywheels does not contain fatigue and fatigue crack evaluations.

Technical Specification 5.5.7, Reactor Coolant Pump Flywheel Inspection Program, and Updated Final Safety Analysis Report 5.4.1.1, Pump Flywheel Integrity, provide the current licensing basis for the Palo Verde flywheel materials and their inspections. The RCP flywheel integrity description in UFSAR 5.4.1.1 was taken from Combustion Engineering Standard Safety Analysis Report (CESSAR) Section 5.4.1.1, which was based on the NRC regulatory position in Regulatory Guide 1.14, Revision 0 (Safety Guide 14). The NRC review of the RCP flywheel integrity descriptions in CESSAR 5.4.1.1 was documented in the Safety Evaluation Report for CESSAR System 80, (NUREG-0852) and Safety Evaluation Report Related to the Operation of PVNGS Units 1, 2, and 3 (NUREG-0857).

Regulatory Guide 1.14, Revision 0, (Safety Guide 14) Section B, Discussion, includes the following:

> *Reactor coolant pump flywheels are of a simple geometric shape, and normally are made of a ductile material. Their quality can be closely controlled, and their service conditions are not severe; therefore, the use of suitable material, and adequate design and inservice inspection can provide a sufficiently small probability of a flywheel failure that the consequences of failure need not be protected against.*

The stress and fracture mechanics analysis in the RCP flywheel design report does not contain any fatigue or time-dependent fatigue crack evaluations. The design report includes an evaluation of cracks that will permit either a ductile or brittle burst at overspeed. This evaluation uses a comparison of applied stress intensity factor K_I to the expected material toughness stress intensity factor K_{IC} to demonstrate that the crack sizes required for brittle fracture at design overspeed are much larger than those detectable by anticipated inspection methods. However, it includes neither a crack growth nor a fatigue analysis; nor any other indication of a time-dependent analysis supporting the safety determination. Therefore, the underlying stress and fracture mechanics analysis does not contain any fatigue or time-dependent fatigue crack evaluations.

Regulatory Guide 1.14, Revision 0, (Safety Guide 14) provided no guidance or recommendation for a crack evaluation similar to that included in the RCP flywheel design report, nor for a crack growth or fatigue analysis, nor for any other analysis that might be a TLAA.

The UFSAR 5.4.1.1 description of the Palo Verde RCP flywheel design criteria includes a minimum dynamic stress intensity factor K_{IC}, consistent with the Regulatory Guide 1.14, Revision 0, (Safety Guide 14) Regulatory Position C.1.c recommendation for a minimum K_{IC} of 100 ksi-in$^{1/2}$ (at operating temperature). The RCP flywheel design report addresses this recommendation for a minimum K_{IC} of 100 ksi-in$^{1/2}$. The RCP flywheel design report closely follows the alternative method suggested by Regulatory Guide 1.14, Revision 0, (Safety Guide 14) Regulatory Position C.1.c.3. That is, it demonstrates compliance with the recommendation for a minimum K_{IC} of 100 ksi-in$^{1/2}$ at operating temperature by (1) identifying a lower-bound K_{IC} curve for the same type of material, and (2) translating the curve along the temperature axis until a minimum K_{IC} of 45 ksi-in$^{1/2}$ occurs at the nil-ductility transition temperature (NDT) of the material.

The flywheels are also not subject to neutron radiation embrittlement or thermal embrittlement, and, therefore, no change in K_{IC} is expected in a design lifetime (including license extension). Therefore the use of a minimum K_{IC} material acceptance criterion does not imply a TLAA.

Technical Specification 5.5.7 requires an RCP flywheel inspection program to provide for the inspection of each RCP flywheel per the recommendations of regulatory position c.4.b of Regulatory Guide 1.14, Revision 0, October 1971.

NRC RAI 4.7.7-2

Discuss the evaluation of past examination results. Discussion should include results from the surface examination of all exposed surfaces and the complete ultrasonic volumetric examinations which were conducted approximately every 10 years. Please confirm that no flaws were indicated from the examinations.

APS Response to RAI 4.7.7-2

APS has performed ultrasonic test examinations approximately every three years and eddy current test examinations every 10 years in accordance with the Inservice Inspection (ISI) program on the flywheel of each of the RCPs at PVNGS Units 1, 2, and 3. No indications of degradation have been found in any of the RCP flywheels, and, therefore, no flaw evaluations have been performed.

Part II. Aging Management Program (AMP) B2.1.15, "Reactor Vessel Surveillance"

NRC RAI B2.1.15-1

LRA Section B2.1.15 states under Program Description: PVNGS, Units 1, 2, and 3, determined neutron embrittlement effects, consistent with RG 1.99, Rev. 2, by option 1[b], "Neutron Embrittlement Using Surveillance Data." This is misleading. The staff found from its review of LRA Section 4.2 that, after having considered option 1b of Generic Aging Lessons Learned Report (GALL) Aging Management Program (AMP) XI.M31, you actually used the more conservative option 1a, "Neutron Embrittlement Using Chemistry Tables" to evaluate USE and pressure-temperature limits for 60 years. Both options are consistent with RG 1.99, Rev. 2. Please make the necessary clarification.

APS Response to RAI B2.1.15-1

See also the response to RAI 4.2.1-2.

This apparent inconsistency between LRA Sections 4.2 and B2.1.15 is due to their different purposes. Section 4.2 includes tables and descriptions of projections of embrittlement parameters through the period of extended operation, particularly Tables 4.2-3 through 4.2-8. For simplicity and uniformity, these tables were constructed using chemistry data and tables, which, if applied to the reactor vessel surveillance program, would correspond to GALL XI.M31 option 1.a. Section 4.2 also briefly describes the surveillance program which is more fully described in Section B2.1.15; but Section 4.2 Tables 4.2-3 through 4.2-8 only present projections of embrittlement effects before results of surveillance data are applied. They do not reflect results of the reactor vessel surveillance program.

Section B2.1.15 contains the description of the reactor vessel surveillance program (the neutron fluence monitoring and vessel material coupon surveillance program) as it presently exists and as it will be implemented through the period of extended operation. This program will evaluate embrittlement parameters based on surveillance data, that is, using GALL XI.M31 option 1.b, as described.

The use of the past tense of the verb in Section B2.1.15 was therefore incorrect. The sentence is being revised to read "PVNGS determines neutron embrittlement effects, consistent with Regulatory Guide 1.99, Rev. 2, by option 1[b]." (See LRA Amendment 5 in Enclosure 2). That is, the *surveillance program* determines effects by option 1.b.

NRC RAI B2.1.15-2

LRA Section B2.1.15 stated under Operating Experience what the recent examination results of the reactor vessel surveillance data revealed regarding neutron fluence, USE, and RT_{NDT}. Evaluation of operating experience of this PVNGS AMP should not be limited to "the recent examination results." Please expand your discussion to include all tested reactor vessel surveillance data.

APS Response to RAI B2.1.15-2

Under "Operating Experience," LRA Section B2.1.15 states that "The recent [reactor vessel coupon] examination results ... show that decreases in USE and increases in RT_{NDT} are less than projected..." (with a minor exception, as described). The most recently examined coupons were exposed for the entire operating lives of the vessels up to the point of their withdrawal, and are, therefore, the best available indicators of lifetime embrittlement effects, superseding the results of examinations of earlier coupon sets withdrawn at lower neutron fluences.

Please see also LRA Section 4.2.1, and the APS response to RAI 4.2.1-2, which includes the citations of the docketed, most recent coupon examination reports for Palo Verde Units 1, 2, and 3. The results of the earlier coupon examinations are consistent with the most recent results, and are summarized in the Appendix D credibility evaluation of these three most-recent examination reports.

ENCLOSURE 2

Palo Verde Nuclear Generating Station
License Renewal Application Amendment No. 5

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<u>Source: RAI 4.2.1-1</u>

LRA Tables 4.2-3, 4.2-4, 4.2-6, 4.2-7, and 4.2-8 (LRA pages 4.2-13, 4.2-14, 4.2-16, 4.2-17, and 4.2-18) are revised as shown on the following pages.

Table 4.2-3 - PVNGS Unit 1 Vessel Material USE Projected at 60 Years (54 EFPY) (Note 1)

Material Description			**Cu wt%**	**Unirradiated USE ft-lbf**	**EOL ¼ T Fluence 10^{19} n/cm² (E>1 MeV)**	**% Drop in USE @ EOL[2]**	**54 EFPY Projected USE ft-lbf**	**EOL USE Acceptance Criterion ft-lbf**
Reactor Vessel Beltline Region Location	**Heat Number**	**Type**						
Lower Shell Plate M-4311-1	62467-1	A 533B	0.040	134	1.681	21.44	105.3	≥50
Lower Shell Plate M-4311-2	62817-1	A 533B	0.030	127	1.681	21.44	99.8	≥50
Lower Shell Plate M-4311-3	62722-1	A 533B	0.030	142	1.681	21.44	111.6	≥50
Intermediate Shell Plate M-6701-1	C4142-1	A 533B	0.070	83	1.681	21.44	65.2	≥50
Intermediate Shell Plate M-6701-2	C4188-2	A 533B	0.060	96	1.681	21.44	75.4~~78.6~~	≥50
Intermediate Shell Plate M-6701-3	C4188-1	A 533B	0.060	100	1.681	21.44	78.6~~75.4~~	≥50
Intermediate Shell Axial Welds 101-124A, B, C	4P6052	Linde 0091	0.047[3]	200	1.681	21.44	157.1	≥50
Lower Shell Axial Welds 101-142A, B, C	90071	Linde 0091	0.035[3]	140	1.681	21.44	110.0	≥50
Circumferential Weld 101-171	4P7869	Linde 124	0.031[3]	90	1.681	21.44	70.7	≥50

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1 The information in this table is extracted from the RVID and generated assuming a clad-metal EOL fluence of 3.29×10^{19} high-energy neutrons/cm².

2 % Drop in USE at EOL is determined using RG 1.99 Rev. 2, Position 1.2.

3 The weld information is the best-estimate value from CE-NPSD-1039, instead of the material's measured value.

Table 4.2-4 - PVNGS Unit 2 Vessel Material USE Projected at 60 Years (54 EFPY) (Note 1)

Material Description			**Cu wt%**	**Unirradiated USE ft-lbf**	**EOL ¼ T Fluence 10^{19} n/cm² (E>1.0MeV)**	**% Drop in USE @ EOL(2)**	**54 EFPY Projected USE ft-lbf**	**EOL USE Acceptance Criterion ft-lbf**
Reactor Vessel Beltline Region Location	**Heat Number**	**Type**						
Lower Shell Plate F-773-1	64071-1	A 533B	0.030	~~134~~ 105	1.681	21.44	82.5	≥50
Lower Shell Plate F-773-2	64065-1	A 533B	0.040	127	1.681	21.44	99.8	≥50
Lower Shell Plate F-773-3	63987-1	A 533B	0.050	~~142~~ 129	1.681	21.44	101.3	≥50
Intermediate Shell Plate F-765-4	63427-1	A 533B	0.030	~~83~~ 114	1.681	21.44	89.6	≥50
Intermediate Shell Plate F-765-5	63464-1	A 533B	0.030	~~96~~ 121	1.681	21.44	95.1	≥50
Intermediate Shell Plate F-765-6	63716-1	A 533B	0.040	~~100~~ 126	1.681	21.44	99.0	≥50
Intermediate Shell Axial Welds 101-124A, B, C	89833	Linde 124	0.046(3)	~~200~~ 100	1.681	21.44	78.6	≥50
Lower Shell Axial Welds 101-142A, B, C	3P7317	Linde 124	0.074(3)	~~140~~ 100	1.681	~~21.44~~23.94	76.1	≥50
Circumferential Welds 101-171	~~3~~4P7869	Linde 124	0.031(3)	~~90~~ 95	1.681	21.44	74.6	≥50

1 The information in this table is extracted from the RVID and generated assuming a clad-metal EOL fluence of 3.29×10^{19} high-energy neutrons/cm².
2 % Drop in USE at EOL is determined using RG 1.99 Rev. 2, Position 1.2.
3 The weld information is the best-estimate value from CE-NPSD-1039, instead of the material's measured value.

Table 4.2-6 - PVNGS Unit 1 Reactor Vessel Limiting RT_{PTS} at 60 Years (54 EFPY) (Note 1)

Material Description			**Chemical Composition**		**Chemistry Factors °F[2]**	**Initial RT_{NDT} °F**	**Neutron Fluence @ EOL 10^{19} n/cm² (E>1 MeV)**	**54 EFPY Fluence Factor**	**54 EFPY ΔRT_{PTS} °F**	**Margin °F[2]**	**54 EFPY RT_{PTS} °F**	**Screening Criteria °F**
Reactor Vessel Beltline Region Location	**Heat Number**	**Type**	**Cu wt%**	**Ni wt%**								
Lower Shell Plate M-4311-1	62467-1	A 533B	0.040	0.650	26	-10	3.29	1.312	34.1	34.0	58.1	≤270
Lower Shell Plate M-4311-2	62817-1	A 533B	0.030	0.620	20	-40	3.29	1.312	26.2	26.2	12.5	≤270
Lower Shell Plate M-4311-3	62722-1	A 533B	0.030	0.640	20	-20	3.29	1.312	26.2	26.2	32.5	≤270
Intermediate Shell Plate M-6701-1	C4142-1	A 533B	0.070	0.660	44	30	3.29	1.312	57.7	34.0	121.7	≤270
Intermediate Shell Plate M-6701-2	C4188-2	A 533B	0.060	0.610	37	40	3.29	1.312	48.5	34.0	122.5	≤270
Intermediate Shell Plate M-6701-3	C4188-1	A 533B	0.060	0.610	37	40	3.29	1.312	48.5	34.0	122.5	≤270
Intermediate Shell Axial Welds 101-124A, B, C	4P6052	Linde 0091	0.047	0.049	30.74	-50	3.29	1.312	40.3	40.3	30.6	≤300
Lower Shell Axial Welds 101-142A, B, C	90071	Linde 0091	0.035	0.0~~35~~79	29.32	-80	3.29	1.312	38.5	38.5	-3.0	≤300
Circumferential Weld 101-171	4P7869	Linde 124	0.031	0.096	28.73	-70	3.29	1.312	37.7	37.7	5.4	≤300

1 The information in this table is extracted from the RVID and generated assuming a clad-metal EOL fluence of 3.29×10^{19} high-energy neutrons/cm².

2 The Chemistry Factors are determined using RG 1.99 Rev. 2, Position 1.1.

Table 4.2-7 - PVNGS Unit 2 Reactor Vessel Limiting RT_{PTS} at 60 Years (54 EFPY) (Note 1)

Material Description			**Chemical Composition**		**Chemistry Factors °F(2)**	**Initial RT_{NDT} °F**	**Neutron Fluence @ EOL 10^{19} n/cm² (E>1 MeV)**	**54 EFPY Fluence Factor**	**54 EFPY ΔRT_{PTS} °F**	**Margin °F(2)**	**54 EFPY RT_{PTS} °F**	**Screening Criteria °F**
Reactor Vessel Beltline Region Location	**Heat Number**	**Type**	**Cu wt%**	**Ni wt%**								
Lower Shell Plate F-773-1	64071-1	A 533B	0.030	0.670	20	10	3.29	1.312	26.2	26.2	62.5	≤270
Lower Shell Plate F-773-2	64065-1	A 533B	0.040	0.640	26	0	3.29	1.312	34.1	34.0	68.1	≤270
Lower Shell Plate F-773-3	63987-1	A 533B	0.050	0.660	31	-60	3.29	1.312	40.7	34.0	14.7	≤270
Intermediate Shell Plate F-765-4	63427-1	A 533B	0.030	0.670	20	-20	3.29	1.312	26.2	26.2	32.5	≤270
Intermediate Shell Plate F-765-5	63464-1	A 533B	0.030	0.650	20	10	3.29	1.312	26.2	26.2	62.5	≤270
Intermediate Shell Plate F-765-6	63716-1	A 533B	0.040	0.670	26	10	3.29	1.312	34.1	34.0	78.1	≤270
Intermediate Shell Axial Welds 101-124A, B, C	89833	Linde 124	0.046	0.059	31.51	-60	3.29	1.312	41.3	41.3	22.6	≤300
Lower Shell Axial Welds 101-142A, B, C	3P7317	Linde 124	0.074	0.067	41.17	-80	3.29	1.312	54.0	54.0	28.0	≤300
Circumferential Weld 101-171	34P7869	Linde 124	0.031	0.096	28.73	-30	3.29	1.312	37.7	37.7	45.4	≤300

1 The information in this table is extracted from the RVID and generated assuming a clad-metal EOL fluence of 3.29×10^{19} high-energy neutrons/cm².
2 The Chemistry Factors are determined using RG 1.99 Rev. 2, Position 1.1.

Table 4.2-8 - PVNGS Unit 3 Reactor Vessel Limiting RT_{PTS} at 60 Years (54 EFPY) (Note 1)

Material Description			**Chemical Composition**		**Chemistry Factors °F(2)**	**Initial RT_{NDT} °F**	**Neutron Fluence @ EOL 10^{19} n/cm² (E>1 MeV)**	**54 EFPY Fluence Factor**	**54 EFPY ΔRT_{PTS} °F**	**Margin °F(2)**	**54 EFPY RT_{PTS} °F**	**Screening Criteria °F**
Reactor Vessel Beltline Region Location	**Heat Number**	**Type**	**Cu wt%**	**Ni wt%**								
Lower Shell Plate F-6411-1	79545-1	A 533B	0.040	0.640	26	-40.0	3.29	1.312	34.1	34.0	~~38.1~~ 28.1	≤270
Lower Shell Plate F-6411-2	79745-1	A 533B	0.040	0.660	26	0.0	3.29	1.312	34.1	34.0	~~54.7~~ 68.1	≤270
Lower Shell Plate F-6411-3	79659-1	A 533B	0.040	0.650	26	-60.0	3.29	1.312	34.1	34.0	~~48.1~~ 8.1	≤270
Intermediate Shell Plate F-6407-4	65202-1	A 533B	0.040	0.620	26	-30.0	3.29	1.312	34.1	34.0	~~28.1~~ 38.1	≤270
Intermediate Shell Plate F-6407-5	65219-1	A 533B	0.050	0.610	31	-20.0	3.29	1.312	40.7	34.0	~~68.1~~ 54.7	≤270
Intermediate Shell Plate F-6407-6	79011-1	A 533B	0.040	0.610	26	-20.0	3.29	1.312	34.1	34.0	~~8.1~~ 48.1	≤270
Intermediate Shell Axial Welds 101-124A, B, C	4P7869	Linde 124 SAW	0.031	0.096	28.73	-50	3.29	1.312	37.7	37.7	25.4	≤300
Lower Shell Axial Welds 101-142A, B, C	4P7869	Linde 124 SAW	0.031	0.096	28.73	-50	3.29	1.312	37.7	37.7	25.4	≤300
Circumferential Weld 101-171	4P7869	Linde 124 SAW	0.031	0.096	28.73	-70	3.29	1.312	37.7	37.7	5.4	≤300

1 The information in this table is extracted from the RVID and generated assuming a clad-metal EOL fluence of 3.29×10^{19} high-energy neutrons/cm².
2 The Chemistry Factors are determined using RG 1.99 Rev. 2, Position 1.1.

Palo Verde Nuclear Generating Station
License Renewal Application
Amendment No. 5

Source RAI B2.1.15-1

The second paragraph in Section B2.1.15 on Page B-54 is being revised as follows:

Since peak neutron fluence at the end of the design life may exceed 10^{17}n/cm^2 (E>1MeV) and since vessel material coupons were installed, PVNGS determines ~~determined~~ neutron embrittlement effects, consistent with Regulatory Guide1.99, Rev.2, by option1[b], "Neutron Embrittlement Using Surveillance Data." Actual reactor vessel coupons are used. Limiting heat-affected-zone (HAZ) materials were selected from limiting plate materials.