VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

January 16, 2002

United States Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555-0001

Serial No.:	01-686
LR/MWH	R0
Docket Nos.:	50-280/281
	50-338/339
License Nos.:	DPR-32/37
	NPF-4/7

Gentlemen:

VIRGINIA ELECTRIC AND POWER COMPANY (DOMINION)
SURRY AND NORTH ANNA POWER STATIONS UNITS 1 AND 2
REQUEST FOR ADDITIONAL INFORMATION
LICENSE RENEWAL APPLICATIONS

In an October 22, 2001 letter, the NRC requested additional information regarding the license renewal applications (LRAs) for Surry and North Anna Power Stations. The attachment to this letter contains the responses to the Requests for Additional Information (RAIs) associated with Sections B2.0, 4.1, 4.3, and 4.7.4 of the LRA.

Responses to RAIs associated with Section 2.1 are not provided herein, but will be provided by separate correspondence at a later date.

Should you have any questions regarding this submittal, please contact Mr. J. E. Wroniewicz at (804) 273-2186.

Very truly yours,

David A. Christian
Senior Vice President – Nuclear Operations and Chief Nuclear Officer

Attachment

Commitments made in this letter: None

A086

cc:

U. S. Nuclear Regulatory Commission
Region II
Sam Nunn Atlanta Federal Center
61 Forsyth Street, SW
Suite 23T85
Atlanta, GA 30303-8931

Mr. M. J. Morgan
NRC Senior Resident Inspector
North Anna Power Station

Mr. R. A. Musser
NRC Senior Resident Inspector
Surry Power Station

Mr. J. E. Reasor, Jr.
Old Dominion Electric Cooperative
Innsbrook Corporate Center
4201 Dominion Blvd.
Suite 300
Glen Allen, VA 23060

Ms. Ellie Irons, EIR Program Manager
Virginia Dept. of Environmental Quality
629 East Main St., 6th Fl
Richmond, VA 23219

Mr. David Paylor, Program Coordinator
Virginia Dept. of Environmental Quality
P.O. Box 10009
Richmond, VA 23240-0009

Mr. Joe Hassell, Environmental Manager
Virginia Dept. of Environmental Quality
Water Division
P.O. Box 10009
Richmond, VA 23240-0009

Mr. Frank Daniel, Regional Director
Virginia Dept. of Environmental Quality
Tidewater Regional Office
5636 Southern Blvd.
Virginia Beach, VA 23462

Mr. Gregory Clayton, Regional Director
Virginia Dept. of Environmental Quality
Northern Virginia Regional Office
13901 Crown Ct.
Woodbridge, VA 22193

Mr. Frank Fulgham, Program Manager
Virginia Dept. of Agriculture & Consumer Services
Office of Plant & Pest Services
1100 Bank St.
Richmond, VA 23219

Mr. David Brickley, Agency Director
Virginia Dept. of Conservation & Recreation
203 Governor St.
Richmond, VA 23219

Mr. William Woodfin, Director
Virginia Dept. of Game & Inland Fisheries
4010 West Broad St.
Richmond, VA 23230

Mr. Robert Hicks, Director
Virginia Dept. of Health
Office of Environmental Health Services
1500 East Main St., Room 115
Richmond, VA 23219

Ms. Kathleen S. Kilpatrick, Director
Virginia Dept. of Historic Resources
State Historic Preservation Office
2801 Kensington Ave.
Richmond, VA 23221

Dr. Ethel Eaton, Archeologist Senior
Virginia Dept. of Historic Resources
State Historic Preservation Office
2801 Kensington Ave.
Richmond, VA 23221

Mr. Robert W. Grabb, Assistant Commissioner
Virginia Marine Resources Commission
2600 Washington Ave.
Newport News, VA 23607

Dr. John Olney, Associate Professor
Virginia Institute of Marine Science
School of Marine Science
Gloucester Point, VA 23062

Mr. John Simkins
Virginia Dept. of Transportation
Environmental Division
1401 East Broad St.
Richmond, VA 23219

Mr. Robert Burnley
Virginia Economic Development Partnership
901 East Byrd St.
Richmond, VA 23219

Mr. William F. Stephens, Director
Virginia State Corporation Commission
Division of Energy Regulation
1300 East Main St., 4th Fl., Tyler Bldg.
Richmond, VA 23219

Mr. Michael Cline, State Coordinator
Commonwealth of Virginia
Department of Emergency Management
10501 Trade Rd.
Richmond, VA 23236-3713

Mr. Terry Lewis, County Administrator
P.O. Box 65
Surry, VA 23883

Mr. Lee Lintecum
Louisa County Administrator
P.O. Box 160
Louisa, VA 23093

Mr. Douglas C. Walker
Acting Spotsylvania County Administrator
P.O. Box 99
Spotsylvania, VA 22553

Ms. Brenda G. Bailey, County Administrator
P.O. Box 11
Orange, VA 22960

Chairman Reeva Tilley
Virginia Council on Indians
P.O. Box 1475
Richmond, VA 23218

Mr. Don Lillywhite, Director
Economics Information Services
Virginia Employment Commission
State Data Center
703 East Main St., Room 213
Richmond, VA 23219

Mr. Alan Zoellner
Government Information Department
Swem Library
College of William and Mary
Landrum Dr.
P.O. Box 8794
Williamsburg, VA 23187-8794

Mr. Walter Newsome
Government Information Resources
Alderman Library
University of Virginia
160 McCormick Rd.
P.O. Box 400154
Charlottesville, VA 22904-4154

COMMONWEALTH OF VIRGINIA)
)
COUNTY OF HENRICO)

The foregoing document was acknowledged before me, in and for the County and Commonwealth aforesaid, today by David A. Christian who is Senior Vice President and Chief Nuclear Officer of Virginia Electric and Power Company. He has affirmed before me that he is duly authorized to execute and file the foregoing document in behalf of that Company, and that the statements in the document are true to the best of his knowledge and belief.

Acknowledged before me this 16th day of January, 2001.

My Commission Expires: 3-31-04.

Maggie McClure
Notary Public

(SEAL)

Attachment

License Renewal – Response to RAI
Serial No. 01-686

Response to Request for Additional Information

Dated October 22, 2001

Surry and North Anna Power Stations, Units 1 and 2

License Renewal Applications

Sections B2.0, 4.1, 4.3, and 4.7.4

Virginia Electric and Power Company
(Dominion)

Appendix B - Aging Management Activities

RAI B2.0-1:

In the past, applicants have described the aging management programs in term other than the ten elements as defined in the Standard Review Plan (SRP). On the bases of this concern, the staff is asking that the applicant define the elements of their aging management activities for the staff to clearly understand its application throughout Appendix B. The applicant has the option to verify that they used the same definition presented in the SRP in its development of its aging management activities.

In addition, the applicant takes credit for its 10 CFR Part 50, Appendix B program to satisfy three of the ten elements of an aging management program. The staff generically accepts Appendix B activities in fulfillment of the corrective action, confirmation process, and administrative control attributes for an aging management program. However, the staff needs to verify that an applicant is correctly applying its Appendix B program to these attributes. Therefore, please provide a description of how Appendix B is applied to the corrective action, confirmation process, and administrative control attributes for an aging management program. In addition, the applicant needs to add a summary description of the QAP as it specifically addresses the corrective action, confirmation process, and administrative controls attributes for an aging management programs to its FSAR Supplement.

Dominion Response:

For the ten elements of the Aging Management Activity descriptions, Dominion uses the definitions provided in the NRC Standard Review Plan (SRP).

The Dominion Quality Assurance Program implements the requirements of 10 CFR 50, Appendix B, and is consistent with Section A.2 of the NRC Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants (NUREG-1800, April, 2001). The requirements of 10 CFR 50, Appendix B are implemented by the Quality Assurance Program as described in Chapter 17 of the Updated Final Safety Analysis Reports for the Surry and North Anna Power Stations. The Quality Assurance Program includes the elements of corrective action, confirmation process, and administrative controls. License renewal takes credit for these three elements and confirms their applicability to the safety-related and non-safety-related structures, systems, and components that are within the scope of license renewal.

SCOPE

The Corrective Action System, which includes the elements of corrective action, confirmation process, and administrative controls, applies to all safety-related and non-safety-related structures, systems, and components that are within the scope of license renewal. Corrective actions implement the requirements of 10 CFR 50, Appendix B.

PREVENTIVE ACTIONS

The Corrective Action System provides a means to correct conditions that are determined to be adverse to quality. No preventive actions are performed.

PARAMETERS MONITORED OR INSPECTED

The Corrective Action System does not involve monitoring any particular system, structure or component within the scope of license renewal. However, if unexpected inspection results occur, the Corrective Action System provides the mechanism to perform necessary evaluations, repairs, or replacements; and to confirm that corrective actions are performed to ensure that intended functions of systems, structures, and components are maintained.

DETECTION OF AGING EFFECTS

The Corrective Action System does not detect aging effects; it provides the mechanism for evaluating and resolving unexpected results from other monitoring activities.

MONITORING AND TRENDING

The Corrective Action System provides the mechanism for timely evaluation and resolution of unexpected results from inspection and testing activities. Plant Issue documents, which summarize these unexpected results, are monitored and trended. Resolution of plant issues ensures that intended functions of systems, structures, and components are maintained.

ACCEPTANCE CRITERIA

The Corrective Action System does not include acceptance criteria; it provides the means to resolve unexpected inspection and testing results that do not comply with acceptance criteria defined in other aging management activities.

CORRECTIVE ACTIONS

Corrective actions for conditions that are adverse to quality are performed in accordance with the Corrective Action System as part of the Quality Assurance Program. Any resultant maintenance or repair activities are performed in accordance with the Work Control Process. The corrective action process provides reasonable assurance that deficiencies adverse to quality are either promptly corrected or are evaluated to be acceptable. Where evaluations are performed without repair or replacement, engineering analysis reasonably assures that the component intended function is maintained consistent with the current licensing basis. If the deficiency is assessed to be significantly adverse to quality, the cause of the condition is determined, and an action plan is developed to preclude repetition. The Corrective Action System identifies repetitive discrepancies and initiates additional corrective action to preclude recurrence.

CONFIRMATION PROCESS

Confirmation of activities taken in accordance with the Corrective Action System

ensures that resolutions are timely and effective.

ADMINISTRATIVE CONTROLS

Administrative and implementation procedures are reviewed, approved, and maintained as controlled documents in accordance with the procedure control process and the Quality Assurance Program.

OPERATING EXPERIENCE

Implementing procedures for the Corrective Action System provide direction to document unexpected inspection and testing results as Plant Issues. Plant Issues document aging affects and significant operating events. Resolution of Plant Issues in accordance with 10 CFR 50, Appendix B maintains the intended functions of systems, structures, and components. Findings from the Corrective Action System are used to enhance the Corrective Action System.

SUMMARY

The Corrective Action System includes the requirements of 10 CFR 50, Appendix B as implemented by the Quality Assurance Program that is described in Chapter 17 of the Updated Final Safety Analysis Reports for Surry and North Anna. The Quality Assurance Program includes the elements of corrective action, confirmation process, and administrative controls; and is applicable to the safety-related and non-safety-related structures, systems, and components that are within the scope of license renewal.

Implementing procedures for the Corrective Action System ensure that unexpected inspection and testing results are documented as Plant Issues and resolved to ensure that intended functions of systems, structures, and components are maintained. The Corrective Action System will remain in place during the period of extended operation.

Section 4.1, "Identification of Time-Limited Aging Analyses"

RAI 4.1-1:

In both LRAs, Table 4.1-1 the applicant did not identify pipe break postulation based on cumulative usage factor (CUF) as a TLAA. Section 3A.46 of the NAS updated final safety analysis report (UFSAR) describes the criteria used to provide protection against pipe whip inside the containment. Part of the criteria specifies the postulation of pipe breaks at locations where the CUF exceeds 0.1. Although the fatigue usage factor calculation was identified as a TLAA, the pipe break criterion was not identified as a TLAA. However, the usage factor calculation used to identify postulated pipe break locations meets the definition of a TLAA as specified in 10 CFR 54.3 and, therefore, the staff considers the associated criteria for pipe break postulation to be a TLAA. Provide a description of the TLAA performed to address the pipe break criteria for North Anna. Also identify any pipe break postulations based on CUF at Surry and describe the TLAA performed for these locations. Indicate how these TLAAs meet the requirements of 10 CFR 54.21(c).

Dominion Response:

To meet the General Design Criteria (GDC-4), intermediate breaks between the terminal points of a Class 1 pipe are postulated when the calculated cumulative usage factor (CUF) is equal to or greater than 0.1. Our review has established that the expected number of transients in 60 years of North Anna Power Station (NAPS) operation will be fewer than the design transients used in the analyses. Consequently, the CUFs will not change for the period of extended operation. Therefore, the CLB pipe break locations remain the same for the period of extended operation.

Only the pressurizer surge lines of Surry Power Station (SPS) have been analyzed to ASME Section III Class 1 rules. Our review establishes that the expected number of transients in 60 years of SPS operation will be fewer than the design transients used in the analyses. Consequently, the CUFs will not change for the period of extended operation. Therefore, the current SPS CLB pipe break locations remain the same for the period of extended operation.

Section 4.3, "Metal Fatigue"

RAI 4.3-1:

In both LRAs, Section 4.3.1, the applicant discusses its evaluation of the fatigue TLAA for ASME Class 1 components. In this discussion, the applicant indicates that, on the bases of its review of the plant operating history, the number of cycles assumed in the design of the ASME Class 1 components are conservative and bounding for the period of extended operation. Table 5.2-4 of the North Anna UFSAR and Table 4.1-8 of the Surry UFSAR contain a list transient design conditions and associated design cycles. Provide the following information for each transient listed in these tables:

a. The current number of operating cycles and a description of the method used to determine the number and severity of the design transients from the plant operating history.

b. The number of operating cycles estimated for 60 years of plant operation and a description of the method used to estimate the number of cycles at 60 years.

c. A comparison of the design transients listed in the UFSAR with the transients monitored by the Transient Cyclic Counting Program (TCCP) as shown in Section B3.2 of the LRAs. Identify any transients listed in the UFSAR that are not monitored by the TCCP and explain why it is not necessary to monitor these transients.

d. Section B3.2 of the NAS LRA indicates that the charging line nozzle has been instrumented to evaluate the impact of charging line flow transients. Describe the instrumentation used to monitor charging flow transients explain how the data obtained from this instrumentation is used by the TCCP.

e. In both LRAs, Table 3.1.3-W1, the applicant provides the response to Renewal Applicant Action Item 11 specified in WCAP -14577, Revision 1-A regarding fatigue TLAA of the reactor vessel internals. The response indicates that the TCCP will assure that the transients will remain within their design values for the period of extended operation. List the transients that contribute to the fatigue usage for each component listed in Table 3-3 of WCAP-14577, Revision 1-A and discuss how the TCCP monitors these transients.

Dominion Response:

a. The transient cycle counting program has been an on-going program at NAPS since the initial startup of each unit, as required by Technical Specifications. The SPS Transient Cycle Counting Program was initiated in January 2000 and operational data since the initial startup of each SPS unit have been included in the cycle counting program.

   Tables 4.3-1-1 through 4.3-1-4 list the design transients from the Updated Final Safety Analysis Report (Table 4.1-8 for SPS and Table 5.2-4 for NAPS), the number of design cycles, the number of cycles experienced, and the number of cycles

projected for 60 years of operation. The bases for exclusion from the Transient Cycle Counting Program are provided in the footnote. The transient cycle counting data provided in Tables 4.3-1-1 through 4.3-1-4 are compiled through December 31, 1999, for SPS and May 18, 2001, for NAPS. Tables 4.3-1-5 and 4.3-1-6 provide a description of the methods used to determine the number and severity of the design transients monitored for SPS and NAPS, respectively. The Transient Cycle Counting Program will be continued through the period of extended operation for both plants.

b. Projected cycles for 60 years of SPS and NAPS operation are given in Tables 4.3-1-1 through 4.3-1-4. For NAPS, the number of transient occurrences observed for heatup, cooldown, step load increase of 10% full power, step load decrease of 10% full power, large step load decrease, loss of flow, and reactor trip from full power are linearly extrapolated to 60 years. This methodology provides a very conservative prediction of future cycles, in that the units typically experience a greater number of transients during the early years of operation.

   For SPS, the number of transient occurrences observed for a step load increase of 10% full power, a step load decrease of 10% full power, and a step load reduction from 100% to 50% load are linearly extrapolated to 60 years. For the heatup, cooldown, and reactor trip transients, it was concluded that utilizing the data for all years of operation through December 1999 and linearly extrapolating to 60 years would result in overly conservative projections. This over-projection is due to the significantly large number of startup, shutdown, and reactor trip events that occurred during the first ten years of plant operation. Subsequent to this time period, the frequency of heatups, cooldowns, and reactor trips have been reduced significantly. In addition, the plant fuel cycle length was changed from 12 months to 18 months during the first ten years of operation, which also contributes to fewer transient events. Therefore, a more realistic approach has been used to extrapolate events out to 60 years. The average frequency on a per-year basis has been derived from more recent operating history (i.e., the last ten years of operation) and has been used as the basis for projecting future cycles. The average frequency of occurrences on a per year basis for the past 10 years has been doubled for conservatism.

c. Tables 4.3-1-1 through 4.3-1-4 list the design transients from the Updated Final Safety Analysis Report (Table 4.1-8 for Surry and Table 5.2-4 for North Anna), the number of design cycles, the number of cycles experienced, and the projected 60-year cycles. The bases for exclusion from the Transient Cycle Counting Program are provided by footnote.

d. Temperature data from the existing plant instrumentation for the charging lines is being collected so that the operating and design transients for the charging nozzles can be reviewed to validate the design transients.

e. The SPS and NAPS reactor internals were designed to Westinghouse criteria, which were established prior to the issuance of the ASME Code Section III Subsection NG. The Westinghouse criteria contained no TLAAs and used pressure load

calculations instead of fatigue calculations.

The Westinghouse design of the reactor vessel internals is similar for both the three-loop plants designed with the Westinghouse criteria and the three-loop plants designed with ASME Section III Subsection NG criteria. In addition, the design transients for the reactor coolant system (RCS), including the reactor vessel internals are similar.

Dominion has evaluated the 40-year reactor coolant system design transients and has concluded that they are applicable to the period of extended operation. The transient cycle counting program will monitor the design transients to provide reasonable assurance that the design cycles are not exceeded.

TABLE 4.3-1-1. SURRY UNIT 1 PROJECTED CYCLES AT 60 YEARS OF OPERATION (OCCURRENCES BASED ON DATA FROM INITIAL PLANT STARTUP THROUGH 12/31/99)

TRANSIENT NUMBER	TRANSIENT DESCRIPTION	UFSAR TABLE 4.1-8 DESIGN CYCLES	UNIT 1 CYCLES	UNIT 1 PROJECTED CYCLES
1	Heatup at 100°F/hour	200	100	171
2	Cooldown at 100°F/hour	200	99	170
3	Loading at 5% of full power per min.(15% to 100% equals one cycle)	29,000	Note 1	<29,000
4	Unloading at 5% of full power per min. (100% to 15% equals one cycle)	29,000	Note 1	<29,000
5	Step load increase of 10% full power (but not to exceed full power)	2000	6	13
6	Step load decrease of 10% full power	2000	7	15
7	Step load reduction from 100% to 50%load	200	29	63
8	Reactor Trip from full power	400	124	208
9	Hydrostatic test pressure, 3107 psi and 100°F	5	1	Note 2
10	Hydrostatic test pressure, 2485 psi and 400°F	40	Note 3	<40
11	Steady State Fluctuation	∞	Note 4	∞

Notes:

1. Not counted. Basis for design cycle estimate is load follow operation. Unit does not operate in load follow mode. Instead, the unit is being operated in base-load mode.
2. Not Counted. One test performed during pre-operational testing. No additional testing is planned for the current or extended license period.
3. Not counted. ASME Code Case N-498-1 does not require a hydrostatic pressure test above normal operating pressure. No additional testing to be performed.
4. Not counted. Infinite number of fluctuations assumed.

TABLE 4.3-1-2. SURRY UNIT 2 PROJECTED CYCLES AT 60 YEARS OF OPERATION (OCCURRENCES BASED ON DATA FROM INITIAL PLANT STARTUP THROUGH 12/31/99)

TRANSIENT NUMBER	TRANSIENT DESCRIPTION	UFSAR TABLE 4.1-8 DESIGN CYCLES	UNIT 2 CYCLES	UNIT 2 PROJECTED CYCLES
1	Heatup at 100°F/hour	200	92	191
2	Cooldown at 100°F/hour	200	91	190
3	Loading at 5% of full power per min. (15% to 100% equals one cycle)	29,000	Note 1	<29,000
4	Unloading at 5% of full power per min.(100% to 15% equals one cycle)	29,000	Note 1	<29,000
5	Step load increase of 10% full power (but not to exceed full power)	2000	7	16
6	Step load decrease of 10% full power	2000	8	18
7	Step load reduction from 100% to 50% load	200	31	69
8	Reactor Trip from full power	400	119	232
9	Hydrostatic test pressure, 3107 psi and 100°F	5	1	Note 2
10	Hydrostatic test pressure, 2485 psi and 400°F	40	Note 3	<40
11	Steady State Fluctuation	∞	Note 4	∞

Notes:

1. Not counted. Basis for design cycle estimate is load follow operation. Unit does not operate in load follow mode. Instead, the unit is being operated in base-load mode.
2. Not Counted. One test performed during pre-operational testing. No additional testing is planned for the current or extended license period.
3. Not counted. ASME Code Case N-498-1 does not require a hydrostatic pressure test above normal operating pressure. No additional testing to be performed.
4. Not counted. Infinite number of fluctuations assumed.

TABLE-4.3-1-3 NORTH ANNA UNIT 1 PROJECTED CYCLES AT 60 YEARS OF OPERATION (OCCURRENCES BASED ON DATA FROM INITIAL PLANT STARTUP THROUGH 5/18/01)

TRANSIENT NUMBER	TRANSIENT DESCRIPTION	UFSAR TABLE 5.2-4 DESIGN CYCLES	UNIT 1 CYCLES	UNIT 1 PROJECTED CYCLES
Normal Transients				
1	Heatup at 100°F/hour	200	40	105
2	Cooldown at 100°F/hour	200	39	104
3	Loading at 5% of full power per min. (15% to 100% equals one cycle)	18,300	Note 1	<18,300
4	Unloading at 5% of full power per min.(100% to 15% equals one cycle)	18,300	Note 1	<18,300
5	Step load increase of 10% full power	2000	3	9
6	Step load decrease of 10% full power	2000	3	9
7	Large step loads decrease	200	2	6
8	Steady State Fluctuation	∞	Note 2	∞
Upset Conditions				
9	Loss of load, w/o immediate turbine or Rx trip	80	0	Note 3
10	Loss of power (blackout with natural circulation in RCS)	40	0	Note 3
11	Loss of Flow (partial loss of flow 1 pump only)	80	1	3
12	Reactor Trip from full power	400	70	186
13	Inadvertent auxiliary spray	10	0	Note 3
Faulted Conditions				
14	Main reactor coolant pipe break	1	Note 5	Note 5
15	Steam pipe break	1	Note 5	Note 5
16	Design-basis earthquake	1	Note 5	Note 5
Test Conditions				
17	Turbine roll test	10	Note 4	<10
Hydrostatic Test Conditions				
18	Primary side	5	Note 4	<5
19	Secondary side	5	Note 4	<5
20	Primary-side leak test	50	Note 6	<50

TABLE-4.3-1-3 NORTH ANNA UNIT 1 PROJECTED CYCLES AT 60 YEARS OF OPERATION (OCCURRENCES BASED ON DATA FROM INITIAL PLANT STARTUP THROUGH 5/18/01) (CONT.)

Notes:

1. Not counted. Basis for design cycle estimate is load follow operation. Unit does not operate in load follow mode. Instead, the unit is being operated in base-load mode.
2. Not counted. Infinite number of fluctuations assumed.
3. This transient is tracked. However, it has not occurred in the operational history of the units. No changes to operational philosophy are expected, therefore, this transient is not expected to occur during current license period or period of extended operation. Hence, this transient is projected to not exceed its design cycles limits.
4. Not counted. Test is an initial startup test only.
5. Not counted. Faulted condition occurs only once.
6. Not counted. Leak tests are performed at operating pressure. No system cyclic loading is involved.

TABLE-4.3-1-4 NORTH ANNA UNIT 2 PROJECTED CYCLES AT 60 YEARS OF OPERATION (OCCURRENCES BASED ON DATA FROM INITIAL PLANT STARTUP THROUGH 5/18/01)

TRANSIENT NUMBER	TRANSIENT DESCRIPTION	UFSAR TABLE 5.2-4 DESIGN CYCLES	UNIT 2 CYCLES	UNIT 2 PROJECTED CYCLES
Normal Transients				
1	Heatup at 100°F/hour	200	34	98
2	Cooldown at 100°F/hour	200	33	97
3	Loading at 5% of full power per min. (15% to 100% equals one cycle)	18,300	Note 1	<18,300
4	Unloading at 5% of full power per min. (100% to 15% equals one cycle)	18,300	Note 1	<18,300
5	Step load increase of 10% full power	2000	3	9
6	Step load decrease of 10% full power	2000	3	9
7	Large step loads decrease	200	2	6
8	Steady State Fluctuation	∞	Note 2	∞
Upset Conditions				
9	Loss of load, w/o immediate turbine or Rx trip	80	0	Note 3
10	Loss of power (blackout with natural circulation in RCS)	40	0	Note 3
11	Loss of Flow (partial loss of flow 1 pump only)	80	1	3
12	Reactor Trip from full power	400	49	140
13	Inadvertent auxiliary spray	10	0	Note 3
Faulted Conditions				
14	Main reactor coolant pipe break	1	Note 5	Note 5
15	Steam pipe break	1	Note 5	Note 5
16	Design-basis earthquake	1	Note 5	Note 5
Test Conditions				
17	Turbine roll test	10	Note 4	<10
Hydrostatic Test Conditions				
18	Primary side	5	Note 4	<5
19	Secondary side	5	Note 4	<5
20	Primary-side leak test	50	Note 6	<50

TABLE-4.3-1-4 NORTH ANNA UNIT 2 PROJECTED CYCLES AT 60 YEARS OF OPERATION (OCCURRENCES BASED ON DATA FROM INITIAL PLANT STARTUP THROUGH 5/18/01)

TRANSIENT NUMBER	TRANSIENT DESCRIPTION	UFSAR TABLE 5.2-4 DESIGN CYCLES	UNIT 2 CYCLES	UNIT 2 PROJECTED CYCLES

Notes:

1. Not counted. Basis for design cycle estimate is load follow operation. Unit does not operate in load follow mode. Instead, the unit is being operated in base-load mode.
2. Not counted. Infinite number of fluctuations assumed.
3. This transient is tracked. However, it has not occurred in the operational history of the units. No changes to operational philosophy are expected, therefore, this transient is not expected to occur during current license period or period of extended operation. Hence, this transient is projected to not exceed its design cycles limits.
4. Not counted. Test is an initial startup test only.
5. Not counted. Faulted condition occurs only once.
6. Not counted. Leak tests are performed at operating pressure. No system cyclic loading is involved.

TABLE 4.3-1-5. SURRY TRANSIENT CYCLE COUNTING - EVENT CRITERIA

Heatup	RCS temperature increase from Cold Shutdown to above 350°F.
Cooldown	RCS temperature decrease from Hot Shutdown to below 350°F.
Step Load Increase	Δ Load ≥ 5%, turbine online, 5% = ~40 MWe.
Step Load Decrease	5% ≤ Δ Load ≤ 15%, turbine online, 5% = ~40 MWe.
Large Load Reduction (Includes Turbine Runbacks)	Δ Load ≥ 15%, 15% = ~120 MWe, no Rx trip.
Reactor Trips	Reactor power > 25%

TABLE 4.3-1-6. NORTH ANNA TRANSIENT CYCLE COUNTING - EVENT CRITERIA

Heatup	RCS temperature increase from Cold Shutdown to above 550°F.
Cooldown	RCS temperature decrease from Hot Shutdown to below 200°F.
Step Load Increase	Δ Load = 10%, turbine online.
Step Load Decrease	Δ Load = 10%, turbine online.
Large Load Reduction (Includes Turbine Runbacks)	Δ Load ≥ 50%, no Rx trip.
Loss of Flow	Loss of 1, 2, or 3, RC pumps. Includes loss of AC.
Reactor Trips	Reactor power > 25%.

RAI 4.3-2:

As discussed in RAI 4.3.1-1, the applicant indicates that the existing design transients and cycle frequencies are conservative and bounding for the period of extended operation. However, the applicant also indicates that the North Anna reactor pressure vessel closure studs and reactor coolant systems (RCS) loop stop valves were reanalyzed. Explain why additional analyses were required for these components in light of the statement in the LRAs that design transients and frequencies are conservative and bounding for the period of extended operation.

Dominion Response:

The reactor pressure vessel (RPV) closure studs were originally analyzed for 57 events of tensioning and de-tensioning. Since tensioning and de-tensioning of the RPV head is a subset of heatups and cooldowns, it was decided to re-analyze the RPV closure studs for 200 tensioning and de-tensioning events. The re-analysis made the tensioning and de-tensioning cycles consistent with the heatup and cooldown cycles.

The reactor coolant system (RCS) loop stop valves were originally analyzed for one event of a steam generator tube rupture. Since there was a tube rupture event at NAPS, the loop stop valves were re-analyzed during the license renewal evaluation to upgrade the values for five (5) steam generator tube rupture events. The re-analysis has shown that the structural integrity of the loop stop valves is maintained with five steam generator tube rupture events.

RAI 4.3-3:

For both LRAs, identify whether calculations that meet the definition of a TLAA were performed in response to NRC Bulletin 88-08, "Thermal Stresses in Piping Connected to Reactor Coolant Systems." Describe the actions taken to address this bulletin during the period of extended operation.

Dominion Response:

No fatigue usage calculations that meet the definition of a TLAA were performed to address NRC Bulletin 88-08. No additional actions are intended, beyond the current licensing basis commitments, to address Bulletin 88-08 for the period of extended operation.

RAI 4.3-4:

The Westinghouse Owners Group issued Topical Report WCAP-14575-A, "Aging Management Evaluation for Class 1 Piping and Associated Pressure Boundary Components," to address aging management of the RCS piping. In both LRAs, Section 3.1.1, the applicant addresses the applicability of WCAP-14575-A to North Anna and Surry. Table 3.1.1-W1 of the LRAs contain the response to the renewal applicant action items developed as a result of the staff review of the topical report. Renewal Applicant Action Item 8 requests that applicants address components labeled I-M and I-RA in Tables 3-2 through 3-16 of WCAP-14575-A. The applicant indicates that the components in Tables 3-2 through 3-16 were addressed by an aging management activity, plant-specific fatigue evaluation, or code evaluation. However, the applicant did not provide specific details for each component. Provide a summary of the resolution for each of the components labeled I-M and I-RA in Tables 3-2 through 3-16.

Dominion Response:

The components labeled I-M and I-RA in Tables 3-2 through 3-16 are all piping components such as elbows, nozzles, straight pipe etc., which are Class 1 piping and associated pressure boundary components. These components are analyzed in accordance with the rules of B31.7 for NAPS and the rules of B31.1 for SPS, satisfying the requirements of the appropriate code.

RAI 4.3-5:

The Westinghouse Owners Group has issued the generic Topical Report WCAP-14574-A to address aging management of pressurizers. In both LRAs, Section 3.1.4, the applicant discusses the applicability of WCAP-14574-A to North Anna and Surry. In both LRAs, Table 3.1.4-W1, the applicant provides the response to the renewal applicant action items developed as a result of the staff review of the topical report. Renewal Applicant Action Item 1 requests that the applicant demonstrate that the pressurizer sub-component CUFs remain below 1.0 for the period of extended operation. Table 2-10 of WCAP-14574-A indicates that the ASME Section III Class 1 fatigue CUF criterion could be exceed at several pressurizer sub-component locations during the period of extended operation. WCAP-14574-A also identified recent unanticipated transients that were not considered in the original ASME Section III Class 1 fatigue analyses, including inflow/outflow thermal transients. The response to applicant action item 1 refers to the TLAA evaluation in Section 4.3 of the LRA. The discussion of the pressurizer surge line indicates that the inflow/outflow transients have been evaluated for the pressurizer components. Provide the following information:

a. Confirm that the additional transients discussed in WCAP-14574-A, not considered in the original design, have been addressed at North Anna and Surry.

b. Show the ASME Section III Class 1 CLB CUFs for the applicable sub-components of the North Anna and Surry pressurizers specified in Table 2-10 of WCAP-14574-A and the corresponding CUFs for the extended period of operation.

c. Discuss the impact of the environmental fatigue correlations provided in NUREG/CR-6583, "Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low-Alloy Steels," and NUREG/CR-5704, "Effects of LWR Coolant Environments on Fatigue on Fatigue Design Curves of Austenitic Stainless Steels," on the above results.

Dominion Response:

a. Plant-specific NAPS and SPS pressurizer analyses have been performed based on the recommendations made in WCAP-14574-A. As a result, plant-specific CUF values for all pressurizer locations are available which supersede those documented in WCAP-14574-A. The plant-specific NAPS and SPS analyses include the effects of all additional transients discussed in WCAP-14574-A that were not considered in the original design, including pressurizer insurge/outsurge and stratification effects. The CUF values calculated in the re-analyses are less than the allowable value of 1.0. These values are given in Table 4.3-5-1.

b. The re-analyses of the NAPS and SPS pressurizers account for the anticipated number of transients up to the end of the period of extended operation. The CUFs are below 1.0 through the period of extended operation.

   The ASME Section III Class 1 CLB CUFs for the applicable sub-components of the North Anna and Surry pressurizers, based on the plant-specific analysis discussed

under the response to the previous item, are shown in the table below through the period of extended operation. Section 4.3.1 of the LRA provides justification that the existing pressurizer component design cycles and cycle frequencies are conservative and bounding for the period of extended operation. As such, the CLB CUF values for the North Anna and Surry pressurizers provided below in Table 4.3-5-1 are considered to be conservative and bounding through the period of extended operation.

TABLE 4.3-5-1 CUF FOR PRESSURIZER SUBCOMPONENTS

Pressurizer Subcomponent	North Anna CLB CUF[4]	Surry CLB CUF[4]
Surge Nozzle + Corner + Nozzle-to-safe end weld + Safe end-to-pipe weld	 0.17 [1] 0.02 [1] 0.11 [1]	 0.29 [1] 0.30 [1] 0.05 [1]
Spray Nozzle[6]	0.848	0.848
Safety and Relief Nozzle[6]	0.148	0.148
Lower Head, Heater Well	0.82 [1]	0.82 [1]
Lower Head to Shell Weld	0.15 [1]	0.14 [1]
Upper Head and Shell	0.849 [2]	0.849 [2]
Support Skirt/Flange	0.0011	0.0011
Manway Pad	0.141 [3]	0.141 [3]
Manway Cover	0.0	0.0
Manway Bolts	0.875 [3]	0.875 [3]
Support Lug	0.048	0.048
Instrument Nozzle	0.13	0.1084
Immersion Heater	0.004	0.004
Valve Support Bracket	NA[5]	NA[5]

Notes:
1. CUF value reflects a more recent plant-specific analysis to incorporate the effects of pressurizer insurge/outsurge and stratification transients, as recommended by WCAP-14574-A.
2. Calculated fatigue usage factor is based on a conservative assumption that all spray transients will impinge directly on the pressurizer shell.
3. Plant-specific CUFs not determined; values from WCAP-14574-A are reported.
4. CUF values are presented for the wetted surface side which may be affected by EAF.
5. Not applicable since the safety valves are supported by belly bands.
6. Most limiting location is considered.

c. The impact of the environmental fatigue correlations provided in NUREG/CR-6583 and NUREG/CR-5704 are discussed below.

Screening

The effects of environmentally assisted fatigue (EAF) are a function of several parameters, including material type, temperature, and dissolved oxygen content. These effects on individual CUF load pairs can be potentially as high as a factor of fifteen for stainless steel when all relevant conditions are present. However, it is typical for the overall effects of all load pairs for a given component location to be significantly less, since environmental effects do not affect all individual load pairs (due to thresholds beyond which environmental effects are negligible).

Based on the environmental fatigue correlations provided in NUREG/CR-6583, "Effects of LWR Coolant Environments of Fatigue Design Curves of Carbon and Low-Alloy Steels," and NUREG/CR-5704, "Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels," bounding F_{en} multipliers have been determined for the pressurizer to establish a screening value for CUF. These are provided below for each material:

Low Alloy Steel

$F_{en} = \exp(0.929 - 0.00124T - 0.101S^*T^*O^*\epsilon^*)$

For a PWR environment, DO < 0.05, so $O^* = 0$.

Therefore, F_{en} is only dependent upon T.

T (°C)	F_{en}
0	2.53
50	2.38
100	2.24
150	2.10
200	1.98
250	1.86
300	1.75
Thus, maximum F_{en} = 2.53	

Carbon Steel

F_{en} = exp(0.585 - 0.00124T - 0.101S*T*O*ε*)

For a PWR environment, DO < 0.05, so O* = 0.

Therefore, F_{en} is only dependent upon T.

T (°C)	F_{en}
0	1.79
50	1.69
100	1.59
150	1.49
200	1.40
250	1.32
300	1.24
Thus, maximum F_{en} = 1.79	

Stainless Steel

F_{en} = exp(0.935 - T*ε*O*)

For a PWR environment, DO < 0.05, so O* = 0.260

T* = 0 for T < 200°C or T* = 1 for T > 200°C. Conservatively use T* = 1

Therefore, F_{en} is only dependent upon the strain rate parameter, ε*.

ε* = 0 for ε > 0.4%/sec so F_{en} = 2.55

ε* = ln(ε/0.4) for 0.0004 <= ε <= 0.4%/sec so F_{en} = 2.55 to 15.35

ε* = ln(0.0004/0.4) for ε < 0.0004%/sec so F_{en} = 15.35

Thus, maximum F_{en} = 15.35

Using a bounding F_{en} multiplier of 2.53 for carbon/low-alloy steel and 15.35 for stainless steel, the revised plant-specific CUF values for all pressurizer subcomponents, including maximum bounding EAF effects, are shown in Tables 4.3-5-2 and 4.3-5-3.

TABLE 4.3-5-2. CUF FOR NAPS
BASED ON BOUNDING F_{EN}

Pressurizer SUB-Component	Material (1)	Maximum F_{en} (5)	North Anna EAF CUF
Surge Nozzle (Inside Surface) + Corner + Nozzle-to-safe end weld + Safe end-to-pipe weld	 CS-LAS SS(3) SS	 2.53 15.35 15.35	 0.430 0.307 1.689(4)
Spray Nozzle(6)	LAS	2.53	2.145(4)
Safety and Relief Nozzle(6)	LAS	2.53	0.374
Lower Head, Heater Well	LAS, SS	15.35	12.59(4)
Lower Head-to-Shell Weld	LAS	2.53	0.380
Upper Head, Shell	LAS	2.53	2.148(4)
Support Skirt/Flange	CS	1.00 (2)	0.001
Manway Pad	LAS	2.53	0.357
Manway Cover	LAS	2.53	0.000
Manway Bolts	LAS	1.00 (2)	0.875
Support Lug	LAS	1.00 (2)	0.048
Instrument Nozzle	SS	15.35	1.996(4)
Immersion Heater	SS	15.35	0.0614
Valve Support Bracket	Safety valves are supported by a belly band.		

Notes:
1. LAS = low alloy steel, SS = stainless steel. CS = carbon steel
2. Location is not exposed to the water environment.
3. Inconel Buttering considered to be stainless steel for environmental effect on fatigue.
4. Acceptance of these locations is discussed later in Plant-Specific Evaluation.
5. Conservative F_{en} value used if more than one material involved.
6. Most limiting location is considered.

TABLE 4.3-5-3. CUF FOR SPS
BASED ON BOUNDING F_{EN}

Pressurizer SUB-Component	Material (1)	Maximum F_{en} (5)	SPS EAF CUF
Surge Nozzle (Inside Surface) + Corner + Nozzle-to-safe end weld + Safe end-to-pipe weld	 CS-LAS SS(3) SS	 2.53 15.35 15.35	 0.734 4.605(4) 0.768
Spray Nozzle(6)	CS	1.79	1.518(4)
Safety and Relief Nozzle(6)	CS	1.79	0.265
Lower Head, Heater Well	CS, SS	15.35	12.59(4)
Lower Head-to–shell Weld	LAS	2.53	0.354
Upper Head, Shell	CS, LAS	2.53	2.148(4)
Support Skirt/Flange	CS	1.00 (2)	0.001
Manway Pad	CS	1.79	0.253
Manway Cover	LAS	2.53	0.000
Manway Bolts	LAS	1.00 (2)	0.875
Support Lug	LAS	1.0(2)	0.048
Instrument Nozzle	SS	15.35	1.662(4)
Immersion Heater	SS	15.35	0.0614
Valve Support Bracket	Safety valves are supported by a belly band.		

Notes:
1. LAS = low-alloy steel, SS = stainless steel, CS = carbon steel.
2. Location is not exposed to the water environment.
3. Inconel Buttering considered to be stainless steel for environmental effect on fatigue.
4. Acceptance of these locations is discussed later in plant-specific evaluation.
5. Conservative F_{en} value used if more than one material is involved.
6. Most limiting location is considered.

Based on the values shown in the tables above, pressurizer sub-components have acceptable CUF values (i.e., less than the allowable value of 1.0) and are eliminated from further consideration, with the exception of the following:

- Surge Nozzle
- Spray Nozzle
- Lower Head, Heater Well
- Upper Head and Shell
- Instrument Nozzle

Note that the use of the bounding F_{en} multipliers developed above provide a very conservative basis for assessing the pressurizer with respect to the CUF allowable of 1.0, since actual F_{en} multipliers are likely to be significantly lower.

The spray nozzle, lower head heater well, upper head and shell, and instrument nozzle are addressed via a plant specific evaluation, as discussed below. The surge nozzle is addressed by aging management, also as discussed below.

Plant-Specific Evaluation

Spray Nozzles:

Based on a review of the North Anna and Surry pressurizer analysis, the total spray nozzle CUF of 0.848 is comprised primarily from the fatigue damage of four transient combinations. The four transients are (1) inadvertent auxiliary spray, (2) normal spray above the differential temperature limits allowed by plant procedures during heatup/cooldown, (3) normal spray during heatup/cooldown within differential temperature limits allowed by plant procedures, and (4) normal spray during plant loading and unloading at 5% per minute.

To have thorough mixing in the pressurizer and to prevent an uneven concentration of boron, at Surry and North Anna, the heaters and the pressurizer spray are kept continuously on. Since the pressurizer spray is continuously on, the above transients, which have been considered in the design analysis, are not expected to occur. As a result, the calculated fatigue value becomes almost negligible.

As a result, environmental effects on fatigue on pressurizer spray nozzles are insignificant.

Lower Head and Heater Well:

The heater penetrations in the lower head were not explicitly modeled, but were accounted for in the stress evaluations with appropriate stress intensification factors, which resulted in a very conservative and artificially high value of the CUF. If detailed finite element calculations are performed for this sub-component, the CUF will be reduced significantly. No additional actions beyond present aging management activities identified in LRA-Table 3.1.4-1 are planned for the heater wells for the period of extended operation for the following reasons:

- Inherent margins in the calculation process.
- The low risk significance associated with these penetrations.
- Current visual inspections performed on these penetrations as part of the ASME Section XI, Subsections IWB Inservice Inspection Program (as described in LRA Appendix B, Section B2.2.11).
- The fact that the surge line weld at the hot leg pipe connection is limiting from a fatigue perspective when considering reactor water environmental effects. Any observed effect of EAF in this surge line weld will result in evaluation of EAF for the pressurizer lower head and heater well location.

Upper Head and Shell:

With respect to the upper head and shell, the original North Anna and Surry

pressurizer analyses have conservatively assumed that cold spray flow impinged directly on the upper shell of the pressurizer during the controlling spray transient. The transient associated with this impingement has contributed to almost all of the CUF for the upper head and shell.

A study conducted by Westinghouse in 1989 established that water droplets from the pressurizer spray nozzle do not impinge on the pressurizer shell at a pressurizer pressure above 320 psig. In accordance with North Anna and Surry operating procedures, the pressurizer bubble is collapsed and the pressurizer is taken water solid at pressures between 325 and 350 psig. As such, the Westinghouse study is applicable to North Anna and Surry, and direct spray impingement does not occur in the North Anna or Surry pressurizers. Without direct impingement, the associated transient is eliminated, and the reported original CUF of 0.849 for the upper head and shell reduces to a negligible value and, therefore, CUF with EAF will be less than 1.0.

Instrument Nozzle:

No additional actions beyond present aging management activities identified in LRA-Table 3.1.4-1 are planned for the instrument nozzle for the period of extended operation for the following reasons:

- Inherent margins in the calculation process,
- The low risk significance associated with the nozzle (primarily due to its small size),
- Current visual inspections performed on the instrument nozzle as part of the ASME Section XI, Subsections IWB Inservice Inspection Program (as described in LRA Appendix B, Section B2.2.11),
- The fact that the surge line weld at the hot leg pipe connection is limiting from a fatigue perspective when considering reactor water environmental effects. Any observed effect of EAF in this surge line weld will result in evaluation of EAF for the pressurizer instrument nozzle.

As indicated in Section 4.3.4 of the application, Dominion will re-assess environmentally assisted fatigue based on the results of on-going industry activities currently underway by EPRI and the Materials Reliability Program, as well as continued field experience.

Aging Management

Surge Nozzle:

As discussed in the response to RAI 4.3-7, pressurizer surge line weld at the hot leg pipe connection will be inspected and used as the leading indicator for EAF concerns. This inspection commitment is documented in the application as a Licensee Follow-up Action Item in Section B2.2.1 and Table B4.0-1, and will be included in the UFSAR Supplement. The results of these inspections and the results

of planned research by the EPRI-sponsored Materials Reliability Program will be utilized to assess the appropriate approach for addressing the issue of environmentally assisted fatigue for applicable components including the surge nozzle.

Thus, EAF has been adequately addressed for all pressurizer subcomponents for the period of extended operation.

RAI 4.3-6:

In both LRAs, Section 4.3.4, the applicant discusses the impact of the reactor water environment on the fatigue life of components. The applicant references the fatigue sensitive component locations for an early vintage Westinghouse plant identified in NUREG/CR-6260, "Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components." The LRAs indicates that the results of the NUREG/CR-6260 studies were used to scale up the North Anna and Surry plant-specific usage factors for the same locations to account for environmental effects. The LRAs also indicates that the later environmental fatigue correlations contained in NUREG/CR-6583, "Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low-Alloy Steels," and NUREG/CR-5704, "Effects of LWR Coolant Environments on Fatigue on Fatigue Design Curves of Austenitic Stainless Steels," were considered in the evaluation. Provide the results of the usage factor evaluation for each of the six component locations listed in NUREG/CR-6260. Discuss how the factors used to scale up the North Anna and Surry plant-specific usage factors were derived. Also discuss how the later environmental data provided in NUREG/CR-6583 and NUREG/CR-5704 were factored in the evaluations. Discuss the how the North Anna charging line flow transients monitored by the TCCP are factored in these evaluations.

Dominion Response:

Section 4.3.4 of the LRA describes Dominion's evaluation of the impact of the reactor water environment on the fatigue life of the components identified in NUREG/CR-6260, "Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components." In particular, that evaluation relies on several industry background studies that have been performed to address EAF effects in RCS components. Those studies have been used to provide an assessment of NAPS and SPS environmental effects on the locations identified in NUREG/CR-6260, using a scaling factor approach. These locations are identified in Table 4.3-6-1.

TABLE 4.3-6-1. OLDER VINTAGE WESTINGHOUSE PLANT LOCATIONS IDENTIFIED IN NUREG 6260

Reactor vessel	At core support guide weld
	Inlet Nozzle
	Outlet Nozzle
Surge line	Hot leg nozzle safe end
Charging nozzle	Nozzle
Safety injection nozzle	Nozzle
Residual heat removal line	Tee

Because of the more recent issues raised by the NRC staff relative to the use of the EPRI/GE F_{en} methodology (Reference EPRI Report No. TR-105759, "An Environmental Factor Approach to Account for Reactor Water Effects in Light Water Reactor Pressure Vessel and Piping Fatigue Evaluations") in various industry applications as well as additional laboratory fatigue data in simulated LWR environments that have been generated by Argonne National Laboratory (ANL) for carbon, low-alloy, and stainless steels, calculations have been revised since the original submittal of the LRA, for the seven locations identified in NUREG/CR-6260 for NAPS and SPS. These calculations are summarized below, and utilize the most recent F_{en} methodology, as published in NUREG/CR-6583 and NUREG/CR-5704. The conclusions identified in the LRA for NAPS and SPS, which were reached based on the original calculations have been validated by the revised calculations.

RPV Locations

The environmental fatigue calculations for the three RPV components identified in NUREG/CR-6260 (RPV shell at core support pads, RPV inlet nozzle, and RPV outlet nozzle) are shown in Table 4.3-6-2. The results show EAF-adjusted CUF values for these three locations of less than 1.0, which are acceptable. The results shown in Table 4.3-6-2 are very conservative, in that the maximum bounding F_{en} multiplier was conservatively used.

TABLE 4.3-6-2. PLANT-SPECIFIC EAF EVALUATION FOR RPV LOCATIONS AND SURGE LINE

Location	Original Design Basis CUFs		Maximum F_{en}	Environmental CUFs	
	U_{NAPS}	U_{SPS}		U_{NAPS}	U_{SPS}
RPV Shell at Core Support Pads	0.092	0.01	2.53	0.233	0.025
RPV Inlet Nozzle	0.022	0.011	2.53	0.056	0.028
RPV Outlet Nozzle	0.074	0.256	2.53	0.187	0.648
Surge Line Hot Leg Nozzle	0.966	0.861	See Note 1	N/A	N/A

Note: 1. Inspection aging management will be used for this location.

TABLE 4.3-6-2. PLANT-SPECIFIC EAF EVALUATION FOR RPV LOCATIONS AND SURGE LINE (CONT.)

Low Alloy Steel

$F_{en} = \exp(0.929 - 0.00124T - 0.101S^*T^*O^*\varepsilon^*)$

For a PWR environment, DO < 0.05, so O* = 0.

Therefore, F_{en} is only dependent upon T.

T (°C)	F_{en}
0	2.53
50	2.38
100	2.24
150	2.10
200	1.98
250	1.86
300	1.75
Thus, maximum F_{en} = 2.53	

Carbon Steel

$F_{en} = \exp(0.585 - 0.00124T - 0.101S^*T^*O^*\varepsilon^*)$

For a PWR environment, DO < 0.05, so O* = 0.

Therefore, F_{en} is only dependent upon T.

T (°C)	F_{en}
0	1.79
50	1.69
100	1.59
150	1.49
200	1.40
250	1.32
300	1.24
Thus, maximum F_{en} = 1.79	

Stainless Steel

$F_{en} = \exp(0.935 - T^*\varepsilon^*O^*)$

For a PWR environment, DO < 0.05, so O* = 0.260

T* = 0 for T < 200°C or T* = 1 for T > 200°C. Conservatively use T* = 1

Therefore, F_{en} is only dependent upon the strain rate parameter, ε*.

$\varepsilon^* = 0$ for $\varepsilon > 0.4\%$/sec so $F_{en} = 2.55$

$\varepsilon^* = \ln(\varepsilon/0.4)$ for $0.0004 <= \varepsilon <= 0.4\%$/sec so $F_{en} = 2.55$ to 15.35

$\varepsilon^* = \ln(0.0004/0.4)$ for $\varepsilon < 0.0004\%$/sec so $F_{en} = 15.35$

Thus, maximum $F_{en} = 15.35$

Charging Nozzle Location

For the charging nozzle, CUF results exist only for NAPS, because the design basis for the SPS piping is USAS B31.1, which does not require explicit fatigue analysis. However, the detailed plant-specific charging nozzle fatigue calculations for NAPS are not readily retrievable. Therefore, NAPS plant-specific fatigue calculations have been reconstituted based on the inputs used in NUREG/CR-6260. A detailed EAF evaluation for this location was subsequently performed.

The environmental fatigue charging nozzle calculations are shown in Table 4.3-6-3. The results show an EAF adjusted CUF value of less than 1.0, which is acceptable.

Since the SPS design code for the charging piping is USAS B31.1, no explicit fatigue analysis has been performed. However, since the physical attributes of the SPS piping, nozzles, and transient characteristics are similar to those at NAPS, it is concluded that the SPS charging nozzles are likewise acceptable with consideration of EAF.

TABLE 4.3-6-3. PLANT-SPECIFIC EAF EVALUATION FOR CHARGING NOZZLE

Step #1: Reproduce NUREG/CR-6260 Calculations from Table 5-90
Note 1: $S_{alt} = S_{alt\text{-}NB\text{-}3600}$ since this is the limiting NB-3600 CUF location.

Branch connection/nozzle body	S_{alt}	N_{allow}	n	U	**Comments**
(NB-3600)	363.53	44	20	0.452	
	46.00	51,814	80	0.002	Note 2
	46.00	51,814	120	0.002	Note 2
			Total = 0.456		

Note 2: Small difference from NUREG/CR-6260; neglected.

Nozzle-to-pipe weld	S_{alt}	N_{allow}	n	U	
(NB-3600)	84.62	3,340	20	0.006	
	70.04	6,951	80	0.012	
	52.11	27,505	120	0.004	
			Total = 0.022		Note 3

Note 3: $S_{alt} = S_{alt\text{-}NB\text{-}3200}$. Although this is not the limiting NB-3200 CUF location (by only a very small amount), the strain rate is lower than for the nozzle-to-pipe weld location, so this location becomes limiting when environmental effects are considered.

TABLE 4.3-6-3. PLANT-SPECIFIC EAF EVALUATION FOR CHARGING NOZZLE (CONT.)					
Branch connection/nozzle body	S_{alt}	N_{allow}	n	U	
(NB-3200)	87.69	2,922	20	0.007	
	80.94	3,947	80	0.020	
	29.47	724,304	120	0.000	
			Total = 0.027		Note 4
Note 4: Limiting NB-3200 location; see note 3 above.					
Nozzle region upstream of thermal sleeve	S_{alt}	N_{allow}	n	U	Note 5
(NB-3200)	84.79	3,314	20	0.006	
	82.86	3,614	80	0.022	
	46.15	50,897	120	0.002	
			Total = 0.031		
Conclusions: **1. NUREG/CR-6260 calculations are reproduced.** **2. Appropriate values of "n" are obtained.**					
Step #2: Reproduce NAPS CUF by scaling up S_{alt} for Limiting NB-3600 Location					
Note 5: $S_{alt} = S_{alt\text{-}NB\text{-}3600\text{-}NAPS}$					
	Multiplier for S_{alt} = 1.313				
Branch connection/nozzle body:	S_{alt}	N_{allow}	n	U	
(NB-3600)	477.31	24	20	0.850	
	60.40	13,253	80	0.006	
	60.40	13,253	120	0.009	
			Total = 0.8647		Note 6
Note 6: The difference is insignificant between this value and NAPS design basis CUF of 0.8646					
Conclusion: 1. Above calculation is a reconstituted calculation for NAPS.					

TABLE 4.3-6-3. PLANT-SPECIFIC EAF EVALUATION FOR CHARGING NOZZLE (CONT.)					
Step #3: Create a NAPS NB-3200 CUF from the NB-3600 Calculation					
Note 7: $S_{alt} = S_{alt\text{-}NB\text{-}3600\text{-}NAPS} * (S_{alt\text{-}NB\text{-}3200}/S_{alt\text{-}NB\text{-}3600})$					
Branch connection/nozzle body:	**S_{alt}**	**N_{allow}**	**n**	**U**	Note 7
(NB-3200)	115.14	1,118	20	0.018	
	106.27	1,468	80	0.055	
	38.69	133,333	120	0.001	
			Total = 0.0733		Note 8
Note 8: Predicted $U_{NB\text{-}3200}$ for NAPS					
Conclusion: 1. Above calculation is reconstituted for NAPS.					
Step #4: Determine F_{en} Multiplier and Environmental CUF					
F_{en} for Stainless Steel:	**F_{en} = exp(0.935 - T*ε*O*)**				
For a PWR environment, DO < 0.05, so O* = 0.260 T* = 0 for T < 200°C or = 1 for T > 200°C. Conservatively use T* = 1 ε = 0.08%/sec per strain rate calculation, so ε* = ln(ε/0.4) for 0.0004 < ε < 0.4 %/s. Calculated F_{en} = 3.87 Environmental CUF for NAPS = F_{en} * $U_{NB\text{-}3200}$ for NAPS = 0.284					

Safety Injection Nozzle Location

For the safety injection nozzle, CUF results exist only for NAPS, because the design basis for the SPS piping is USAS B31.1, which does not require explicit fatigue analysis. However, the detailed plant-specific safety injection nozzle fatigue calculations for NAPS are not readily retrievable. Therefore, plant-specific fatigue calculations have been reconstituted for NAPS based on the inputs used in NUREG/CR-6260. A detailed EAF evaluation was subsequently performed for this location.

The environmental fatigue calculations for the safety injection nozzles are shown in Table 4.3-6-4. The results show an EAF adjusted CUF value of less than 1.0, which is acceptable.

Since the design code for the safety injection piping is USAS B31.1 for SPS, no explicit fatigue analysis has been performed. However, since the physical attributes of the SPS piping, nozzles, and transient characteristics are similar to those at NAPS, it is

concluded that the SPS Safety Injection nozzles are likewise acceptable with consideration of EAF.

TABLE 4.3-6-4. PLANT-SPECIFIC EAF EVALUATION FOR SAFETY INJECTION NOZZLE

Step #1: Reproduce NUREG/CR-6260 Calculations from Tables 5-93 and 5-94					
Note 1: $S_{alt} = S_{alt\text{-}NB\text{-}3600}$ since this is the limiting NB-3600 CUF location.					
Branch connection/nozzle body	$\mathbf{S_{alt}}$	$\mathbf{N_{allow}}$	**n**	**U**	**Note 1**
(NB-3600)	400.22	35	70	1.976	
					Note 2
			Total = 1.976		
Note 2:The S_{alt} values for the two other load pairs are unknown; however, the CUF matches so their contribution is negligible.					
Nozzle-to-pipe weld	$\mathbf{S_{alt}}$	$\mathbf{N_{allow}}$	**N**	**U**	
(NB-3600)	102.57	1,655	70	0.042	
	46.79	47,408	50	0.001	Note 3
	45.49	55,079	150	0.003	Note 3
			Total = 0.046		
Note 3: S_{alt} and n values obtained from Table 5-94.					
Branch connection/nozzle body	$\mathbf{S_{alt}}$	$\mathbf{N_{allow}}$	**N**	**U**	
(NB-3200)	32.88	346,189	70	0.000	
				0.002	Note 4
			Total = 0.002		
Note 4: The S_{alt} values for the two other load pairs are unknown; this equivalent incremental CUF is chosen so the total CUF results agree.					
Nozzle region upstream of thermal sleeve	S_{alt}	N_{allow}	N	U	
(NB-3200)	92.48	2,393	70	0.029	
				0.002	Note 5
			Total = 0.031		Note 6
Note 5: The S_{alt} values for the two other load pairs are unknown; this equivalent incremental CUF is chosen so the total CUF results agree.					
Note 6: $S_{alt} = S_{alt\text{-}NB\text{-}3200}$ since this is the limiting NB-3200 CUF location.					

TABLE 4.3-6-4. PLANT-SPECIFIC EAF EVALUATION FOR SAFETY INJECTION NOZZLE (CONT.)

Nozzle-to-pipe weld	S_{alt}	N_{allow}	N	U	
(NB-3200)	125.14	852	70	0.082	
				0.013	Note 7
			Total = 0.095		

Note 7: The S_{alt} values for the two other load pairs are unknown; this equivalent incremental CUF is chosen so the total CUF results agree.

Conclusions:

1. NUREG/CR-6260 calculations are reproduced.
2. Appropriate values of "n" are obtained.

Step #2: Reproduce NAPS CUF by scaling up S_{alt} for Limiting NB-3600 Location

	multiplier for S_{alt} = 0.669		

Note 8: $S_{alt} = S_{alt\text{-}NB\text{-}3600\text{-}NAPS}$

Branch connection/nozzle body:	S_{alt}	N_{allow}	n	U	Note 8
(NB-3600)	267.75	94	70	0.746	
			Total = 0.746		Note 9

Note 9: Matches NAPS design basis CUF of 0.746.= $U_{NB\text{-}3600}$ for NAPS

Conclusion:

1. Above calculation is a reconstituted for NAPS.

Step #3: Create a NAPS NB-3200 CUF from the NB-3600 Calculation

Note 10: $S_{alt} = S_{alt\text{-}NB\text{-}3600\text{-}NAPS} * (S_{alt\text{-}NB\text{-}3200}/S_{alt\text{-}NB\text{-}3600})$

Branch connection/nozzle body:	S_{alt}	N_{allow}	n	U	Note 10
(NB-3200)	83.72	3,477	70	0.020	
				0.013	Note 11
			Total = 0.033		Note 12

Note 11: Conservatively use this incremental CUF.

Note 12: Predicted $U_{NB\text{-}3200}$ for NAPS

Conclusion:

1. Above calculation is a reconstituted for NAPS.

TABLE 4.3-6-4. PLANT-SPECIFIC EAF EVALUATION FOR SAFETY INJECTION NOZZLE (CONT.)
Step #4: Determine F_{en} Multiplier and Environmental CUF:
F_{en} for Stainless Steel: $F_{en} = exp(0.935 - T^*\varepsilon^*O^*)$
For a PWR environment, DO < 0.05, so $O^* = 0.260$ $T^* = 0$ for $T < 200°C$ or $= 1$ for $T > 200°C$. Conservatively use $T^* = 1$ $\varepsilon = 1.23\%/sec$ per strain rate calculation, so $\varepsilon^* = 0$ for $\varepsilon > 0.4\ \%/s$. Calculated $F_{en} = 2.55$ Environmental CUF for NAPS = 0.084 = $F_{en} * U_{NB-3200}$ for NAPS

RHR Tee Location

Detailed plant-specific fatigue calculations are available for the NAPS RHR tee. Therefore, appropriate detailed F_{en} factors have been calculated for this location to apply to the individual fatigue contributing load pairs and a resulting EAF-adjusted CUF value has been determined.

The environmental fatigue calculations for the RHR tee shown in Table 4.3-6-5 are based on NAPS plant-specific input. The results show an EAF adjusted CUF value of less than 1.0, which is acceptable. Since the design code for the SPS RHR piping is USAS B31.1, no explicit fatigue analysis has been performed. Transient stresses are expected to be similar for NAPS and SPS, since the geometry and material are similar for all four units. Therefore, the results are considered to apply to SPS as well.

TABLE 4.3-6-5. PLANT-SPECIFIC EAF EVALUATION FOR RHR TEE

Step #1: Reproduce design basis fatigue calculation

Note 1: NUREG-6260 calculations are not required for this location since a plant-specific calculation exists.

Tee	S_{alt}	N_{allow}	n	U	
(NB-3600)	126.161	657	200	0.304	
			Total = 0.304		

Conclusion:

1. Design basis fatigue calculation is adequately reproduced.

Step #2: Determine F_{en} Multiplier and Environmental CUF:

F_{en} for Stainless Steel:

$$F_{en} = \exp(0.935 - T^*\varepsilon^*O^*)$$

For a PWR environment, DO < 0.05, so $O^* = 0.260$.

$T^* = 0$ for $T < 200°C$ or $= 1$ for $T > 200°C$.
All temperatures for the controlling RHR transient are less than 200°C (392°F), so $T^* = 0$.

ε^* does not matter when $T^* = 0$.

Calculated $F_{en} = 2.55$

Environmental CUF for NAPS = 0.775

Surge Line/Nozzle Location

For the surge line, which is a high CUF location, an aging management program that includes inspection has already been planned to satisfy EAF considerations, as discussed in Sections 4.3.4 and B4.0 of the LRA. Therefore, additional EAF evaluation for this location has not been performed.

Summary of Results for EAF Evaluation

The EAF results for all NAPS/SPS locations evaluated above are summarized in Table 4.3-6-6. The results demonstrate that the CUFs for all locations, including postulated environmental effects, remain within the allowable value of 1.0 for 60 years.

TABLE 4.3-6-6. SUMMARY OF NAPS/SPS ENVIRONMENTAL FATIGUE CALCULATIONS

No.	Component	Maximum Design CUF	Multiplier	Environmental CUF
1	RPV Shell at Core Support Pads	0.092	2.53	0.233
2	RPV Inlet Nozzle	0.022	2.53	0.056
3	RPV Outlet Nozzle	0.256	2.53	0.648
4	Charging Nozzle	0.073	3.87	0.283
5	Safety Injection Nozzle	0.033	2.55	0.084
6	RHR Tee	0.304	2.55	0.775

Charging Line Flow Transients

Temperature data (as an indication of flow transients) from the existing plant instrumentation is being collected so that a comparison of operating and design transients for the charging nozzles can be made to validate the design transients. These data are planned to be used only for validating the design transients. They are not included in the above evaluations.

RAI 4.3-7:

In both LRAs, Section 4.3.4, the applicant indicates that the pressurizer surge line required further evaluation for environmental fatigue during the period of extended operation. The applicant further indicates that it would use an aging management program to address fatigue of the surge line during the period of extended operation. The aging management program would rely on an augmented inspection program to address surge line fatigue during the period of extended operation. As indicated in the draft safety evaluation on Westinghouse Owners Group generic technical report WCAP -14575, License Renewal Evaluation: Aging Management Evaluation for Class 1 Piping and Associated Pressure Boundary Components," the NRC has not endorsed a procedure on a generic basis which allows for augmented inspections in lieu of meeting the fatigue usage criteria. The applicant has not provided a technical basis demonstrating the technical adequacy of its proposal. Provide a detailed technical evaluation which demonstrates the proposed inspections provide an adequate technical basis for detecting fatigue cracking before such cracking leads to through wall cracking or pipe failure. The detailed technical evaluation should be sufficiently conservative to address all uncertainties associated with the technical evaluation (e.g., fatigue crack initiation and detection, fatigue crack size, and fatigue crack growth rate considering environmental factors). As an alternative to the detailed technical evaluation, provide a commitment monitor the fatigue usage, including environmental effects, during the period of extended operation, and to take corrective actions, as approved by the staff, if the usage is projected to exceed one.

Dominion Response:

The proposed aging management program to address environmentally assisted fatigue of the SPS and NAPS pressurizer surge lines during the period of extended operation is to inspect in accordance with the appropriate requirements of ASME Boiler and Pressure Vessel Code, Section XI, Rules for Inservice Inspection of Nuclear Power Plant Components. Inspection frequency will be supported by methods similar to the approach documented in the ASME Boiler and Pressure Vessel Code, Section XI, Non-mandatory Appendix L. However, Dominion recognizes that to date, the NRC has not endorsed the Appendix L approach. The primary NRC concerns with Appendix L include crack aspect ratio and acceptable fatigue crack growth rates (including environmental effects).

Some of the pressurizer surge line welds at SPS and NAPS have been ultrasonically examined in the past. No reportable indications have been found. Section 4.3.4 of the application states that the surge line weld at the hot leg pipe connection will be included in an augmented inspection program, so that flaw initiation and growth can be detected and/or monitored. Baseline inspections of these welds are planned prior to entry into the period of extended operation, and they will also be inspected once per period. The results of these inspections and the results of planned research by the EPRI-sponsored Materials Reliability Program will be utilized to assess the appropriate approach for addressing environmentally assisted fatigue of the surge lines. The approach developed could include one or more of the following:

1. Further refinement of the fatigue analysis to lower the CUF(s) to below 1.0, or
2. Repair of the affected locations, or
3. Replacement of the affected locations, or
4. Manage the effects of fatigue by an inspection program that has been reviewed and approved by the NRC (e.g., periodic non-destructive examination of the affected locations at inspection intervals to be determined by a method accepted by the NRC).

Should Dominion select Option 4 (i.e., inspection) to manage environmentally assisted fatigue during the period of extended operation, inspection details such as scope, qualification, method, and frequency will be provided to the NRC for review prior to entering the period of extended operation. This position is consistent with previous applicants' positions. Licensee Follow-up Action for surge line inspection has been identified in LRA Table B4.0-1 and will be included in the UFSAR Supplement.

Section 4.7.4, "Spent Fuel Pool Liner"

RAI 4.7.4-1:

Please provide a tabulated summary of the number of cycles considered in the fatigue analysis for normal, upset, emergency, and faulted conditions together with the temperature ranges considered for each condition.

Dominion Response:

CONDITIONS	DESCRIPTION	DESIGN CYCLES	TEMPERATURE RANGE
Condition 1 (Normal)	1/3 Core Initial Load	1	70°F – 121°F
Condition 2 (Normal)	1/3 Core Refuel with 10 years fuel in the pool	80	70°F – 135°F
Condition 3 Upset	1 core offload – 45 days after refueling abnormal condition	8	70°F – 170°F
Condition 4 Faulted	Faulted condition	1	70°F – 212°F

The above cycles envelop the operating cycles for 60 years. These cycles are applicable to both SPS and NAPS.

RAI 4.7.4-2:

What is the temperature range considered in calculating the allowable thermal cycles for the most severe thermal cycles?

Dominion Response:

The NAPS and SPS analyses consider the range of 70°F – 212°F for the most severe thermal cycles.

RAI 4.7.4-3:

As the stainless pool liner is attached to the concrete walls and the bottom slab (or basemat), the fatigue characteristics of the liner will be influenced by the integrity of its anchorages to the concrete, and the effects of high sustained (> 15 days) temperature on the concrete. Please provide a summary of procedures used to incorporate these effects in the pool liner time-limited fatigue analysis.

Dominion Response:

The NAPS pool liner was designed in accordance with the design criteria provided in the ASME Boiler and Pressure Vessel Code, Section III, Division 1 – 1974 edition; Nuclear Power Plant Components, Subsection NA (with addenda up to Summer 1976). The SPS pool liner was designed in accordance with the design criteria provided in the ASME Boiler and Pressure Vessel Code, Section III, Division 1 – 1971 edition. The following procedure was used in the existing calculations to qualify the fuel pool liner.

Procedure:

1. Membrane plus bending stresses caused by differential thermal expansion was calculated using linear elastic methods of analysis.

2. If one of the following two conditions were met, the liner was considered to be acceptable.

   - If stresses calculated at points, which are not welds or points of stress concentrations, the calculated stresses should be less than $3S_m$.

   - If stresses calculated at points, which are welds or points of stress concentrations, the calculated stresses should be less than $0.75S_m$. (Note: The limit of $0.75S_m$ results in a stress concentration factor of 4.0.)

3. If the calculated stresses exceed either $3S_m$ or $0.75S_m$ as identified in 2, then the liner is evaluated on the basis of fatigue life. The liner integrity was assessed in accordance with the cyclic loading design procedure, Paragraph XIV- 1221.3, Pages 336 and 337 of Section III. Stresses to determine the fatigue life was calculated as follows:

   - Stress concentration factor of 1.0 was used for points, which are not welds or points of stress concentrations.

   - Stress concentration factor of 4.0 was used for points, which are welds or points of stress concentrations.

4. Appropriate design stress intensity values from Section III were used.

5. The applicable design fatigue curve from Section III was used.

For the locations with welds or stress concentrations, a factor of 4 has been used. The most limiting condition is that the concrete structure is deformed to its maximum

permissible limit and is at room temperature. The liner design calculations show that the allowable cycles for the liner reaching 212°F are 100 for North Anna and 95 for Surry. This number of allowable cycles exceeds the total number of expected operating cycles (90) as identified in Section 4.7.4 of the application. Furthermore, the temperature of the fuel pool is expected to be below 135°F under normal conditions. Since the operating temperature is low, effects of sustained high temperature is not a concern.

The welding at the Surry anchorages has been analyzed for fatigue usage factor using the operating cycles listed above. It was found to be 0.578, which is less than the allowable value of 1.0. No additional analysis has been performed for North Anna, since the SPS liner is the most limiting.