Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Rope Ferry Road
Waterford, CT 06385

Dominion

February 8, 2005

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

Serial No.:	05-047
LR/RJG	R0
Docket Nos.:	50-336
	50-423
License Nos.:	DPR-65
	NPF-49

DOMINION NUCLEAR CONNECTICUT, INC. (DNC)
MILLSTONE POWER STATION UNITS 2 AND 3
ADDITIONAL INFORMATION IN SUPPORT OF
LICENSE RENEWAL APPLICATIONS

The Nuclear Regulatory Commission (NRC) has requested additional information as a result of audits of the Aging Management Programs and Aging Management Reviews. This additional information in support of the Millstone Power Station Units 2 and 3 LRAs is being submitted as Attachment 1. Also, on January 24, 2005, the NRC requested supplemental information pertaining to previous Requests for Additional Information responses. Dominion's response to those items is provided as Attachment 2.

Should you have any questions regarding this letter, please contact Mr. William D. Corbin, Director, Nuclear Projects, Dominion Resources Services, Inc., at (804) 273-2365.

Very truly yours,

Leslie N. Hartz
Vice President – Nuclear Engineering

Attachments:

1. Additional Information in Support of Applications for Renewed Operating Licenses
2. Supplemental Information to Previous Request for Additional Information Responses

A106

Commitments made in this letter:

This letter identifies License Renewal Commitments to be added to Table A6.0-1 of the Final Safety Analysis Report (FSAR) Supplement and is proposed to support approval of the renewed operating licenses. These commitments may change during the NRC review period. A revised FSAR Supplement which contains these commitments will be submitted to the staff as input to the Millstone License Renewal Safety Evaluation Report.

cc:

U.S. Nuclear Regulatory
Commission
Region I
475 Allendale Road
King of Prussia, PA 19406-1415

Mr. V. Nerses
Senior Project Manager
U.S. Nuclear Regulatory
Commission
One White Flint North
11555 Rockville Pike
Mail Stop 8C2
Rockville, MD 20852-2738

Mr. J. H. Eads
Senior Project Manager
U.S. Nuclear Regulatory
Commission
One White Flint North
11555 Rockville Pike
Mail Stop 11F1
Rockville, MD 20852-2738

Mr. S. M. Schneider
NRC Senior Resident Inspector
Millstone Power Station

Honorable Wayne L. Fraser
First Selectman
P.O. Box 519
Niantic, CT 06357-0519

Mr. Stephen Page
Central VT PSC
77 Grove Street
Rutland, VT 06701

Honorable Andrea Stillman
CT State Senate
5 Coolidge Court
Waterford, CT 06385

Mr. Denny Galloway
Supervising Radiation Control
Physicist
State of Connecticut – DEP
Division of Radiation
79 Elm Street
Hartford, CT 06106-5127

Honorable Christopher Dodd
US Senate
100 Great Meadow Road
Wethersfield, CT 06109

Ms. Catherine Marx
Governor's Eastern Office
P.O. Box 1007
171 Salem Turnpike
Norwich, CT 06360-1007

Mr. William Meinert
MMWEC
P.O. Box 426
Ludlow, MA 01056-0426

Honorable Elizabeth Ritter
CT House of Representatives
24 Old Mill Road
Quaker Hill, CT 06375

Honorable Ed Jutila
CT House of Representatives
23 Brainard Road
Niantic, CT 06357

Honorable Robert Simmons
US Congress
2 Courthouse Square
Norwich, CT 06360

Mr. Thomas Wagner
Town of Waterford
Town Planner
15 Rope Ferry Road
Waterford, CT 06385

Dr. Edward L. Wilds
Director, Division of Radiation
State of Connecticut – DEP
79 Elm Street
Hartford, CT 06106-5127

Chairman Donald Downes
DPUC
10 Franklin Square
New Britain, CT 06051

Honorable Dennis L. Popp
Chairman – Council of
Governments
Municipal Building
295 Meridian Street
Groton, CT 06340

Chief Murray J. Pendleton
Director of Emergency
Management
41 Avery Lane
Waterford, CT 06385-2806

Honorable Richard Blumenthal
Attorney General
55 Elm Street
Hartford, CT 06106-1774

Mr. John Markowicz
Co-Chairman – NEAC
9 Susan Terrace
Waterford, CT 06385

Mr. Evan Woolacott
Co-Chairman – NEAC
128 Terry's Plain Road
Simsbury, CT 06070

Honorable M. Jodi Rell
Governor
State Capitol
Hartford, CT 06106

Mr. Mark Powers
4 Round Rock Road
Niantic, CT 06357

Mr. Jay Levin
23 Worthington Road
New London, CT 06320

Mr. Jim Butler
Executive Director – Council of
Governments
8 Connecticut Avenue
Norwich, CT 06360

Mr. Bill Palomba
Executive Director, DPUC
10 Franklin Square
New Britain, CT 06051

Honorable Ernest Hewett
CT House of Representatives
29 Colman Street
New London, CT 06320

Honorable Terry Backer
CT House of Representatives
Legislative Office Building
Room 2102
Hartford, CT 06106

Honorable Kevin DelGobbo
CT House of Representatives
83 Meadow Street
Naugautuck, CT 06770

Honorable Thomas Herlihy
CT Senate
12 Riverwalk
Simsbury, CT 06089

Honorable Cathy Cook
CT Senate, 18th District
43 Pequot Avenue
Mystic, CT 06355

Mr. Edward Mann
Office of Senator Dodd
Putnam Park
100 Great Meadow Road
Wethersfield, CT 06109

Chairperson Pam Katz
CT Siting Council
10 Franklin Square
New Britain, CT 06051

Mr. Ken Decko
CBIA
350 Church Street
Hartford, CT 06103

Honorable Paul Eccard
First Selectman
Town of Waterford
15 Rope Ferry Road
Waterford, CT 06385

Mr. Richard Brown
City Manager
New London City Hall
181 State Street
New London, CT 06320

Honorable Jane G. Glover
Mayor, New London
New London City Hall
181 State Street
New London, CT 06320

SN: 05-047
Docket Nos.: 50-336/423
Subject: Additional Information in Support of License Renewal Application

COMMONWEALTH OF VIRGINIA)
)
COUNTY OF HENRICO)

The foregoing document was acknowledged before me, in and for the County and Commonwealth aforesaid, today by Leslie N. Hartz, who is Vice President - Nuclear Engineering, of Dominion Nuclear Connecticut, Inc. She has affirmed before me that she 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 her knowledge and belief.

Acknowledged before me this 8TH day of February, 2005.

My Commission Expires: May 31, 2006.

Vicki L. Hull
Notary Public

(SEAL)

Attachment 1

Additional Information in Support of Applications for Renewed Operating Licenses

Millstone Power Station Units 2 & 3

In a conference call on October 12, 2004, the staff made the following request associated with Section 4.3 of the Millstone Unit 2 LRA:

Question:

For late model CE plants with low alloy steel charging and SI nozzles, the CUFs are high in the nozzle safe ends. The Unit 2 LRA indicates the high CUFs are in the low alloy nozzles. Verify that high CUFs are at the nozzles and not the safe ends.

Dominion Response:

Table 1 summarizes the 60-year fatigue usage factors for the Millstone Unit 2 charging nozzle, charging nozzle safe end, safety injection nozzle and the safety injection nozzle safe end. For the charging nozzle and charging nozzle safe end, the Cumulative Usage Factor (CUF) values are based on the projected number of transients expected during the 60-year period of operation. For the safety injection nozzle and safety injection nozzle safe end, the projected 60-year cycles are bounded by the 40-year design cycles. Since the 40-year design cycles are bounding, the safety injection nozzle and the safety injection nozzle safe end CUF values were conservatively developed using the 40-year design cycles. This table lists CUF for the nozzles and safe ends with and without consideration of Environmentally Assisted Fatigue (EAF). Worst-case environmental factor, F_{en}, values are used in development of the CUF_{EAF} values.

As shown in Table 1, the charging and safety injection nozzles have the highest CUF without consideration of EAF. When considering EAF, the safety injection nozzle still has a higher CUF_{EAF} than the safety injection nozzle safe end; however, the charging nozzle safe end has a higher CUF_{EAF} than the charging nozzle.

Table 1
Millstone Unit 2
Projected 60-Year CUF/CUF_{EAF} Values

Location	Material	CUF	F_{en}	CUF_{EAF}
Charging Nozzle	A-105 Grade 2	0.1499	2.53	0.3796
Charging Nozzle Safe End	A-182 Type 316	0.0618	15.35	0.949
Safety Injection Nozzle	SA-182 Grade F1	0.1660	2.53	0.4204
Safety Injection Nozzle Safe End	A-351 CF8M	0.0197	15.35	0.3024

In an E-mail from the NRC to Dominion, dated January 5, 2005, two clarifying questions were asked, associated the Millstone Unit 2 and Millstone Unit 3 Leak Before Break (LBB) analyses. Dominion's response to those questions is included below.

Question #1:

What systems are covered by leak-before-break analyses for each unit? Include components and materials evaluated. It appears that LBB methodology has been approved for Millstone Unit 2 for pressurizer surge line piping and portions of the safety injection and shutdown cooling systems. Yet, only RCS is addressed in the applicant's LRA. Will the LBB methodology for the pressurizer surge line piping and portions of the safety injection and shutdown cooling systems no longer be used for the period of extended operation? What evaluation will be used instead to evaluate the dynamic effects associated with postulated pipe ruptures? Will pipe whip restraints and supports be added?

Dominion Response:

For Millstone Unit 2, the systems and components that have been analyzed for Leak-Before-Break (LBB) include the reactor coolant loop piping (hot leg, cold leg, and crossover piping), the pressurizer surge line, and portions of the safety injection and shutdown cooling systems. Each of the LBB analyses associated with these systems and components were evaluated for the period of extended operation. The discussion and conclusions in LRA Section 4.7.4 are intended to envelope all of the current design basis LBB analyses. As a result, the current design basis for LBB carries forward to the end of the period of extended operation and no additional evaluations or plant modifications are necessary to evaluate the dynamic effects associated with postulated pipe ruptures. The materials evaluated for these components include carbon and low alloy steels, stainless steel (including cast austenitic stainless steel (CASS)) and nickel-based alloys.

For Millstone Unit 3, the reactor coolant system loop piping (hot leg, cold leg and crossover piping) has been evaluated for LBB. The materials evaluated for these components include carbon and low alloy steels, stainless steel (including CASS), and nickel-based alloys.

Question #2:

How did the applicant perform its TLAA evaluations on the systems covered by LBB analyses? What did the re-analysis involve (include aging of cast austenitic stainless steel components, fatigue analysis, the recent issue of PWSCC of nickel-based alloy components and weldments, and changes to the plant since the time that the analyses were reviewed and approved by the NRC staff that may have an effect on the LBB analyses)?

Provide documented justification that the LBB analyses for systems covered by leak-before-break remain valid for the period of extended operation. Provide justification that the analyses have been projected to the end of the period of extended operation or that the effects of aging on the intended functions of the systems covered by a LBB analysis will be adequately managed for the period of extended operation.

Dominion Response:

The Millstone Unit 2 and 3 LBB analyses were determined to remain valid for the period of extended operation by evaluating the time-based inputs to the LBB analyses. The design basis LBB analyses were determined for Millstone Units 2 and 3. For each LBB analysis, the inputs to the evaluation were reviewed to identify time-limited assumptions. Thermal aging of cast austenitic stainless steel (CASS) materials and fatigue crack growth calculations were determined to be time-based inputs as defined in 10 CFR 54.3 and required evaluation for the period of extended operation. Both of these analysis components were evaluated and were projected to be acceptable to the end of the period of extended operation.

The TLAA evaluations of metal fatigue for Millstone Unit 2 and Unit 3 are discussed in LRA Sections 4.3.1, as supplemented by the responses to NRC RAIs 4.3.1-1, 4.3.1-2, 4.3.1-4, and 4.7.4-1. The metal fatigue TLAA evaluations conclude that design basis limits are not exceeded for ASME Class 1 components (which envelopes the components evaluated for LBB) through the period of extended operation.

Thermal aging of CASS materials for components that have been evaluated for LBB has been evaluated as a TLAA since long-term exposure of CASS materials to reactor coolant system operating temperatures results in an increase in material hardness while its ductility, impact strength and fracture toughness decrease. Fracture toughness represents one of the more important design inputs in a LBB evaluation. The degree of reduction in CASS fracture toughness is dependent on the time of thermal exposure. However, the change in material properties due to thermal aging reaches a saturation value, after which material property changes resulting from additional thermal exposure are not significant. The evaluation of the thermal aging of CASS material for Millstone Unit 2 and Unit 3 LBB evaluations consisted of a review to determine whether the fracture toughness value used in the analyses was conservative relative to the fully-aged value for fracture toughness for the CASS components. The review concluded that the analysis values were either equal to or lower than the worst-case saturation (fully aged) values for fracture toughness in all cases. Therefore, since the CASS material property values used in current design basis LBB evaluations represent fully aged (saturation) values, and since these values would not change with further exposure time, the LBB evaluations are not affected by thermal aging of CASS materials for the period of extended operation.

Corrosion of materials (including nickel-based alloy welds) for components analyzed for LBB was considered for the current design basis LBB evaluations, consistent with the requirements for performing the analyses. Although not determined to be a time-based issue for LBB in accordance with 10 CFR 54.3, and therefore not considered in the TLAA evaluation, corrosion of nickel-based alloys (including PWSCC) has been identified as an aging effect requiring management in LRA Section 3.0. Cracking due to PWSCC of nickel-based alloys is managed by the Inservice Inspection Program: Systems, Components, and Supports AMP described in LRA Section B2.1.18. Industry programs are in place whose objectives include the investigation of aging effects applicable to nickel-based alloys (i.e., PWSCC in Alloy 600 base metal and Alloy 82/182 weld metals). Millstone Unit 2 and Millstone Unit 3 have committed to follow these industry efforts and to identify and implement the appropriate aging management activities resulting from industry recommendations. These commitments are identified in the Millstone Unit 2 and Millstone Unit 3 LRAs, Appendix A, Table A6.0-1 License Renewal Commitments, Item 14 and 15, respectively.

There have been no changes to the plant that materially affect the LBB analyses since the LBB analyses were reviewed and approved by the NRC for Millstone Units 2 and 3.

As a result of the TLAA evaluation performed for the LBB analyses for Millstone Unit 2 and Unit 3, the analyses have been projected to the end of the period of extended operation consistent with 10 CFR 54.21(c)(1), Option (ii).

Attachment 2

Supplemental Information to Previous Request for Additional Information Responses

Millstone Power Station Units 2 & 3
Dominion Nuclear Connecticut, Inc.

Request for Additional Information (RAI) Supplementary Items for Millstone, Units 2 and 3, LRA

RAI Supplement 3.1.1-2, Units 2 & 3

In response to **RAI 3.1.1-2**, in a letter dated December 3, 2004, the applicant stated that the closure head stud assembly does not experience relative motion other than normal stud removal and installation during refueling activities. These activities are closely monitored by procedure and any degradation is dispositioned by supplemental examination, corrective measures or repairs, analytical evaluation of the component function, or replacement of the component to ensure continued structural integrity and function of the component. There is no significant continuing wear to the reactor vessel closure studs that would lead to a loss of component function and require monitoring by an aging management program. Therefore, the applicant did not consider loss of material due to wear as an applicable aging effect for the closure head stud assembly. However, AMP XI.M3 of NUREG-1801 and RG 1.65 indicates that reactor closure studs are susceptible to loss of material due to wear. In addition, RG 1.65 requires, and the applicant uses coatings and lubrication which are used to reduce wear. Therefore, the staff requests that the LRA specify loss of material due to wear as an aging effect for the closure head stud assembly and specify the AMP to be applied.

Dominion Response:

Although wear of the reactor vessel closure studs, due to reactor disassembly and reassembly, is not expected to affect the intended function of the bolting, the LRA is being supplemented to include loss of material due to wear as an aging effect requiring management for Millstone Unit 2 and 3, consistent with NUREG-1801 item IV.A2.1-d. The LRA is also being supplemented to reflect that this aging effect will be managed by the Inservice Inspection Program: Systems, Components, and Supports AMP. The results of the aging management review for wear of the closure studs are provided in Table 3.1.2-1a.

RAI Supplement 3.1.1-3, Unit 3

In response to **RAI 3.1.1-3**, in a letter dated December 3, 2004, the applicant stated loss of fracture toughness due to neutron irradiation embrittlement is an applicable aging effect for those reactor pressure vessel subcomponents exposed to a neutron fluence greater than $1x10^{17}$ n/cm^2 (E>1MeV). This threshold level of fluence is experienced by the beltline region subcomponents identified in LRA Table 3.1.2-1 as susceptible to loss of fracture toughness. Based on a supplemental evaluation performed by the applicant, the upper shell and primary inlet nozzles, and their associated welds are subjected to loss of fracture toughness due to neutron irradiation embrittlement and will be managed with the Reactor Vessel Surveillance AMP. However, the staff notes that the applicant did not provide the USE and PTS evaluation for these reactor pressure vessel subcomponents as required by Appendix G to10 CFR Part 50, and 10 CFR 50.61, respectively. Therefore to confirm that the USE and PTS evaluation for these sub-components meet regulatory requirements at the end of the period of extended operation, the staff requests the applicant to include the USE and PTS evaluation (similar to the data currently in Tables 1 and 2 of the FSAR for the other reactor vessel sub-components) for the upper shell and primary nozzles, and their associated welds into Tables 1 and 2 of the Millstone Unit 3 FSAR supplement and determine the effect on the limiting materials.

Dominion Response:

The Millstone Unit 2 upper shelf energy values for the limiting reactor pressure vessel beltline materials have been calculated in accordance with 10 CFR Part 50, Appendix G using the most recent material property information through the period of extended operation. These results, discussed in Millstone Unit 2 LRA Section 4.2.2 (presented in LRA Table 4.2-1), demonstrate acceptable USE values through the period of extended operation. The Millstone Unit 2 limiting beltline materials and their associated USE values are identified in Dominion's response to RAI 4.2.2-4.

The RT_{PTS} values for the limiting Millstone Unit 2 reactor pressure vessel beltline materials have been calculated consistent with Regulatory Guide 1.99, Revision 2 requirements through the period of extended operation. These results, discussed in Millstone Unit 2 LRA Section 4.2.3 (presented in LRA Table 4.2-2), demonstrate that the RT_{PTS} screening criteria have been met in all cases through the period of extended operation. The Millstone Unit 2 limiting beltline materials and their associated RT_{PTS} values, developed in accordance with 10 CFR 50.61, are identified in Dominion's response to RAI 4.2.3-1.

As required by 10 CFR Part 50, Appendix G and 10 CFR 50.61, the upper shelf energy and RT_{PTS} values for the expanded beltline regions of the Millstone Unit 2 reactor pressure vessel subcomponents are contained in Table 3.1.1-3-1 - Upper Shelf Energy Values at 54 EFPY (Expanded Beltline), and Table-3.1.1-3-2 - RT_{PTS} Values at 54 EFPY (Expanded Beltline).

Millstone Unit 2 LRA Appendix A "FSAR Supplement", Section A3.1.1 – Upper Shelf Energy, and Section A3.1.2 – Pressurized Thermal Shock has been reviewed. No changes to the limiting materials have been identified. Sections A3.1.1 and A3.1.2 are correct as written.

Table 3.1.1-3-1
Millstone Unit 2
Upper Shelf Energy Values at 54 EFPY
(Expanded Beltline)

Material Description				Cu Wt. %	Initial USE Ft-lbs	Fluence 1/4t n/cm^2	USE Ft-lbs	% Drop USE Position 1.2
Reactor Pressure Vessel Beltline Region Location	**Matl. Ident.**	**Heat Number**	**Type**					
Upper Shell Plate	C-504-1	C5804-2	SA-533B Cl.1	0.13	76.7	1.45E18	66.0	14
Upper Shell Plate	C-504-2	C5809-2	SA-533B Cl.1	0.13	85.2	1.45E18	73.3	14
Upper Shell Plate	C-504-3	C5809-1	SA-533B Cl.1	0.13	81.3	1.45E18	69.9	14
Upper Shell Axial Welds	1-203 A/C	12008/21935 (3869)	Linde 1092	0.22	97	1.45E18	74.7	23
Upper Shell Axial Welds	1-203 A/C	12008/21935 (3889)	Linde 1092	0.20	97	1.45E18	75.7	22
Upper Girth Seam Weld	8-203	33A277 (3922)	Linde 0091	0.30	101	1.45E18	72.2	28.5
Upper Girth Seam Weld	8-203	10137 (3999)	Linde 0091	0.23	101	1.45E18	77.3	23.5
Outlet Nozzle Forging	C-502-1	9-7356-001	SA-508 Cl.2	-	-	6.35E16	-	N/A
Outlet Nozzle Forging	C-502-2	9-7375-002	SA-508 Cl.2	-	-	6.35E16	-	N/A
Outlet Nozzle Welds	10-205 A/B	IOBJ	(a)	-	-	6.35E16	-	N/A
Outlet Nozzle Welds	10-205 A/B	ICJJ	SA-316	-	-	6.35E16	-	N/A
Outlet Nozzle Welds	10-205 A/B	JADJ	SA-316	-	-	6.35E16	-	N/A
Outlet Nozzle Welds	10-205 A/B	KBEJ	SA-316	-	-	6.35E16	-	N/A
Outlet Nozzle Welds	10-205 A/B	BOIA	SA-316	-	-	6.35E16	-	N/A
Outlet Nozzle Welds	10-205 A/B	BOLA	SA-316	-	-	6.35E16	-	N/A
Outlet Nozzle Welds	10-205 A/B	KAHJ	SA-316	-	-	6.35E16	-	N/A
Outlet Nozzle Welds	10-205 A/B	KOIJ	SA-316	-	-	6.35E16	-	N/A

(a) Heat IOBJ CMTRs not available.

Table 3.1.1-3-2
Millstone Unit 2
RT_{PTS} Values at 54 EFPY
(Expanded Beltline)

Material Description				Chemical Composition		Initial RT_{NDT} °F	Chemistry Factor °F	Inner Surface Fluence n/cm²	Margin °F	ΔRT_{PTS} °F	RT_{PTS} °F
Reactor Vessel Beltline Region Location	Matl. Ident.	Heat Number	Type	Cu Wt. %	Ni Wt. %						
Upper Shell Plate	C-504-1	C5804-2	SA-533B Cl.1	0.13	0.58	22	90.4	2.43E18	34	55.8	111.8
Upper Shell Plate	C-504-2	C5809-2	SA-533B Cl.1	0.13	0.56	22	89.8	2.43E18	34	55.4	111.4
Upper Shell Plate	C-504-3	C5809-1	SA-533B Cl.1	0.13	0.56	15	89.8	2.43E18	34	55.4	104.4
Upper Shell Axial Welds	1-203 A/C	12008/21935 (3869)	Linde 1092	0.22	0.867	-56	210.7	2.43E18	65.5	130.0	139.5
Upper Shell Axial Welds	1-203 A/C	12008/21935 (3889)	Linde 1092	0.20	0.867	-56	203.7	2.43E18	65.5	125.7	135.2
Upper Girth Seam Weld	8-203	33A277 (3922)	Linde 0091	0.30	0.165	-56	143.4	2.43E18	65.5	88.4	97.9
Upper Girth Seam Weld	8-203	10137 (3999)	Linde 0091	0.23	0.043	-56	104.4	2.43E18	65.5	64.4	73.9
Outlet Nozzle Forging	C-502-1	9-7356-001	SA-508 Cl.2	0.09	0.78	-60	58	1.21E17	7.2	7.2	-45.6
Outlet Nozzle Forging	C-502-2	9-7375-002	SA-508 Cl.2	0.07	0.82	-24	44	1.21E17	5.5	5.5	-13.0
Outlet Nozzle Welds	10-205 A/B	IOBJ	(a)	0.03	1.08	10	41	1.21E17	5.1	5.1	20.2
Outlet Nozzle Welds	10-205 A/B	ICJJ	SA-316	0.03	0.99	10	41	1.21E17	5.1	5.1	20.2
Outlet Nozzle Welds	10-205 A/B	JADJ	SA-316	0.03	0.96	10	41	1.21E17	5.1	5.1	20.2
Outlet Nozzle Welds	10-205 A/B	KBEJ	SA-316	0.03	1.04	10	41	1.21E17	5.1	5.1	20.2
Outlet Nozzle Welds	10-205 A/B	BOIA	SA-316	0.02	0.93	10	27	1.21E17	3.4	3.4	16.8
Outlet Nozzle Welds	10-205 A/B	BOLA	SA-316	0.02	0.93	-60	27	1.21E17	3.4	3.4	-53.2
Outlet Nozzle Welds	10-205 A/B	KAHJ	SA-316	0.03	1.08	10	41	1.21E17	5.1	5.1	20.2
Outlet Nozzle Welds	10-205 A/B	KOIJ	SA-316	0.03	1.0	10	41	1.21E17	5.1	5.1	20.2

(a) Heat IOBJ CMTRs not available.

The Millstone Unit 3 upper shelf energy values for the limiting 3 reactor pressure vessel beltline materials have been calculated in accordance with 10 CFR 50, Appendix G using the most recent material property information through the period of extended operation. These results, discussed in Millstone Unit 3 LRA Section 4.2.2 (presented in LRA Table 4.2-1), demonstrate acceptable USE values through the period of extended operation. The Millstone Unit 3 limiting beltline materials and their associated USE values are identified in Dominion's response to RAI 4.2.2-4.

The RT_{PTS} values for the limiting Millstone Unit 3 reactor pressure vessel beltline materials have been calculated consistent with Regulatory Guide 1.99, Revision 2 requirements through the period of extended operation. These results, discussed in Millstone Unit 3 LRA Section 4.2.3 (presented in LRA Table 4.2-2), demonstrate that the RT_{PTS} screening criteria have been met in all cases through the period of extended operation. The Millstone Unit 3 limiting beltline materials and their associated RT_{PTS} values, developed in accordance with 10 CFR 50.61, are identified in Dominion's response to RAI 4.2.3-1.

As required by 10 CFR Part 50, Appendix G and 10 CFR 50.61, the upper shelf energy and RT_{PTS} values for the expanded beltline regions of the Millstone Unit 3 reactor pressure vessel subcomponents are contained in Table 3.1.1-3-3 - Upper Shelf Energy Values at 54 EFPY (Expanded Beltline), and Table 3.1.1-3-4 - RT_{PTS} Values at 54 EFPY (Expanded Beltline).

Millstone Unit 3 LRA Appendix A "FSAR Supplement", Section A3.1.1 – Upper Shelf Energy, and Section A3.1.2 – Pressurized Thermal Shock has been reviewed. No changes to the limiting materials have been identified. Sections A3.1.1 and A3.1.2 are correct as written.

Table 3.1.1-3-3
Millstone Unit 3
Upper Shelf Energy Values at 54 EFPY
(Expanded Beltline)

Material Description				$Cu^{(a)}$ Wt. %	Initial USE Ft-lbs	Fluence 1/4t n/cm^2	USE Ft-lbs	% Drop USE Position 1.2
Reactor Pressure Vessel Beltline Region Location	Matl. Ident.	Heat Number	Type					
Nozzle Shell Plate	B9804-1	C4036-1	SA-533B Cl.1	0.05	85.5	3.76E17	78	9
Nozzle Shell Plate	B9804-2	C4021-2	SA-533B Cl.1	0.08	104	3.76E17	95	9
Nozzle Shell Plate	B9804-3	C4068-2	SA-533B Cl.1	0.05	103	3.76E17	94	9
Inlet Nozzle	B9806-3	11-5627	SA-508 Cl.2	0.09	162	5.46E16	-	N/A
Inlet Nozzle	B9806-4	11-5627	SA-508 Cl.2	0.09	158	5.46E16	-	N/A
Inlet Nozzle	R5-3	4-2804	SA-508 Cl.2	0.07	130	5.46E16	-	N/A
Inlet Nozzle	R5-4	4-2894	SA-508 Cl.2	0.08	136	5.46E16	-	N/A
Nozzle Shell Long. Weld	101-122A	86998 & 87011	Linde 0091	0.05	>101[b]	3.76E17	>92	9
Nozzle Shell Long. Weld	101-122B & C	87011	Linde 0091	0.05	>123[b]	3.76E17	>112	9
Inlet Nozzle Weld	105-121A	5P7388 & HAACE	Linde 0091	0.09	>89	5.46E16	-	N/A
Inlet Nozzle Weld	105-121B	GABFE & 4P6524	Linde 0091	0.16	177	5.46E16	-	N/A
Inlet Nozzle Weld	105-121C	(5P7388 & 4P6524)[c]	Linde 0091	0.16	>89	5.46E16	-	N/A
Inlet Nozzle Weld	105-121D	(4P6524 & HAOEE)[d]	Linde 0091	0.16	147	5.46E16	-	N/A
Nozzle Shell to Inter. Shell Girth	103-121	87000	Linde 0091	0.05	132	4.27E17	120	9

a) For those materials with multiple heat numbers, the highest copper weight percent was used to determine the percent decrease in USE.
b) Charpy test did not reach 100% shear. This is the highest energy value at the highest achieved shear value. Since this value is not at 100% shear, it is conservative to use for USE.
c) Also made from Heats 30502, JAACE, IAOCE, IAOJE and HAACE.
d) Also made from Heats FAOJE, HABIE and HAACE.

Table 3.1.1-3-4
Millstone Unit 3
RT_{PTS} Values at 54 EFPY
(Expanded Beltline)

Material Description				Chemical Composition		Initial RT_{NDT}	Chemistry Factor	Inner Surface Fluence	Margin	ΔRT_{PTS}	RT_{PTS}
Reactor Vessel Beltline Region Location	Matl. Ident.	Heat Number	Type	Cu Wt. %	Ni Wt. %	°F	°F	n/cm^2	°F	°F	°F
Nozzle Shell Plate	B9804-1	C4036-1	SA-533B Cl.1	0.05	0.62	40	31	7.16E17	10.85	10.85	62
Nozzle Shell Plate	B9804-2	C4021-2	SA-533B Cl.1	0.08	0.64	20	51	7.16E17	17.85	17.85	56
Nozzle Shell Plate	B9804-3	C4068-2	SA-533B Cl.1	0.05	0.65	0	31	7.16E17	10.85	10.85	22
Inlet Nozzle	B9806-3	11-5627	SA-508 Cl.2	0.09	0.83	10	58	1.04E17	6.38	6.38	23
Inlet Nozzle	B9806-4	11-5627	SA-508 Cl.2	0.09	0.82	0	58	1.04E17	6.38	6.38	13
Inlet Nozzle	R5-3	4-2804	SA-508 Cl.2	0.07	0.80	-10	44	1.04E17	4.84	4.84	0
Inlet Nozzle	R5-4	4-2894	SA-508 Cl.2	0.08	0.81	0	51	1.04E17	5.61	5.61	11
Nozzle Shell Long. Weld	101-122A	86998 & 87011	Linde 0091	0.05	0.12	-10	39.8	7.16E17	13.93	13.93	18
Nozzle Shell Long. Weld	101-122B & C	87011	Linde 0091	0.05	0.12	-50	39.8	7.16E17	13.93	13.93	-22
Inlet Nozzle Weld	105-121A	5P7388 & HAACE	Linde 0091	0.09	0.05	-60	45.3	1.04E17	4.98	4.98	-50
Inlet Nozzle Weld	105-121B	GABFE & 4P6524	Linde 0091	0.16	0.06	-50	75.4	1.04E17	8.29	8.29	-33
Inlet Nozzle Weld	105-121C	(5P7388 & 4P6524)[(a)]	Linde 0091	0.16	0.06	-50	75.4	1.04E17	8.29	8.29	-33
Inlet Nozzle Weld	105-121D	(4P6524 & HAOEE)[(b)]	Linde 0091	0.16	0.06	-50	75.4	1.04E17	8.29	8.29	-33
Nozzle Shell to Inter. Shell Girth	103-121	87000	Linde 0091	0.05	0.13	-40	41	7.16E17	14.35	14.35	-11

(a) Also made from Heats 30502, JAACE, IAOCE, IAOJE and HAACE.
(b) Also made from Heats FAOJE, HABIE and HAACE.

RAI Supplement 3.1.2-1, Units 2 & 3

In response to RAI 3.1.2-1, in a letter dated December 3, 2004, the applicant stated that loss of material due to wear was not considered an applicable aging effect for the reactor vessel flange and core support ledge since they do not experience relative motion other than normal reactor disassembly and reassembly during refueling activities. These activities are closely monitored by procedure and any degradation is dispositioned by supplemental examination, corrective measures or repairs, analytical evaluation of the component function, or replacement of the component to ensure continued structural integrity and function of the component. There is no significant continuing wear to the reactor vessel flange and core support ledge that would lead to a loss of component function that would require monitoring by an aging management program. However, the staff considers wear to be an aging effect as identified by NUREG-1801, section IVA.2.5-f because the reactor vessel flange and support ledge do experience relative motion during reactor disassembly and reassembly during refueling activities. This aging effect should then be monitored. Since the applicant states this refueling activity is monitored by procedures, some type of inspection must be performed to monitor wear of these components. Therefore, the staff requests that the LRA specify loss of material due to wear as an aging effect for the reactor vessel flange and core support ledge. In addition the applicant is requested to discuss the inspections performed by the refueling activity procedures that monitors wear for these components or include the corresponding aging management program recommended by NUREG-1801 (AMP XI.M1, "Inservice Inspection").

Dominion Response:

Although wear of the reactor vessel flange and core support ledge, due to reactor disassembly and reassembly, is not expected to affect the intended function of the component, loss of material due to wear will be considered an aging effect requiring management for Millstone Unit 2 and Unit 3, consistent with NUREG-1801 item IV.A2.5-f. The aging effect will be managed by the Inservice Inspection Program: Reactor Vessel Internals AMP. The results of the aging management review for wear of the reactor vessel flange are provided in Table 3.1.2-1a.

Millstone Unit 2 and Unit 3 LRA

Table 3.1.2-1a: Reactor Vessel, Internals, and Reactor Coolant System – Reactor Vessel – Aging Management Evaluation

Subcomponent	Intended Function(s)	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Closure Head Stud Assembly	PB	Low-alloy Steel	(E) Air	Loss of Material	Inservice Inspection Program; Systems, Components and Supports	IV.A2.1-d	3.1.1-47	B
Vessel Flange and Core Support Ledge (and cladding)	PB, SS	Low-alloy Steel	(E) Air	Loss of Material	Inservice Inspection Program; Reactor Vessel Internals	IV.A2.5-f	3.1.1-40	B

RAI Supplement 3.1.3-2B, Unit 3

In response to RAI 3.1.3-2, in a letter dated December 3, 2004, the applicant stated that material for the Millstone Unit 3 pressurizer spray head is CASS. The plant specific aging management program for managing the aging effects associated with the pressurizer spray head is the Chemistry Control for Primary Systems Program.

The reactor coolant system stainless steel materials, including the pressurizer spray head, are exposed internally to a high-quality primary water and/or steam environment that is not expected to result in significant SCC. Therefore, the applicant states that the Chemistry Control for Primary Systems Program AMP provides reasonable assurance that cracking resulting from SCC will not prevent the spray head from performing its intended function. In Section 3.1.2.2.7 of NUREG-1800, the staff recommends that a plant-specific aging management program be proposed to manage crack initiation and unacceptable crack growth in pressurizer spray heads because existing programs may not be capable of mitigating or detecting crack initiation and growth due to SCC. This inspection should be capable of detecting and resolving cracks in the pressurizer spray heads. Therefore, the staff agrees that the water chemistry can be used to mitigate SCC, but an inspection is necessary to indicate whether the water chemistry has prevented SCC and to characterize any cracking in the CASS pressurizer spray heads.

Dominion Response:

The following commitment will be added to Millstone Unit 3 LRA Appendix A, "FSAR Supplement" Section A2.1.17:

"• Pressurizer Spray Head Assembly Cracking"

"The pressurizer spray head assembly will be either replaced or inspected utilizing the best currently available (at the time of inspection) techniques for detecting cracking resulting from SCC. This commitment is identified in Appendix A, Table A6.0-1 License Renewal Commitments, Item 37".

An additional item will be added to Millstone Power Station Unit 3 Appendix A "FSAR Supplement", Table A6.0 -1 as follows:

Item: 37

Commitment:

"The pressurizer spray head assembly will be either replaced or inspected utilizing the best currently available (at the time of inspection) techniques for detecting cracking resulting from SCC. "

Source:

"Inservice Inspection Program: Systems, Components and Supports."

Schedule:

"Prior to Period of Extended Operation"

RAI Supplement 3.1.3-3, Unit 3

The applicant stated in Section 4.3.1 of the Millstone Unit 3 LRA that the cast austenitic stainless steel pressurizer spray head assembly has been evaluated for susceptibility to thermal embrittlement using the guidance and information contained in ERPI Report TR-106092. In addition the applicant stated that acceptable results employing applicable loads (e.g., thermal cycles) and material properties have been calculated over the 60 year license renewal period. The staff notes that NUREG-1801, Section XI.M12 recommends the CASS material to be evaluated based on the criteria set forth in the May 19, 2000, NRC letter to determine susceptibility to thermal aging embrittlement. This letter provided the staff's position on thermal aging embrittlement. The staff requests that the applicant confirm that the evaluation performed meets the guidelines of the May 19, 2000, NRC letter and NUREG-1801. If the evaluation does not conform to these guidelines, provide the results of an evaluation that meets the guidelines of the May 19, 2000, NRC letter and the information (i.e., Molybdenum content, casting method and percent ferrite) to confirm that the spray head satisfies the criteria in the staff's letter dated May 19, 2000. The applicant is also requested to discuss how this evaluation meets the requirements of 10 CFR 54.21(c)(1)(i), (ii) or (iii).

Dominion Response:

This item was originally RAI 4.3.1-5, to which Dominion responded in Letter 04-720A dated January 11, 2005. As a follow-up, RAI Supplement 4.7.3-1(a) was submitted to Dominion. The response to that item is provided in the Dominion Response to RAI Supplement 4.7.3-1(a) later in this attachment.

RAI Supplement 3.6-2

1. The applicant is requested to document why the Noryl insulated bus duct is not subjected to aging when it is in an air conditioned service building.

2. The applicant is also requested to document why the bolting of the bus conductors is not subject to thermal cycling between the normal no load and the reserve 67% loading during shutdown.

Dominion Response:

1. The 4.16kV and 6.9kV non-segregated bus duct conductors are insulated with a NORYL® resin material. The 6.9kV bus ducts are not normally loaded and therefore the evaluation of the normally loaded 4.16kV bus duct is considered bounding for aging of the bus insulation. NORYL® is a polyphenylene oxide (PPO) resin made by General Electric Company. An evaluation of the NORYL® bus insulation used in these bus ducts was performed to determine the functional life of the material under the actual operating conditions at Millstone Unit 3. The review was performed by an outside consultant and the results are summarized here.

   There are several grades of NORYL® resin used in the electrical insulation application, each with differing resistance to aging stressors. The evaluation of the bus insulation for these bus ducts determined that the type of NORYL® used is the EN265 compound.

   An analysis of the estimated operating loads and ambient air conditions determined that the normal operating temperature of the NORYL® resin insulated bus would be less than 65°C. This conclusion is based on a maximum ambient air temperature in the air conditioned space of 30°C, and the maximum temperature rise caused by ohmic heating (based on bus loading of 2000 amps) of less than 35°C.

   The thermal life of the material was estimated based on the Arrhenius equation methodology. There is no published Arrhenius thermal life curve available for NORYL® resin EN265, so an approximation of its thermal life was obtained by comparing it to a similar NORYL® resin (SE1-GFN3) that has the same UL Continuous Use Temperature rating of 105°C. The acceptance criterion established for the evaluation was 50% retention of the Tensile Impact Strength property of the material. The evaluation concluded that the maximum allowable service temperature for NORYL® resin EN265 for a 60-year life is 81.2°C without reducing the tensile impact strength below 50% of the original value. As indicated above, the maximum normal operating temperature for this bus would not exceed 65°C, providing considerable margin to thermal degradation.

Therefore, the evaluation concluded that there is reasonable assurance that the NORYL® resin has a minimum 60-year life in this application.

The estimated loading of 2000 amps that was used for the 4.16kV bus duct in this evaluation was a conservatively high estimate. Actual measured bus loading indicates that the bus duct normally operates at less than 400 amps, and the temperature rise due to ohmic heating is considerably less than the 35°C used in the evaluation. Actual measurements also indicated a start-up (i.e., when the RSST is supplying power) maximum load of less than 1400 amps. This condition occurs for less than a few hours per year, however, and is bounded by the conservative loading estimate used in the evaluation. Therefore, based on actual bus loading, there is significant margin in the evaluation of the insulation thermal lifetime.

In consideration of the conservative nature of the evaluation performed for the NORYL® bus insulation, and the design features of the bus duct that preclude the entrance of moisture and debris in the duct, it is concluded that cracking of the insulation material leading to a loss of insulating function, as described in Information Notices 89-64 and 98-36, is not expected to occur. Therefore, an aging management program is not required for the bus insulation.

2. As stated in the response to RAI 3.6-2, the 6.9kV bus ducts are normally energized but not loaded; therefore, thermal cycling does not occur. The 4.16kV bus ducts were initially evaluated based on an estimated load of 30% of rating under normal operating conditions and 67% of rating when the RSST is supplying power.

   As discussed in proposed Interim Staff Guidance (ISG)-17 on Periodic Inspection of Bus Ducts, bus ducts exposed to appreciable ohmic heating during operation may experience loosening of bolted connections because of repeated cycling of connected loads. This phenomenon can occur in heavily loaded circuits, i.e., those exposed to appreciable ohmic heating.

   The 4.16kV bus ducts are rated for 3000 amp load. As discussed in 1. above, actual measured bus loading indicates that the bus duct normally operates at less than 400 amps load (14% of rating), and during plant start-up (i.e., when the RSST is supplying power) operates at a maximum load of less than 1400 amps (47% of rating). These bus ducts would not be considered heavily loaded and are not subject to appreciable ohmic heating. The higher loading levels for these bus ducts is experienced during plant start-up when the normal station service bus is backfed from the 4kV vital bus, and this occurs infrequently. An estimate of the number of cycles (normal load to start-up load) can be derived from the number of expected refueling outages and assuming a mid-cycle outage (although mid-cycle outages are not normally required). Over a 60-year plant life, an estimate of the number of load, and thus thermal, cycles would be 80. If this number is conservatively doubled to account for the possibility of unexpected outages, the estimated number of thermal cycles would still only be 160. Therefore, considering the relatively light bus loading, the small magnitude of the thermal cycle due to the bus loading change, and the small number of thermal cycles, relaxation of bus bar splice bolting torque is not expected.

As an additional measure, the bus bar splice joint bolting includes the use of Belleville spring washers to ensure that the effects of thermal expansion and contraction of the splice and bus materials are accommodated without resulting in torque relaxation of the bolted joint.

RAI Supplement 4.2.1-3, Unit 2

In response to **RAI 4.2.1-3**, in a letter dated December 3, 2004, the applicant stated that Millstone Unit 2 does not use a fluence methodology in accordance with RG 1.190, and therefore may be less conservative. The staff has concluded from experience that fluence values calculated using methods not in compliance with the guidance in RG 1.190 could differ by as much as 40% had they been calculated with a method adhering to the RG 1.190 guidance. Assuming that the above fluence value is underestimated by 40% the value would be 5.67×10^{19} n/cm^2. Using equation 2 in RG 1.99 with the chemistry factor (CF) and the initial value from Table 4.2-2 of 110°F and 7.0°F respectively, the RT_{PTS} is equal to 197.9°F. This value is well within the screening criterion of 270°F of 10 CFR 50.61. Therefore, the staff concludes that the material properties are well within the safety limits and therefore, the proposed fluence value is acceptable. Similar results are obtained with the USE and are discussed in Section 4.2.2.2 of this SER. Therefore, even with a conservative estimated fluence values, the USE and RT_{PTS} values still meet the applicable screening criteria. In addition, the applicant is planning to submit P-T limit curves for 54 EFPY to the NRC in 2005. Since the applicant will be providing new P-T limit curves for 54 EFPY, the staff requests that the applicant commit to submit the reactor vessel fluence calculations using a methodology in accordance with RG 1.190, which will also support the P-T limit curve submittal, to the NRC along with the 54 EFPY P-T limit curves in 2005. The applicant is also requested to add to its list of commitments the submittal of a re-evaluation of the USE and RT_{PTS} to update the licensing basis to be consistent with the fluence values used in the P-T limit curves.

Dominion Response:

The response to RAI 4.2.1-1, in the letter dated December 3, 2004, indicated that past Millstone Unit 2 fluence calculations, though using an approved methodology, did not comply with Regulatory Guide 1.190 and that future fluence calculations will be in compliance with Regulatory Guide 1.190.

In the response to RAI 4.2.1-3, also contained in this letter, Dominion committed to add the following information to the Millstone Unit 2 and Millstone Unit 3 LRA Appendix A "FSAR Supplement", Section A3.1.3, Pressure-Temperature Limits:

> "Millstone Unit [2] [3] will continue to calculate P-T limits based on fluence values developed in accordance with Regulatory Guide 1.190 requirements, as amended or superseded by future regulatory guidance changes, through the period of extended operation."

Since this information does not specifically address USE and RT_{PTS}, the response is being supplemented as follows to clarify the compliance with Regulatory Guide 1.190 for each unit.

The following will be added to the Millstone Unit 2 LRA Appendix A "FSAR Supplement", Section A3.1.3, Pressure-Temperature Limits:

> "Millstone Unit 2 will calculate USE, RT_{PTS}, and P-T limits based on fluence values developed in accordance with Regulatory Guide 1.190 requirements, as amended or superseded by future regulatory guidance changes, through the period of extended operation.
>
> "Actions to be taken:
>
> "Updated USE, RT_{PTS}, and P-T limits based on fluence values developed in accordance with Regulatory Guide-1.190 requirements, as amended or superseded by future regulatory guidance changes, will be submitted to the NRC for review at least two years prior to the period of extended operation. This commitment is identified in Appendix A, Table A6.0-1, Item 37. "

The following will be added to the Millstone Unit 2 LRA Appendix A "FSAR Supplement", Table A6.0-1:

Item: "37"

Commitment: "Updated USE, RT_{PTS}, and P-T limits based on fluence values developed in accordance with Regulatory Guide 1.190 requirements, as amended or superseded by future regulatory guidance changes, will be submitted to the NRC for review at least two years prior to the period of extended operation. "

Source: "Reactor Vessel Neutron Embrittlement TLAA"

Schedule: "At Least Two Years Prior to the Period of Extended Operation"

The following will be added to the Millstone Unit 3 LRA Appendix A "FSAR Supplement", Section A3.1.3, Pressure-Temperature Limits:

"Millstone Unit 3 will calculate USE, RT_{PTS}, and P-T limits based on fluence values developed in accordance with Regulatory Guide 1.190 requirements, as amended or superseded by future regulatory guidance changes, through the period of extended operation.

RAI Supplement 4.2.2-1, Unit 2

In response to **RAI 4.2.2-1**, in a letter dated December 3, 2004, the applicant provided an USE evaluation for the Millstone Unit 2 reactor pressure vessel upper to middle circumferential weld (weld No. 8-203) which used weld wire heats 33A277 and 10137. The evaluation provided a USE value for weld No. 8-203 of 72.2 ft-lb for heat 33A277 and 77.3 ft-lb for heat 10137. The staff verified these values and determined that they are conservative and that a USE value of 72.2 ft-lb for weld No. 8-203 is acceptable. However, the applicant stated in their letter dated December 3, 2004, that this weld does not meet the definition of beltline region in Appendix G to 10 CFR Part 50, because it is above the active core. However, 10 CFR Part 50, Appendix G, paragraph II.F also defines the beltline region to include adjacent regions of the reactor vessel that are predicted to experience sufficient neutron radiation damage to be considered in the selection of the most limited material with regards to radiation damage. In addition, 10 CFR Part 50, Appendix H specifies that material exposed to peak neutron fluence that exceed 10^{17} n/cm^2 must be monitored by a surveillance program complying with ASTM E 185. Also, RG 1.99, Revision 2 has criteria for evaluating the USE and PTS for material exceeding this fluence value. The applicant determined that the inner surface fluence value for this weld to be 2.43×10^{18} n/cm^2. Therefore, the applicant is requested to update the FSAR supplement for Millstone Unit 2 by adding weld 8-203 and the corresponding USE value to Table 1 of the Millstone Unit 2 FSAR supplement.

Dominion Response:

The requested values are contained in this letter, in Dominion's supplemental response to RAI 3.1.1-3, Table 3.1.1-3-1 – Millstone Unit 2 "Upper Shelf Energy Values at 54 EFPY (Expanded Beltline)".

In keeping consistent with the level of detail in the Millstone Unit 2 and Millstone Unit 3 LRA Appendix A "FSAR Supplement" section, and Appendix A sections of previous applications, Dominion believes that the current wording, indicating that the USE value is greater than 50 ft-lbs, combined with this LRA supplementary information, should be adequate. Therefore, Dominion does not propose to modify its Millstone Unit 2 and Millstone Unit 3 LRA Appendix A "FSAR Supplement" sections to include the specific USE value.

RAI Supplement 4.2.2-2, Unit 2

In response to **RAI 4.2.2-2**, in a letter dated December 3, 2004, the applicant provided an USE evaluation for the Millstone Unit 2 reactor pressure vessel lower shell plate C-506-1, Heat C5667-1 using all available surveillance data as required by RG 1.99, Revision 2. The evaluation provided a USE value for shell plate C-506-1, Heat C5667-1 of 54.5 ft-lb using surveillance capsule W-97. The staff verified this value and determined that it is conservative and acceptable. However, the applicant did not include this revised USE value of 54.5 ft-lb in the FSAR supplement. Therefore, the applicant is requested to update the FSAR supplement for Millstone Unit 2 by revising the USE value from 76.1 ft-lb to 54.5 ft-lb for shell plate C-506-1, Heat C5667-1 in Table 1 of Section A3.1.1 to the Millstone Unit 2 FSAR supplement.

Dominion Response:

The calculated USE value at 54 EFPY for the Millstone Unit 2 reactor pressure vessel lower shell plate C-506-1, heat number C5667-1 is 65.3 ft-lbs, not the 54.5 ft-lbs originally identified in Table 1 of Dominion's response to RAI 4.2.2-2. The 65.3 ft-lbs represents the reactor pressure vessel lower shell plate C-506-1, heat number C5667-1, USE value developed using all available surveillance data. The RAI response number of 54.5 ft-lbs, used by the staff in it USE evaluation is more conservative than the actual USE value of 65.3 ft-lbs. Therefore, the evaluation results still hold true that the USE value is acceptable.

In keeping consistent with the level of detail in the Millstone Unit 2 and Millstone Unit 3 LRA Appendix A "FSAR Supplement" section, and Appendix A sections of previous applications, Dominion believes that the current wording, indicating that the USE value is greater than 50 ft-lbs, combined with this LRA supplementary information, should be adequate. Therefore, Dominion does not propose to modify its Millstone Unit 2 and Millstone Unit 3 LRA Appendix A "FSAR Supplement" sections to include the specific USE value.

RAI Supplement 4.2.2-4, Units 2 & 3

In response to **RAI 4.2.2-4**, in a letter dated December 3, 2004, the applicant provided information, including the beltline USE values, that will be incorporated into Section A3.1.1 of the Millstone Units 2 and 3 FSAR supplements concerning the limiting beltline material and that they are in compliance with the applicable requirements in 10 CFR Part 50, Appendix G. The staff has reviewed this information and requests the following supplemental information. The applicant stated that the USE values for the limiting beltline materials have been calculated in accordance with 10 CFR 50.61 and demonstrate acceptable USE values through the period of extended operation. Confirm that the USE was performed in accordance with 10 CFR Part 50, Appendix G, and not 10 CFR 50.61 (which is used for PTS evaluation). The confirmed information should be incorporated into the FSAR supplement accordingly.

Dominion Response:

The Millstone Unit 2 and 3 USE values were developed using 10 CFR Part 50, Appendix G, not 10 CFR 50.61. Dominion erroneously cited 10CFR 50.61 in its response to RAI 4.2.2-4.

Both Millstone Unit 2 and Millstone Unit 3 LRA Appendix A "FSAR Supplements", Section A3.1.1 - Upper Shelf Energy, have been reviewed. Each section correctly references the use of 10 CFR Part 50, Appendix G in developing USE values. Therefore, no related correction is required for these LRA sections.

RAI Supplement 4.2.3-1, Units 2 & 3

In response to **RAI 4.2.3-1**, in a letter dated December 3, 2004, the applicant provided information that will be incorporated into Section A3.1.1 of the Millstone Units 2 and 3 FSAR supplement concerning the limiting beltline material and that they are in compliance with the applicable requirements in 10 CFR Part 50.61. The staff has reviewed this information and requests the following supplemental information. The applicant stated that the RT_{PTS} values for the limiting beltline materials have been calculated in accordance with RG 1.99, Revision 2 through the period of extended operation and demonstrate acceptable RT_{PTS} values through the period of extended operation. Confirm that the RT_{PTS} was performed in accordance with 10 CFR 50.61. The confirmed information should be incorporated into the FSAR supplement accordingly.

Dominion Response:

Dominion has confirmed that the Millstone Unit 2 and Millstone Unit 3 RT_{PTS} values were developed in accordance with 10 CFR 50.61.

Dominion has also confirmed that the Millstone Unit 2 and Millstone Unit 3 LRA Appendix A "FSAR Supplements", Section A3.1.2 – Pressurized Thermal Shock, correctly reference the use of 10 CFR 50.61 in developing the RT_{PTS} values. Therefore, no related correction is required for these LRA sections.

RAI Supplement 4.2.3-2, Unit 2

In response to **RAI 4.2.2-1**, in a letter dated December 3, 2004, the applicant provided an RT_{PTS} evaluation for the Millstone Unit 2 reactor pressure vessel upper to middle circumferential weld (weld No. 8-203) which used weld wire heats 33A277 and 10137. The evaluation provided a RT_{PTS} value for weld No. 8-203 of 97.9 °F for heat 33A277 and 73.9 °F for heat 10137. The staff verified these values and determined that they are conservative and are acceptable. However, the applicant stated in their letter dated December 3, 2004, that this weld does not meet the Appendix G definition of beltline region, since it is above the active core. However, 10 CFR Part 50, Appendix G, paragraph II.F also defines the beltline region to include adjacent regions of the reactor vessel that are predicted to experience sufficient neutron radiation damage to be considered in the selection of the most limited material with regards to radiation damage. In addition, 10 CFR Part 50, Appendix H specifies that material exposed to peak neutron fluence that exceed 10^{17} n/cm^2 must be monitored by a surveillance program complying with ASTM E 185. Also, RG 1.99, Revision 2 has criteria for evaluating the USE and RT_{PTS} for material exceeding this fluence value. The applicant determined that the inner surface fluence value for this weld to be 2.43 x10^{18} n/cm^2. Therefore, the applicant is requested to update the FSAR supplement for Millstone Unit 2 by adding weld 8-203 and the corresponding RT_{PTS} value to Table 1 of the Millstone Unit 2 FSAR supplement.

Dominion Response:

The requested values are contained in this letter, in Dominion's supplemental response to RAI 3.1.1-3, Table 3.1.1-3-2 Millstone Unit 2 "RT_{PTS} Values at 54 EFPY (Expanded Beltline)".

RAI Supplement 4.7.3-1(a), Unit 2

In response to **RAI 4.7.3-1(a)**, in a letter dated December 3, 2004, the applicant stated that a fracture mechanics evaluation, performed as a part of a Combustion Engineering Owners Group CEN-412, Revision 2, Supplement 2 activity, has been performed for the Millstone Unit 2 reactor coolant pumps. The applicant also stated that for Millstone Unit 2, the limiting end-point crack size is 0.39t, significantly greater than the 1/4t flaw postulated in ASME Code Case N-481. The time for the Millstone Unit 2 reactor coolant pump casing to reach the limiting end-point crack size is 103 years. To confirm the methodology and fracture mechanics results, the applicant is requested to provide the fracture mechanics evaluation.

In a follow-up draft response, the applicant stated that the material's composition was not available and therefore the aged fracture toughness was determined using the procedure outlined in Section 3.1 of NUREG-4315, Rev.1. This approach produced the lower bound aged fracture toughness value that was used in the evaluation.

The staff requests the applicant to provide this lower bound aged fracture toughness value that was calculated and the following information:

1) Is the CASS material ASTM A351?
2) What is the material grade?
3) What is casting method?
4) What is the service temperature?
5) What is the ferrite content and how was it determined?

The applicant also stated that a conservative LEFM was used and the acceptance criteria for the LEFM approach was consistent with IWB-3610 of Section XI of the ASME Code.

To verify this evaluation, the staff requests the following:

1) Limiting stress
2) Limiting transient
3) Maximum flaw size calculated vs. the critical flaw size
4) Stress intensity factors (KI, KIa, and KC)
5) Summary of the evaluation and how the stresses were determined.

Dominion Response:

A copy of Combustion Engineering Owners Group report CEN-412, Revision 2, Supplement 2 *Relaxation of Reactor Coolant Pump Casing Inspection Requirements at*

Millstone Unit 2 is publicly available. A copy was sent to the License Renewal Project Manager as two "pdf" files (CEN-412, part 1 and CEN-412, part 2) on January 27, 2005.

The following information is specific to the Millstone Unit 3 CASS pressurizer spray head, supplemental information contained in Dominion's response to RAI 4.3.1-5.

<u>The following responds to the first set of five questions.</u>

The lower bound aged fracture toughness value is 155.3 (ksi$\sqrt{in.}$). Additional detail on how this value was developed is contained in Dominion's response to the second set of questions (number 5).

1. The spray head is specified as A-296 CF-8M as indicated in Westinghouse Specification 2656A93. However, communication with the manufacturer indicates that one of the cross-reference numbers is A-351 (SA-351). The properties of SA-351 CF-8M were used in the Millstone Unit 3 CASS pressurizer spray head analysis. Comparison of the nominal chemical compositions of A-296 CF-8M and SA-351 CF-8M show a minor difference in maximum silicon content (2.00% for A-296 and 1.50% for SA-351) and no difference in mechanical properties.

2. Material grade: CF-8M S.S.

3. Casting method: sand casting.

4. Service temperature: 650°F.

5. The ferrite content is in the range of 17.5% - 22.5% as determined by specification.

<u>The following responds to the second set of five questions.</u>

1) Limiting stress – a through-wall stress distribution that occurred at 2 seconds into the transient, represented by a third order polynomial for both axial and circumferential stress. Pressure stress is 600 psi axial and 1200 psi circumferential.

2) Limiting transient - 10 cycles of temperature step from 425°F to 70°F back to 425°F after 600 seconds with a 35 gpm flow rate.

3) Maximum flaw size calculated vs. the critical flaw size – not applicable since a stress intensity factor-based criterion was used.

4) Stress intensity factors – Circumferential flaw: total applied K = 39.7 (ksi$\sqrt{in.}$), K_{Ic} = 155.3 (ksi$\sqrt{in.}$), $K_{Ic}/\sqrt{10}$ = 49.1 (ksi$\sqrt{in.}$), $K_{Ic}/\sqrt{2}$ = 109.8 (ksi$\sqrt{in.}$). Axial flaw: total applied K = 36.2 (ksi$\sqrt{in.}$), K_{Ic} = 155.3 (ksi$\sqrt{in.}$), $K_{Ic}/\sqrt{10}$ = 49.1 (ksi$\sqrt{in.}$), $K_{Ic}/\sqrt{2}$ = 109.8 (ksi$\sqrt{in.}$).

5) Evaluation summaries:

- Loads: Loads that were considered included thermal shock (10 cycles of temperature step from 425°F to 70°F back to 425°F after 600 seconds with a 35 gpm flow rate), pressure of 100 psi, and thrust loads as a function of flow rate.

- Stress Analysis: For thermal shock, a finite element analysis was used to determine the critical stress distribution. Hand calculations were used to determine thrust and pressure stress.

- Fracture Toughness: Fracture toughness was determined considering thermal embrittlement and lower bound saturation fracture toughness. A fracture toughness of 155.3 (ksi$\sqrt{in.}$) was determined.

- Flaw Evaluation: Linear elastic fracture mechanics principles were used. Applied K for thermal loading, thrust and pressure was calculated for both axial and circumferential flaws. The applied K was compared to the fracture toughness using a safety factor of $\sqrt{10}$ for normal operating conditions and $\sqrt{2}$ for emergency and faulted conditions.

- Crack Growth: A crack growth evaluation was performed to show that there is minimal growth over the extended life. An ASME Section XI, Appendix C crack growth law was used with an assumed initial flaw size of 12.5% of wall. For both axial and circumferential flaws, the crack growth was insignificant for a 60-year life. The final crack size at 60 years is computed to be 0.0994 inches (40% of wall) for a circumferential flaw and 0.1016 inches (41% of wall) for an axial flaw. The stress intensity factor K corresponding to these final crack sizes is 32.81 (ksi$\sqrt{in.}$) for the circumferential flaw and 37.01 (ksi$\sqrt{in.}$) for the axial flaw. These K values are less than the $K_{Ic}/\sqrt{10}$ limit of 49.1 (ksi$\sqrt{in.}$).

RAI Supplement B2.1.3-2, Units 2 & 3

The staff issued RAI-B2.1.3-2 to assure that the applicant's discussion in its FSAR Supplement summary description for the Borated Water Leakage Assessment and Evaluation Program was consistent with relevant NRC generic communications and the CLB for the plants. The applicant's response to RAI B2.1.3-3b indicates that the applicant will not amended the FSAR Supplement summary description for the Borated Water Leakage Assessment and Evaluation Program to include a reference to the applicant's responses and commitments provided in the applicant's responses to NRC Bulletins 2001-01, 2002-01, 2002-02, 2003-02, and the response to NRC Order EA-03-009, as amended by applicant's the response to the first revision of the Order. The staff finds this unacceptable because the summary description is not current with the CLB for the facilities and does not reference Dominion's responses and commitments to NRC generic communications that are relevant to the scope and implementation of the AMP. It should be noted that the LRA only addressed Generic Letter (GL) 88-05, and NRC Bulletins 2002-1 and 2002-2. NRC Bulletin 2003-02 and NRC Order EA-03-009 were not included in the LRA. Therefore, the staff requests that the applicant amend the FSAR supplement to ensure that the summary description is current with the CLB for the facilities and references Dominion's responses and commitments to NRC generic communications that are relevant to the scope and implementation of the AMP. This is consistent with other applicants such as Farley that have included their responses and commitments to NRC Bulletins 2001-01, 2002-01, 2002-02, 2003-02, and the response to NRC Order EA-03-009.

Dominion Response:

The following will be added after the first paragraph of the Boric Acid Corrosion program description in the Millstone Unit 2 & 3 LRA, Appendix A "FSAR Supplement", Section A2.1.3 [Unit 2] and Section A2.1.2 [Unit 3]:

"Boric Acid Corrosion program implements the requirements of:"

- NRC Bulletin 2001-01 (Reference [A-34 for Unit 2] [A-36 for Unit 3])
- NRC Bulletin 2002-01 (Reference [A-35 for Unit 2] [A-37 for Unit 3])
- NRC Bulletin 2002-02 (Reference [A-36 for Unit 2] [A-38 for Unit 3])
- NRC Bulletin 2003-02 (Reference [A-37 for Unit 2] [A-39 for Unit 3])
- NRC Order EA-03-009 (Reference [A-38 for Unit 2] [A-40 for Unit 3])"

The following references will be added to the Unit 2 & 3 LRA, Appendix A, "FSAR Supplement":

- "[A-34 for Unit 2] [A-36 for Unit 3] NRC Bulletin 2001-01, *Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles*, U.S. Nuclear Regulatory Commission, August 3, 2001
- [A-35 for Unit 2] [A-37 for Unit 3] NRC Bulletin 2002-01, *Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity*, U.S. Nuclear Regulatory Commission, March 18, 2002.
- [A-36 for Unit 2] [A-38 for Unit 3] NRC Bulletin 2002-02, *Reactor Pressure Vessel Head and Vessel Head Penetration Nozzle Inspection Programs*, U.S. Nuclear Regulatory Commission, August 9, 2002.
- [A-37 for Unit 2] [A-39 for Unit 3] NRC Bulletin 2003-02, *Leakage from Reactor Pressure Vessel Lower Head Penetrations and Reactor Coolant Pressure Boundary Integrity*, U.S. Nuclear Regulatory Commission, 08/21/03
- [A-38 for Unit 2] [A-40 for Unit 3] NRC Order EA-03-009, *Issuance Of Order Establishing Interim Inspection Requirements For Reactor Pressure Vessel Heads At Pressurized Water Reactors*, U.S. Nuclear Regulatory Commission, February 11, 2003"

RAI Supplement B2.1.17-1(1), Units 2 & 3

In response to **RAI B2.1.17-1(1)** in a letter dated December 3, 2004, the applicant provided the following:

> The LRAs for Millstone Units 2 and 3 (Appendix A, Table A6.0-1, commitment 13) identify that Millstone will follow the industry efforts on reactor vessel internals regarding such issues as thermal or neutron irradiation embrittlement (loss of fracture toughness), void swelling, stress corrosion cracking (PWSCC and IASCC), and for the Millstone Unit 3 commitment only, loss of pre-load for the baffle and former-assembly bolts. Dominion provided a supplemental response applicable to commitment 13 for both Millstone Unit 2 and 3 as documented in the Dominion letter (Serial Number 04-320) dated July 7, 2004 (Audit Item Number 6). The supplemental response letter identifies that the statement, "The revised program description, including a comparison to the 10 program elements of the NUREG-1801 program, will be submitted to the NRC for approval." should be inserted at the end of the [current] commitment. Appendix A, Table A6.0-1 for both the Unit 2 LRA and the Unit 3 LRA already states that commitment 13 is scheduled to be completed prior to the period of extended operation. The supplemental response letter also identifies the other applicable locations in both the Unit 2 LRA and the Unit 3 LRA where this additional wording should be inserted.

The staff finds this commitment unacceptable since the applicant has not specifically committed to submit the program "three years" prior to the period of extended operation in order for the NRC to review and approve the program prior to its implementation at the facility during the period of extended operation. Therefore, the applicant is requested to revise commitment 13 of Appendix A, Table A6.0-1 of the Millstone, Units 2 and 3, LRAs to state that the revised program implementing the industry efforts on reactor vessel internals will be submitted to the NRC for approval "three years" prior to the period of extended operation.

Dominion Response:

As a result of more recent guidance provided during a teleconference between Dominion and the NRC on January 27, 2005, the schedule for Table 6.0-1, Commitment 13 in Millstone Units 2 and 3 LRA Appendix A, "FSAR Supplement" will be changed to:

> "At Least Two Years Prior to Period of Extended Operation"

RAI Supplement B2.1.17-1(3), Unit 3

The staff also requested in **RAI B2.1.17-1(3)**, that the applicant include loss of preload in List of Commitments, Table A6.0-1 in Appendix A of the Millstone Units 2 and 3 LRA to fully describe all of the necessary aging effects and their management. In response to **RAI B2.1.17-1(3)** in a letter dated December 3, 2004, the applicant that for both Millstone Units 2 and 3, loss of pre-load is an applicable aging effect that is managed by the Inservice Inspection Program: Reactor Vessel Internals Program for bolting used in the reactor vessel. The Millstone Unit 3 LRA (Appendix A, Table A6.0-1, commitment 13) identifies that Millstone Unit 3 will follow the industry efforts on the loss of pre-load for the baffle and former assembly bolts. This is applicable to Millstone Unit 3 only since Millstone Unit 2 is a Combustion Engineering design, and therefore the aging management of the baffle and former assembly bolts are not applicable. The staff finds this acceptable since the bolting in the reactor vessel internals for Millstone Units 2 and 3 will be inspected in accordance with the ASME Code, Section XI, and the baffle and former assembly bolts in Millstone Unit 3 will be have augmented inspections performed. This augmented inspection will be based on industry efforts and will be submitted to the NRC for approval prior to entering the period of extended operation. Since the proposed augmented inspection has not be developed or approved, the staff requests the applicant to commit to submit this inspection plan to the NRC for approval three years prior to entering the extend period.

Dominion Response:

As a result of more recent guidance from the NRC, the schedule for Table 6.0-1, Commitment 13 in Millstone Units 2 and 3 LRA Appendix A, "FSAR Supplement" will be changed to:

"At Least Two Years Prior to Period of Extended Operation"

RAI Supplement B2.1.17-2, Unit 3

In response to **RAI B2.1.17-2** in a letter dated December 3, 2004, the applicant stated that currently, the exact examination method, acceptance criteria and frequency of inspections are in the process of being determined. Currently, commitment 14 of Table A6.0-1 of the Millstone Unit 3 LRA states that the proposed inspection will detect gross indication of loss of preload as an aging effect and be performed prior to the period of extended operation. However, the applicant has stated that as an alternative to performing an augmented inspection, the holddown spring may be replaced prior to the period of extended operation. Therefore, the applicant will include the following statement in commitment 14 of the Millstone Unit 3 LRA, "As an alternative to performing an augmented inspection, the holddown spring will be replaced prior to the period of extended operation." Since the proposed augmented inspection has not been developed or approved, the staff requests the applicant to commit to submit this inspection plan to the NRC for approval three years prior to entering the extend period or commit to replace the holddown springs three years prior to entering the extended period.

Dominion Response:

In the response to RAI B2.1.17-2, Dominion committed to add the words, "As an alternative to performing an augmented inspection, the holddown spring will be replaced prior to the period of extended operation.", following the commitment in Millstone Unit 3 LRA Appendix A, "FSAR Supplement", Section A2.1.17 and Table A6.0-1, Commitment 14. This addition will be reworded to state:

> "As an alternative to performing an augmented inspection, the holddown spring will be replaced."

As a result of more recent guidance from the NRC, the schedule for Table 6.0-1, Commitment ,14 in Millstone Unit 3 LRA Appendix A, "FSAR Supplement" will be changed to:

> "At Least Two Years Prior to Period of Extended Operation"

RAI Supplement B2.1.18-1, Unit 2

In response to **RAI B2.1.18-1** in a letter dated December 3, 2004, the applicant stated Dominion is intending to replace the pressurizer during the Fall of 2006 refueling outage for Millstone, Unit 2 using materials that are resistant to PWSCC, as documented in their letter dated June 3, 2004. To track this commitment, the applicant is requested to revise the List of Commitments (Table A6.0-1 of Appendix A to the Millstone Unit 2 LRA) to include the commitment that the Millstone Unit 2 pressurizer will be replaced in Fall 2006 with material resistant to PWSCC (i.e. Alloy 690 and 52/152).

Dominion Response:

This RAI supplemental question was discussed in a phone conversation between the staff and Dominion on January 25, 2005. In order to meet the staff needs and to retain some scheduler flexibility, it was agreed that the following commitment will be added to the Millstone Unit 2 LRA Appendix A, "FSAR Supplement", Section A2.1.18:

"• Pressurizer Replacement

"Dominion will replace the Millstone Unit 2 pressurizer using materials that are resistant to PWSCC. This commitment is identified in Appendix A, Table A6.0-1 License Renewal Commitments, Item 36".

An additional item will be added to Millstone Unit 2 LRA Appendix A, "FSAR Supplement", Table A6.0 -1 as follows:

Item: 36

Commitment: "Dominion will replace the Millstone Unit 2 pressurizer using materials that are resistant to PWSCC."

Source: "Inservice Inspection Program: Systems, Components and Supports."

Schedule: "Prior to Period of Extended Operation"

RAI Supplement B2.1.18-3(1), Unit 3

In response to **RAI B2.1.18-3** in a letter dated December 3, 2004, the applicant credits the Water Chemistry AMP for controlling contaminants to reduce the potential of stress corrosion cracking in the Flux Thimble Tubes and the Guide Tubes. For the Flux Thimble Tubes, the applicant also credits the existing inspection of the seal table pressure boundary during each refueling outage in accordance with their inservice inspection program. However, the applicant has not specified the type of inspection (i.e. visual inspection or ultrasonic). Therefore, the applicant is requested to provide the type of inspection, inspection frequency and acceptance criteria that will be used to detect stress corrosion cracking in the Flux Thimble Tubes.

Dominion Response:

As discussed in the response to RAI B2.1.18-3, stress corrosion cracking is not expected to occur in the BMI flux thimble tubes due to the existence of primarily compressive loading and the chemistry-controlled water environment. However, cracking due to SCC was conservatively determined to be an aging effect requiring management to ensure that the environment was maintained non-conducive to the aging effect. The Chemistry Control for Primary Systems Program AMP is identified to manage the water chemistry for the thimble tube environment. As an added measure to confirm that the water chemistry program is effective, inspection of the seal table pressure boundary via the Inservice Inspection Program: Systems, Components, and Supports AMP is credited. The seal table pressure boundary is subjected to a VT-2 visual examination during the system leakage test performed at normal operating temperature and pressure, in accordance with Examination Category B-P of the ASME XI, Subsection IWB, each refueling cycle. This examination provides confirmation that cracking due to SCC is not occurring in the thimble tubes as evidenced by the lack of leakage from the seal table area. The acceptance criteria for the examination is no signs of leakage, and any indications of leakage would be evaluated through the corrective action system and the cause determined.

Operating experience related to the thimble tubes at Millstone Unit 3 has been reviewed and no occurrences of SCC were identified. In addition, there is no known operating experience with SCC of thimble tubes having occurred in the nuclear industry.

Based on the minimal concern for SCC of the BMI flux thimble tubes due to the primary water environment and the limited operating stresses, and the lack of operating experience to support a concern for cracking of these components, management of cracking by the Chemistry Control for Primary Systems Program AMP and the Inservice

Inspection Program: Systems, Components, and Supports AMP provides reasonable assurance that the pressure boundary intended function will be maintained through the period of extended operation.

RAI Supplement B2.1.18-3(2), Unit 3

In response to **RAI B2.1.18-3** in a letter dated December 3, 2004, for the Guide Tubes, the applicant credits the Water Chemistry AMP for reducing the potential for stress corrosion cracking and the Inservice Inspection Program: Systems, Components, and Supports AMP for inspecting the most susceptible location to stress corrosion cracking, which is the weld between the BMI Guide Tubes and Instrumentation Tubes on the reactor vessel bottom head, in Table 3.1.2-1 of the LRA. To determine if the inspections of the Inservice Inspection AMP is capable of managing SCC in the Guide tubes the applicant is requested to provide the following:

- specify the type of inspection or the inspection frequency.
- In addition, if indications in this weld are found, what increase in the sampling will be performed since this is being used as an indicator that SCC is occurring?
- Also, the applicant stated that the reduced temperature from that of the RCS operating temperature reduces the potential for SCC. What temperatures does the Guide Tubes experience?
- Generic Letter 88-01 indicates that at temperatures below 200°F stainless steel components are not susceptible to SCC. If the temperature of the Guide Tubes is above 200°F, the potential for SCC is not reduced, and the applicant is requested to determine whether the inspection frequency is acceptable to detect cracking of the guide tube.

Dominion Response:

Stress corrosion cracking of the BMI Guide Tubes is not expected to occur based on the chemistry-controlled water environment. However, cracking due to SCC was conservatively identified as an aging effect requiring management to ensure that the environment was maintained non-conducive to the aging effect. The Chemistry Control for Primary Systems Program AMP is identified to manage the water chemistry for the guide tube environment. As confirmation that the water chemistry program is effective, inspections of the guide tubes are performed by the Inservice Inspection Program: Systems, Components, and Supports AMP. The guide tubes pressure boundary is subjected to a VT-2 visual examination during the system leakage test performed at normal operating temperature and pressure, in accordance with Examination Category B-P of the ASME XI, Subsection IWB, each refueling cycle. In addition, the BMI Guide Tubes are welded to the Instrumentation Tubes, as described in the response to RAI B2.1.18-3, which penetrate the reactor vessel. These components are closer to the reactor vessel than the BMI Guide Tubes and would experience higher temperature conditions. The nickel-based alloy Instrumentation Tubes are managed for cracking, as indicated in LRA Table 3.1.2-1, by the Inservice Inspection Program: Systems, Components, and Supports AMP. Management of Instrumentation Tubes aging includes a bare metal visual examination of the reactor vessel bottom head area as documented in Dominion letter S/N 03-459A dated November 17, 2003. This inspection will be performed each refueling outage and the acceptance criteria is no evidence of leakage, and any indications of leakage would be evaluated through the corrective action system and the cause determined. Leakage in this area would result in further examinations to determine the extent of the condition.

In addition to these bare metal visual examinations, the Millstone Unit 3 Technical Specifications require monitoring of reactor coolant system leakage during plant operation. The established limits are one gallon per minute (gpm) for unidentified leakage, ten gpm for identified leakage, and no leakage from the reactor coolant system pressure boundary.

Operating experience related to the guide tubes at Millstone Unit 3 has been reviewed and no occurrences of SCC were identified. In addition, there is no known operating experience with SCC of guide tubes having occurred in the nuclear industry.

Note that, in LRA Table 3.1.2-1, cracking is an aging effect that is conservatively applied for the entire guide tube and temperature considerations were not used to conclude that cracking does not require management for a portion of the component, only to determine that the portion nearest the reactor vessel is the most susceptible to SCC.

RAI Supplement B2.1.18-5, Unit 3

In response to **RAI B2.1.18-5** in a letter dated December 3, 2004, the applicant stated that Millstone Unit 2 follows the recommendations of RG 1.65. However, for Millstone Unit 3, the applicant's stated that the closure bolting for Unit 3 uses Plasma Bond coating (Nickel-Silver/Palladium). RG 1.65 states that silver plated studs had severe galling and severe corrosion damage in the thread roots of the studs at LaCrosse (BWR) and Yankee Rowe. Therefore, in accordance with RG 1.65, section C.1.b(3), the applicant is requested to demonstrate that the plating will not degrade the quality of the material in any significant way (e.g., corrosion, H_2 embrittlement) or reduce the quality of results attainable by the various required inspection procedures.

Dominion Response:

Regulatory Guide 1.65 identifies preventive measures to mitigate cracking of reactor vessel closure bolting. Preventive measures include the use of manganese phosphate or other acceptable surface treatments and stable lubricants, and avoiding the use of metal-plated stud bolting, which is susceptible to degradation due to corrosion or hydrogen embrittlement.

Consistent with the objectives of Regulatory Guide 1.65, Millstone Unit 3 utilizes a PlasmaBond coating that is applied to the threaded portions of the studs as an acceptable alternative to the manganese phosphate coating. PlasmaBond was developed and tested by Westinghouse (with Texas Utilities) for use on vessel head closure studs and other locations such as steam generator manway studs. This newer antigalling coating was added to provide for enhanced lubrication. The coating has no adverse metallurgical interactions, and will not affect the base metal physical properties. Formerly identified as Mag-Ion, PlasmaBond is a Nickel-Silver/Palladium coating that uses a vapor deposition process (not electrolytic). Therefore, there is no hydrogen generation, and no potential for hydrogen embrittlement of the fastener.

Since the approval of the PlasmaBond coating for Millstone Unit 3, the studs were examined under the Inservice Inspection Program in February of 2001 (3R07). No indications were identified as part of the volumetric and magnetic particle examinations performed. The next scheduled ISI examination is 3R10.

To date PlasmaBond has been used in numerous applications at various nuclear power plants. PlasmaBond has been specifically used for reactor vessel studs at Comanche Peak Units 1 and 2, Catawaba Unit 2, Beaver Valley Units 1 and 2, and Seabrook. A review of industry operating experience identified no examples of coating issues with PlasmaBond. Comanche Peak has had the most operating experience with PlasmaBond, and has completed six operating cycles without any degradation of the studs due to PlasmaBond. The coated surfaces show no signs of flaking or disbondment.

In conclusion, the PlasmaBond *coating is an approved* Westinghouse *coating for use on* the Millstone Unit 3 reactor vessel closure head studs. This coating improves antigalling and does not increase corrosion attack susceptibility or introduce any new material degradation mechanisms. The PlasmaBond coating process precludes degradation due to hydrogen embrittlement, has no effect on ultrasonic, magnetic particle, or dye penetrant techniques, and will not mask any defects.