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June 10, 2010
L-10-025

10 CFR 50.55a

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

SUBJECT:
Beaver Valley Power Station, Unit Nos. 1 and 2
Docket No. 50-334, License No. DPR-66
Docket No. 50-412, License No. NPF-73
10 CFR 50.55a Requests for Alternative Non-Destructive Examination Requirements for ASME Class 1 and Class 2 Piping Components

Pursuant to 10 CFR 50.55a(a)(3), FirstEnergy Nuclear Operating Company (FENOC) is requesting Nuclear Regulatory Commission (NRC) approval for continued use of the existing Beaver Valley Power Station, Unit No. 1 (BVPS-1) and Unit No. 2 (BVPS-2), risk-informed inservice inspection (RI-ISI) program, with updates, relevant to certain non-destructive examination (NDE) requirements associated with American Society of Mechanical Engineers (ASME) Class 1 and Class 2 piping components.

Proposed alternative RI-ISI-1, included as Enclosure A, would be implemented during the BVPS-1 fourth ISI interval. FENOC is requesting approval of alternative RI-ISI-1 by February 1, 2011 to support the scope freeze milestone for the BVPS-1 April 2012 refueling outage.

Proposed alternative RI-ISI-2, included as Enclosure B, would be implemented during the BVPS-2 third ISI interval. FENOC is requesting approval of alternative RI-ISI-2 by July 1, 2011 to support the scope freeze milestone for the BVPS-2 September 2012 refueling outage.

Pursuant to Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 1, a summary of the BVPS-1 and BVPS-2 probabilistic risk assessment model's capability for use in RI-ISI program activities and initiatives, is provided as Enclosure C.

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There are no regulatory commitments contained in this submittal. If there are any questions or additional information is required, please contact Mr. Thomas A. Lentz, Manager – Fleet Licensing, at (330) 761-6071.

Sincerely,

Paul A. Harden

Enclosures:

A. Beaver Valley Power Station Unit No. 1, 10 CFR 50.55a Request RI-ISI-1, Revision 0
B. Beaver Valley Power Station Unit No. 2, 10 CFR 50.55a Request RI-ISI-2, Revision 0
C. FirstEnergy Nuclear Operating Company, Beaver Valley Power Station Unit Nos. 1 and 2, Probabilistic Risk Assessment Technical Adequacy for Risk-Informed Inservice Inspection

cc: NRC Region I Administrator
NRC Resident Inspector
NRC Project Manager
Director BRP/DEP
Site BRP/DEP Representative

Proposed Alternative
in Accordance with 10 CFR 50.55a(a)(3)(i)

--Alternative Provides Acceptable Level of Quality and Safety--

1. **ASME Code Component(s) Affected**

   ASME Code Class 1 and 2 piping welds as listed in Table 1

2. **Applicable Code Edition and Addenda**

   ASME Code Section XI, 2001 Edition, 2003 Addenda

3. **Applicable Code Requirements**

   ASME Code Section XI, 2001 Edition, 2003 Addenda, Inservice Inspection (ISI) requirements for pressure retaining piping welds

   IWB-2500, Examination and Pressure Test Requirements
   Table IWB-2500-1, Examination Categories
   Class 1 Piping Welds

   - Category B-F, Pressure Retaining Dissimilar Metal Welds in Vessel Nozzles
   - Category B-J, Pressure Retaining Welds in Piping

   IWC-2500, Examination and Pressure Test Requirements
   Table IWC-2500-1, Examination Categories
   Class 2 Piping Welds

   - Category C-F-1, Pressure Retaining Welds in Austenitic Stainless Steel or High Alloy Piping
   - Category C-F-2, Pressure Retaining Welds in Carbon or Low Alloy Steel Piping

4. **Reason for Request**

   On April 9, 2004, Nuclear Regulatory Commission (NRC) staff approved FENOC's ASME Code Section XI Class 1 and Class 2 Risk-Informed Inservice Inspection (RI-ISI) Program for Beaver Valley Power Station Unit No. 1 (BVPS-1), third ISI interval.

   In its approval, NRC staff concluded the RI-ISI program is consistent with WCAP-14572, "Westinghouse Owners Group Application of Risk-Informed Methods to Piping Inservice Inspection Topical Report," Revision 1-NP-A, and is an acceptable alternative to the requirements of ASME Code , Section XI, for inservice inspection of ASME Class 1 and 2 piping, examination categories B-F, B-J, C-F-1, and C-F-2.

On May 1, 2006, NRC staff approved the Pressurized Water Reactor (PWR) Owners Group Topical Report WCAP-14572, Revision 1-NP-A, Supplement 2. The safety evaluation report [Reference 1] states:

> "The NRC staff concludes that the proposed RI-ISI program as described in the approved WCAP-14572, and WCAP-14572, Sup. 2, as clarified and revised by the June 22, 2005, supplemental letter, will provide an acceptable level of quality and safety with regard to the number of inspections, locations of inspections, and methods of inspection."

Consistent with the RI-ISI methodology documented in WCAP-14572, including its supplements, [References 2, 3, 4], new information has been incorporated into the RI-ISI analysis as part of the "living" RI-ISI program. The new information includes changes to the BVPS-1 Probabilistic Risk Assessment (PRA) model, revised segments and failure probabilities for some segments based on industry and plant experience and plant modifications, revised consequences based on lessons-learned, and updated test intervals for certain segments and overlays of pressurizer alloy 82/182 welds in the reactor coolant system.

The changes described above required re-performing the risk evaluation. The revised results were reviewed by the RI-ISI expert panel. Compared to the third interval BVPS-1 ISI program, 26 low safety significant (LSS) segments were reclassified as high safety significant (HSS), and 24 HSS segments were reclassified as LSS; 3 quantitative HSS segments were re-categorized by the expert panel as LSS based on the "with operator action consequences" guidance within WCAP-14572, Supplement 2. The expert panel concluded the remaining segment classifications shall remain as-is.

The change in risk evaluation was performed again to compare the original Section XI program with the revised fourth interval RI-ISI program for BVPS-1. Five reactor coolant system segments and one safety injection system segment (six total segments) are to be added to the BVPS-1 RI-ISI program to meet the change in risk criteria discussed in WCAP-14572, page 214. These six additional examinations are VT-2 visual exams.

No examinations were added for defense-in-depth considerations, which is the same as in the previously approved third interval RI-ISI program.

The proposed RI-ISI program, with updates, provides a 77 percent reduction in required examinations. This directly results in reduced outage scope, decreased individual and cumulative occupational radiation exposure, and shortened outage durations. As such, FENOC requests that the BVPS-1 RI-ISI program, with updates, be approved for continued use during the fourth ISI interval.

5. Proposed Alternative and Basis for Use

ASME Section XI categories B-F, B-J, C-F-1, and C-F-2 contain the requirements for examining Class 1 and 2 piping components via non-destructive examination (NDE). The proposed alternative [continued use of the BVPS-1 RI-ISI program, with updates] is limited to ASME Class 1 and 2 piping components, including piping currently exempt from NDE requirements. The proposed alternative will be substituted for the ASME

Section XI category B-F, B-J, C-F-1, and C-F-2 examination requirements. The applicable aspects of ASME Section XI Code not affected by the proposed alternative will be retained.

The basis of the alternative risk-informed inservice inspection program's methodology is fully described in the NRC-endorsed WCAP-14572 and its supplements.

Pursuant to 10 CFR 50.55a(a)(3)(i), the proposed alternative to the ASME Code Section XI examination requirements will continue to provide an acceptable level of quality and safety.

Comparisons of the ASME Section XI inspection program, the third interval RI-ISI program, and the proposed fourth interval RI-ISI program are presented in Table 1.

6. Duration of Proposed Alternative

The proposed alternative shall be implemented during the BVPS-1 fourth ten-year ISI interval and will remain effective until the end of the interval on March 31, 2018.

7. Precedents

NRC letter to FENOC, April 9, 2004, Subject: Beaver Valley Power Station, Unit Nos. 1 and 2 (BVPS-1 and 2) – Risk-Informed Inservice Inspection (RI-ISI) Program.
[ADAMS Accession Number ML040780805]

NRC letter to Tennessee Valley Authority, April 30, 2007, Subject: Sequoyah Nuclear Plant, Units 1 and 2 – Risk-Informed Inservice Inspection Program for the Third 10-Year Intervals.
[ADAMS Accession Number ML071070248]

8. References

1. Nuclear Regulatory Commission (NRC), "Final Safety Evaluation by the Office of Nuclear Reactor Regulation Topical Report WCAP-14572, Revision 1-NP-A, Supplement 2, 'Westinghouse Owners Group Application of Risk-Informed Methods to Piping Inservice Inspection Topical Report Clarifications' Pressurized Water Reactor (PWR) Owners Group Project No. 694," May 1, 2006.
[ADAMS Accession No. ML061160035]

2. Westinghouse Electric, WCAP-14572, "Westinghouse Owners Group Application of Risk-Informed Methods to Piping Inservice Inspection Topical Report," Revision 1-NP-A, February 1999.

3. Westinghouse Electric, WCAP-14572, Supplement 1, "Westinghouse Structural Reliability and Risk Assessment (SRRA) Model for Piping Risk-Informed Inservice Inspection," Revision 1-NP-A, February 1999.

4. Westinghouse Electric, WCAP-14572, Supplement 2, "Pressurized Water Reactor Owners Group Application of Risk-Informed Methods to Piping Inservice Inspection Topical Report Clarifications," Revision 1-NP-A, September 2006.

Table 1
BVPS-1 STRUCTURAL ELEMENT SELECTION
RESULTS AND COMPARISON TO ASME SECTION XI
1989 EDITION REQUIREMENTS

System	High Safety Significant Segments (Qty. of HSS in Augmented Program / Total Qty. of Segments in Aug. Program)	Degradation Mechanism(s)	Safety Class	ASME Code Exam Category	Total Weld Count [Welds requiring Volumetric (Vol) and Surface (Sur)]		ASME XI Program Examinations		Third Interval RI-ISI		Fourth Interval RI-ISI[a]	
					Vol and Sur	Sur only	Vol and Sur	Sur only	SES Matrix Region	Number of Exam Locations	SES Matrix Region	Number of Exam Locations
BD	0 (0 / 27[c])	FAC/TF	Class 2	N/A	0	0	0	0	3	0	3	0
CH	28 (0 / 0)	TF/VF, TF	Class 1	B-J	25	287	7	64	1, 2, 3, 4	0	1, 2, 3, 4	3[e]
			Class 2	C-F-1	317	303	17	18		8 + 19[b] + 2[e]		8 + 17[e]
CI	0 (0 / 0)	FAC/TF, TF	Class 2	N/A	0	0	0	0	4	0	4	0
DV	0 (0 / 0)	TF	Class 1	B-J	0	106	0	27	4	0	4	0
FW	0 (0 / 27[c])	FAC/TF	Class 2	C-F-2	62	0	14	0	3	0	3	0
HY	0 (0 / 0)	TF	Class 2	N/A	0	0	0	0	4	0	4	0
MS	8 (8 / 48[c])	FAC/TF	Class 2	C-F-2	106	0	23	0	1, 3	8	1, 3, 4	8
QS	5 (0 / 0)	TF, VF	Class 2	C-F-1	157	50	12	4	2, 3, 4	3	2, 3, 4	19
RC	23 (0 / 0)	SCC/TF,	Class 1	B-F	18	0	18	0	2, 4	7	2, 4	23 + 5[d]
		SCC/TF/VF/SS	Class 1	B-J	207	181	55	53		13+ 2[d]		
RH	19 (0 / 0)	TF, TF/VF	Class 1	B-J	26	0	6	0	2, 3, 4	2	2, 3, 4	2
			Class 2	C-F-1	177	0	14	0		15 + 2[b]		15 + 2[e]
RS	10 (0 / 0)	TF, VF	Class 2	C-F-1	84	14	7	2	2, 4	10	2, 4	10
SI	37 (0 / 0)	TF	Class 1	B-J	193	108	43	31	1, 2, 4	11 +4[b] + 1[d]	2, 4	12 + 3[e]
			Class 2	C-F-1	826	147	70	16		16 + 1[d]		18 + 5[e] + 1[d]
SS	0 (0 / 0)	TF	Class 1	N/A	0	0	0	0	4	0	4	0
			Class 2	N/A	0	0	0	0		0		0
TOTAL	130 (8 / 102)	FAC/TF, TF, SCC/TF, SCC/TF/VF/SS, TF/VF, VF	Class 1		469	682	129	175		38 NDE + 3 VISUAL		43 NDE + 12 VISUAL
			Class 2		1729	514	157	40		55 NDE + 28 VISUAL		72 NDE + 24 VISUAL
			Total		2198	1196	286	215		93 NDE + 31 VISUAL		115 NDE + 36 VISUAL

Table 1
BVPS-1 STRUCTURAL ELEMENT SELECTION
RESULTS AND COMPARISON TO ASME SECTION XI
1989 EDITION REQUIREMENTS

Summary: ASME Section XI selected a total of 501 welds while the proposed RI-ISI program selects a total of 115 welds (plus 36 visual exams), which results in a 77% reduction.

Degradation Mechanisms:
FAC – Flow-Assisted Corrosion
SCC – Stress Corrosion Cracking
SS – Striping/Stratification
TF – Thermal Fatigue
VF – Vibratory Fatigue
"X/X" indicates combination of mechanisms.

Systems:
BD – Steam Generator Blowdown System
CH – Chemical and Volume Control System
CI – Containment Isolation System
DV – Reactor Plant Drains and Vents Systems
FW – Steam Generator Feedwater System
HY – Hydrogen Control System
MS – Main Steam System
QS – Quench Spray System
RC – Reactor Coolant System
RH – Residual Heat Removal System
RS – Recirculation Spray System
SI – Safety Injection System
SS – Sampling System

Notes for Table 1
a. System pressure test requirements and VT-2 visual examinations shall continue in all ASME Code Class systems.
b. VT-2 visual examination at one location within segment.
c. Augmented programs for erosion-corrosion and high energy line break continue.
d. Examinations added for change in risk considerations (total of six segments- five RC and one SI).
e. VT-2 visual examination for entire segment.

Proposed Alternative
in Accordance with 10 CFR 50.55a(a)(3)(i)

--Alternative Provides Acceptable Level of Quality and Safety--

1. **ASME Code Component(s) Affected**

   ASME Code Class 1 and 2 piping welds as listed in Table 1

2. **Applicable Code Edition and Addenda**

   ASME Code Section XI, 2001 Edition, 2003 Addenda

3. **Applicable Code Requirements**

   ASME Code Section XI, 2001 Edition, 2003 Addenda, Inservice Inspection (ISI) requirements for pressure retaining piping welds

   IWB-2500, Examination and Pressure Test Requirements
   Table IWB-2500-1, Examination Categories
   Class 1 Piping Welds

   - Category B-F, Pressure Retaining Dissimilar Metal Welds in Vessel Nozzles
   - Category B-J, Pressure Retaining Welds in Piping

   IWC-2500, Examination and Pressure Test Requirements
   Table IWC-2500-1, Examination Categories
   Class 2 Piping Welds

   - Category C-F-1, Pressure Retaining Welds in Austenitic Stainless Steel or High Alloy Piping
   - Category C-F-2, Pressure Retaining Welds in Carbon or Low Alloy Steel Piping

4. **Reason for Request**

   On April 9, 2004, Nuclear Regulatory Commission (NRC) staff approved FENOC's ASME Code Section XI Class 1 and Class 2 Risk-Informed Inservice Inspection (RI-ISI) Program for Beaver Valley Power Station Unit No. 2 (BVPS-2), second ISI interval.

   In its approval, NRC staff concluded the RI-ISI program is consistent with WCAP-14572, "Westinghouse Owners Group Application of Risk-Informed Methods to Piping Inservice Inspection Topical Report," Revision 1-NP-A, and is an acceptable alternative to the requirements of ASME Code , Section XI, for inservice inspection of ASME Class 1 and 2 piping, examination categories B-F, B-J, C-F-1, and C-F-2.

On May 1, 2006, NRC staff approved the Pressurized Water Reactor (PWR) Owners Group Topical Report WCAP-14572, Revision 1-NP-A, Supplement 2. The safety evaluation report states:

> "The NRC staff concludes that the proposed RI-ISI program as described in the approved WCAP-14572, and WCAP-14572, Sup. 2, as clarified and revised by the June 22, 2005, supplemental letter, will provide an acceptable level of quality and safety with regard to the number of inspections, locations of inspections, and methods of inspection."

Consistent with the RI-ISI methodology documented in WCAP-14572, including its supplements, [References 2, 3, 4], new information has been incorporated into the RI-ISI analysis as part of the "living" RI-ISI program. The new information includes changes to the BVPS-2 Probabilistic Risk Assessment (PRA) model, revised segments and failure probabilities for some segments based on industry and plant experience and plant modifications, revised consequences based on lessons-learned, and updated test intervals for certain segments and overlays of pressurizer alloy 82/182 welds in the reactor coolant system.

The changes described above required re-performing the risk evaluation. The revised results were reviewed by the RI-ISI expert panel. Compared to the second interval BVPS-2 ISI program, 48 low safety significant (LSS) segments were reclassified as high safety significant (HSS), and 6 HSS segments were reclassified as LSS: 7 quantitative HSS segments were re-categorized by the expert panel as LSS based on the "with operator action consequences" guidance within WCAP-14572, Supplement 2. The expert panel concluded the remaining segment classifications shall remain as-is.

The change in risk evaluation was performed again to compare the original Section XI program with the revised third interval RI-ISI program for BVPS-2. Three reactor coolant system segments and six safety injection system segment (nine total segments) are to be added to the BVPS-2 RI-ISI program to meet the change in risk criteria discussed in WCAP-14572, page 214. These nine additional examinations are VT-2 visual exams.

No examinations were added for defense-in-depth considerations, which is the same as in the previously approved second interval RI-ISI program.

The proposed RI-ISI program, with updates, provides a 78 percent reduction in required examinations. This directly results in reduced outage scope, decreased individual and cumulative occupational radiation exposure, and shortened outage durations. As such, FENOC requests that the BVPS-2 RI-ISI program, with updates, be approved for continued use during the third ISI interval.

5. Proposed Alternative and Basis for Use

ASME Section XI categories B-F, B-J, C-F-1, and C-F-2 contain the requirements for examining Class 1 and 2 piping components via non-destructive examination (NDE). The proposed alternative [continued use of the BVPS-2 RI-ISI program, with updates] is limited to ASME Class 1 and 2 piping components, including piping currently exempt from NDE requirements. The proposed alternative will be substituted for the ASME Section XI category B-F, B-J, C-F-1, and C-F-2 examination requirements.

The applicable aspects of ASME Section XI Code not affected by the proposed alternative will be retained.

The basis of the alternative risk-informed inservice inspection program's methodology is fully described in the NRC-endorsed WCAP-14572 and its supplements.

Pursuant to 10 CFR 50.55a(a)(3)(i), the proposed alternative to the ASME Code Section XI examination requirements will continue to provide an acceptable level of quality and safety.

Comparisons of the ASME Section XI inspection program, the second interval RI-ISI program, and the proposed third interval RI-ISI program are presented in Table 1.

6. Duration of Proposed Alternative

The proposed alternative shall be implemented during the BVPS-2 third ten-year ISI interval and will remain effective until the end of the interval on August 28, 2018.

7. Precedents

NRC letter to FENOC, April 9, 2004, Subject: Beaver Valley Power Station, Unit Nos. 1 and 2 (BVPS-1 and 2) – Risk-Informed Inservice Inspection (RI-ISI) Program. [ADAMS Accession Number ML040780805]

NRC letter to Tennessee Valley Authority, April 30, 2007, Subject: Sequoyah Nuclear Plant, Units 1 and 2 – Risk-Informed Inservice Inspection Program for the Third 10-Year Intervals.
[ADAMS Accession Number ML071070248]

8. References

1. Nuclear Regulatory Commission (NRC), "Final Safety Evaluation by the Office of Nuclear Reactor Regulation Topical Report WCAP-14572, Revision 1-NP-A, Supplement 2, 'Westinghouse Owners Group Application of Risk-Informed Methods to Piping Inservice Inspection Topical Report Clarifications' Pressurized Water Reactor (PWR) Owners Group Project No. 694," May 1, 2006. [ADAMS Accession No. ML061160035]

2. Westinghouse Electric, WCAP-14572, "Westinghouse Owners Group Application of Risk-Informed Methods to Piping Inservice Inspection Topical Report," Revision 1-NP-A, February 1999.

3. Westinghouse Electric, WCAP-14572, Supplement 1, "Westinghouse Structural Reliability and Risk Assessment (SRRA) Model for Piping Risk-Informed Inservice Inspection," Revision 1-NP-A, February 1999.

4. Westinghouse Electric, WCAP-14572, Supplement 2, "Pressurized Water Reactor Owners Group Application of Risk-Informed Methods to Piping Inservice Inspection Topical Report Clarifications," Revision 1-NP-A, September 2006.

Table 1
BVPS-2 STRUCTURAL ELEMENT SELECTION
RESULTS AND COMPARISON TO ASME SECTION XI
1989 EDITION REQUIREMENTS

System	High Safety Significant Segments (Qty. of HSS in Augmented Program / Total Qty. of Segments in Aug. Program)	Degradation Mechanism(s)	Safety Class	ASME Code Exam Category	Total Weld Count [Welds requiring Volumetric (Vol) and Surface (Sur)]		ASME XI Program Examinations		Second Interval RI-ISI		Third Interval RI-ISI[a]	
					Vol and Sur	Sur only	Vol and Sur	Sur only	SES Matrix Region	Number of Exam Locations	SES Matrix Region	Number of Exam Locations
BDG	3 (3 / 24[c])	FAC/TF	Class 2	N/A	0	0	0	0	3	0	1, 3	3 + 3[f]
CHS	34 (0 / 0)	TF/VF, TF	Class 1	B-J	4	369	3	57	2, 3, 4	0	1, 2, 3, 4	0
			Class 2	C-F-1	343	315	26	27		19 + 14[b]		19 + 15[g]
CI	4 (0 / 0)	TF/SCC, TF	Class 2	N/A	0	0	0	0	4	0	2, 4	4
DAS	0 (0 / 0)	TF	Class 1	B-J	0	36	0	24	4	0	4	0
FWA	0 (0 / 57[c])	FAC/TF	Class 2	C-F-2	56	0	9	0	3	0	3	0
GNS	0 (0 / 0)	TF	Class 2	N/A	0	0	0	0	4	0	4	0
HCS	0 (0 / 0)	TF	Class 2	N/A	0	0	0	0	4	0	4	0
MSS	15 (9 / 44[c])	FAC/TF, TF	Class 2	C-F-2	136	3	17	0	1, 3	8	1, 2, 3, 4	12 + 3[g]
QSS	15 (0 / 0)	TF, VF	Class 2	C-F-1	200	0	16	0	1, 2, 4	15 + 4[b]	2, 4	15 + 2[f] + 4[g]
RCS	30 (6 / 7[e])	SCC/TF, SCC/TF/VF/SS, TF	Class 1	B-F	18	0	18	0	2, 4	26 + 2[d]	1, 2, 4	30 + 3[d]
			Class 1	B-J	217	350	57	136				
RHS	1 (0 / 0)	TF/SCC, TF	Class 1	B-J	22	6	7	2	2, 4	1	2, 4	1
			Class 2	C-F-1	283	0	23	0		0		0
RSS	9 (0 / 0)	TF	Class 2	C-F-1	199	0	16	0	4	0	2, 4	9
SIS	47 (0 / 0)	TF	Class 1	B-J	222	157	43	14	2, 4	0	2, 4	12 + 3[d]
			Class 2	C-F-1	934	200	71	17		19 + 5[b]		21 + 2[f] + 14[g] + 3[d]
SSR	0 (0 / 0)	TF	Class 1	N/A	0	0	0	0	4	0	4	0
			Class 2	N/A	0	0	0	0		0		0
TOTAL	158 (18 / 132)	FAC/TF, TF, SCC/TF, SCC/TF/VF/SS, TF/ VF, VF	Class 1		483	918	128	233		27 NDE + 2 VISUAL		43 NDE + 9 VISUAL
			Class 2		2151	518	181	44		61 NDE + 25 VISUAL		83 NDE + 43 VISUAL
			Total		2634	1436	309	277		88 NDE + 27 VISUAL		126 NDE + 52 VISUAL

Table 1
BVPS-2 STRUCTURAL ELEMENT SELECTION
RESULTS AND COMPARISON TO ASME SECTION XI
1989 EDITION REQUIREMENTS

Summary: Prior ASME Section XI selects a total of 586 welds while the proposed RI-ISI program selects a total of 126 welds (plus 52 visual exams), which results in a 78% reduction.

Degradation Mechanisms:
FAC – Flow-Assisted Corrosion
SCC – Stress Corrosion Cracking
SS – Striping/Stratification
TF – Thermal Fatigue
VF - Vibratory Fatigue
"X/X" indicates combination of mechanisms.

Systems:
BDG – Steam Generator Blowdown System
CHS – Chemical and Volume Control System
CI – Containment Isolation System
DAS – Reactor Plant Drains and Vents Systems
FWA – Steam Generator Feedwater System
GNS – Gaseous Nitrogen System
HCS – Hydrogen Control System
MSS – Main Steam System
QSS – Quench Spray System
RCS – Reactor Coolant System
RHS – Residual Heat Removal System
RSS – Recirculation Spray System
SIS – Safety Injection System
SSR – Sampling System

Notes for Table 1
a. System pressure test requirements and VT-2 visual examinations shall continue in all ASME Code Class systems.
b. VT-2 examination at one location within segment.
c. Augmented program for erosion-corrosion continues.
d. Examinations added for change in risk considerations (total of nine segments – three RCS and six SIS).
e. Augmented program for alloy 82/182 welds continue.
f. VT-2 examination on socket welded portion of segment.
g. VT-2 visual examination for entire segment.

Summary Statement of Beaver Valley Power Station Unit No.1 (BVPS-1) and Beaver Valley Power Station Unit No. 2 (BVPS-2) Probabilistic Risk Assessment Model Capability for Use in Risk-Informed Inservice Inspection Program Licensing Actions

Introduction

FirstEnergy Nuclear Operating Company (FENOC) employs a multi-faceted approach to establishing and maintaining the technical adequacy and plant fidelity of the probabilistic risk assessment (PRA) models for all operating FENOC nuclear generation sites. This approach includes both a proceduralized PRA maintenance and update process, and the use of self-assessments and independent peer reviews. The following information describes this approach, as it applies to the BVPS-1 and BVPS-2 PRA models.

PRA Maintenance and Update

The FENOC risk management process ensures that the applicable PRA model remains an accurate reflection of the as-built and as-operated plants. This process is defined in the FENOC risk management program, which consists of a governing procedure and subordinate implementation procedures. These procedures delineate the responsibilities and guidelines for updating the full-power internal events PRA models at all operating FENOC nuclear generation sites and delineate the responsibilities and guidelines for use of the PRA models in applications. The overall FENOC risk management program defines the process for implementing regularly scheduled and interim PRA model updates, for tracking issues identified as potentially affecting the PRA models (e.g., due to changes in the plant, errors or limitations identified in the model, industry operational experience), and for controlling the model and associated computer files. To ensure that the current PRA model remains an accurate reflection of the as-built, as-operated plant, the following activities are routinely performed:

- Design changes and procedure changes are reviewed for their impact on the PRA model.
- New engineering calculations and revisions to existing calculations are reviewed for their impact on the PRA model.
- Unavailability due to maintenance is captured, and the impact on core damage frequency (CDF) is trended.
- Plant-specific initiating event frequencies, failure rates, and unavailability due to maintenance is updated approximately every three years.

In addition to these activities, FENOC risk management procedures provide the guidance for particular risk management and PRA quality and maintenance activities. This guidance includes:

- Documentation of the PRA model, PRA products, and bases documents.
- The approach for controlling electronic storage of Risk Management (RM) products including PRA update information, PRA models, and PRA applications.
- Guidelines for updating the full-power, internal events PRA models for FENOC nuclear generation sites.

- Guidance for use of quantitative and qualitative risk models in support of the On-Line Work Control Process Program for risk evaluations for maintenance tasks (corrective maintenance, preventive maintenance, minor maintenance, surveillance tests and modifications) on systems, structures, and components (SSCs) within the scope of the Maintenance Rule (10 CFR 50.65 (a)(4)).

In accordance with this guidance, regularly scheduled PRA model updates nominally occur every three years, although longer intervals may be justified if it can be shown that the PRA continues to adequately represent the as-built, as-operated plant. FENOC performed a regularly scheduled update to the BVPS-1 PRA model in 2006 and BVPS-2 PRA model in 2007, and is currently in the process of updating the BVPS-1 PRA model.

<u>PRA Self Assessment and Peer Review</u>

Several assessments of technical capability have been made, and continue to be planned, for the BVPS-1 and BVPS-2 PRA models. These assessments are as follows:

- An independent PRA peer review [Reference 1] was conducted in 2002 under the auspices of the Westinghouse Owners Group (WOG), following issuance of the industry PRA peer review process guidance [Reference 2]. This peer review included an assessment of the PRA model maintenance and update process.
- During 2005, the BVPS-1 and BVPS-2 PRA model results were evaluated in the WOG PRA cross-comparisons study performed in support of implementation of the mitigating systems performance indicator (MSPI) process. Results of this cross-comparison are presented in WCAP-16464-NP [Reference 3]. Notably, after allowing for plant-specific features, there are no MSPI cross-comparison outliers for BVPS-1 or BVPS-2.
- In 2007, a gap analysis [Reference 4] was performed against the ASME PRA Standard [Reference 5] and Regulatory Guide 1.200, Revision 1 [Reference 6].
- Follow-up peer review [Reference 7] of the human reliability analysis (HRA) element, following the industry follow-on PRA peer review process [Reference 8], was performed in 2007 to evaluate the change in HRA methodology since the 2002 WOG Peer Review.
- As part of the transition to NFPA-805, an independent PRA peer review [Reference 9] was conducted in January 2009 of the fire PRA model under the auspices of the Pressurized Water Reactor Owners Group (PWROG), following the industry PRA peer review process [Reference 10]. This peer review included an assessment of the PRA model maintenance and update process for both BVPS-1 and BVPS-2.

A summary of the disposition of the 2002 industry PRA peer review facts and observations for the BVPS-1 and BVPS-2 models are documented within FENOC's Corrective Action Program. The resolutions were reviewed and documented in the 2007 gap analysis report, Table A-3 [Reference 4].

A gap analysis for BVPS-2 [Reference 4] and HRA follow-up peer review for the 2006 BVPS-1 and 2007 BVPS-2 PRA models [Reference 7] was performed. These evaluations were performed against the ASME PRA Standard [Reference 5] and Regulatory Guide 1.200, Revision 1 [Reference 6]. The gap analysis identified 67 supporting requirements with potential gaps to Capability Category II of the Standard. The HRA follow-up review identified 10 supporting requirements that did not meet Capability Category II requirements primarily in the analysis for pre-initiator human actions.

The gap analysis [Reference 4] documented 55 facts and observations that were written against the 67 supporting requirements with potential gaps to Capability Category II. Of these 55 facts and observations, 48 were considered to be documentation issues. Of the remaining seven facts and observations that were considered PRA modeling issues, five were categorized as Capability Category I, which was deemed an acceptable categorization for this application. The last two were categorized as not meeting the supporting requirements. However, these were written against the internal flooding PRA analysis, which was not used directly to support the development of the RI-ISI program, but instead was used as one source for identifying potential flooding sources and possible equipment affected by flooding.

In the HRA follow-on peer review [Reference 7], five facts and observations were written against the ten supporting requirements with potential gaps to Capability Category II. Of these five, all were considered to be documentation issues.

Table 1 lists the status and importance to the Risk-Informed Inservice Inspection of these 60 facts and observations that had potential gaps in meeting Capability Category II of the ASME PRA Standard. The gaps pertaining to the internal flooding analysis, fire analysis, HRA and large early release frequency (LERF) analysis will be addressed during the model update process that is ongoing. Specifically, the analysis updates for LERF and pre-initiator human actions will be integrated into the BVPS-1 PRA model update; the analysis updates for flooding and fire will be integrated into the BVPS-1 Level 1 internal events model update; and the analysis updates for flooding, LERF and pre-initiator human actions will be integrated into the BVPS-2 PRA model update. The analysis updates for fire will be integrated into the BVPS-2 level 1 internal events model update that is scheduled to occur in 2011. The other remaining gaps will be reviewed for consideration during the PRA model update process, but are judged to have low impact on the PRA model and its ability to support a full range of PRA applications. The remaining gaps are documented, so they can be tracked and accounted for in applications where appropriate.

General Conclusion Regarding PRA Capability

The BVPS-1 and BVPS-2 PRA maintenance and update processes and technical capability evaluations described above provide a robust basis for concluding that the PRA is suitable for use in risk-informed licensing actions. As specific risk-informed PRA applications are performed, remaining gaps to specific requirements in the PRA standard will be reviewed to determine which, if any, would merit application-specific sensitivity studies in the presentation of the application results.

Assessment of PRA Capability Needed for Risk-Informed Inservice Inspection

In the risk-informed inservice inspection (RI-ISI) program at BVPS-1 and BVPS-2, the PWROG RI-ISI methodology [Reference 11] is used to define alternative inservice inspection requirements. Plant-specific PRA-derived risk significance information is used during the RI-ISI plan development to support the consequence assessment, risk ranking and delta risk evaluation steps.

The importance of PRA consequence results, and therefore the scope of PRA technical capability, is tempered by three processes in the PWROG methodology.

- In the PWROG methodology two sets of consequences are developed based on the operators taking no action to isolate or mitigate the piping failure and based on the operators being perfect in taking action to isolate or mitigate the piping failure, if there is a credible operator action. Based on this, four risk evaluation workbooks are created for core damage frequency (CDF) and large early release frequency (LERF). If the risk metrics from any of these four risk evaluation workbooks are quantitatively high safety significant (HSS), the segment is identified as quantitatively HSS.

- A simplified uncertainty analysis is performed to ensure that no low safety significant segments could move into high safety significance when reasonable variations in the pipe failure and conditional CDF/LERF probabilities are considered.

- The PWROG RI-ISI methodology is a risk-informed process and not a risk-based process. The quantitative results from the risk evaluation along with deterministic insights and other input data are presented to an expert panel in an integrated decision making process. The primary focus of the expert panel is to review all pertinent information and determine the final safety-significance category for each of the piping segments. The expert panel is comprised of plant personnel with a wide breadth and depth of experience as specified in WCAP-14572 [Reference 11]. Segments that have been determined to be quantitatively HSS are typically categorized as HSS by the expert panel. The focus of the expert panel is to add segments to the higher classification. The BVPS-1 and BVPS-2 expert panel categorized 53 BVPS-1 segments and 59 BVPS-2 segments as HSS that were not quantitatively HSS based on deterministic insights, high failure potential and/or high consequences. Additionally, as part of the integrated decision making process the expert panel considers limitations in the process when categorizing segments as HSS or LSS. This may include PRA model limitations and limitations in modeling the consequences using the PRA model.

The limited manner of PRA involvement in the RI-ISI process is also reflected in the risk-informed license application guidance provided in Regulatory Guide 1.174 [Reference 12]. Section 2.2.6 of Regulatory Guide 1.174 provides the following insight into PRA capability requirements for this type of application:

> *There are, however, some applications that, because of the nature of the proposed change, have a limited impact on risk, and this is reflected in the impact on the elements of the risk model.*
>
> *An example is risk-informed inservice inspection (RI-ISI). In this application, risk significance was used as one criterion for selecting pipe segments to be periodically examined for cracking. During the staff review it became clear that a high level of emphasis on PRA technical acceptability was not necessary. Therefore, the staff review of plant-specific RI-ISI typically will include only a limited scope review of PRA technical acceptability.*

In the PWROG RI-ISI process the PRA model is not used as the basis for the risk evaluation, but instead is used as an input to the risk evaluation process. The vast majority of the piping failure consequences are identified as loss of a system or train of a system. The PRA results are then used as an input to the risk evaluation for the relative ranking of the segments. Table 1.3-1 of the ASME PRA Standard[1] [Reference 8] identifies the bases for PRA capability

[1] Table A-1 of Regulatory Guide 1.200 identifies the NRC staff position as "No objection" to Section 1.3 of the ASME PRA Standard, which contains Table 1.3-1.

categories. The bases for Capability Category I for scope and level of detail attributes of the PRA states:

> *Resolution and specificity sufficient to identify the relative importance of the contributors at the system or train level including associated human actions.*

Based on the above, in general, Capability Category I should be sufficient for PRA quality for a RI-ISI application.

In addition to the above, it is noted that segments and their associated welds determined to be low risk significant are not eliminated from the ISI program on the basis of risk information. For example, the risk significance of a segment may be determined by the expert panel to be low safety significant, resulting in it not being a candidate for inspection. However, it remains in the program and, if in the future the assessment of its ranking changes (either by damage mechanism, PRA risk, or deterministic insight), then it can again become a candidate for inspection. If it is discovered during the RI-ISI update process that a segment is now susceptible to flow-accelerated corrosion (FAC), inter-granular stress corrosion cracking (IGSCC), or microbiological induced cracking (MIC), it is addressed in an augmented program where it is monitored for those special damage mechanisms. That occurs no matter what the risk ranking of the segment or weld is determined to be.

Conclusion Regarding PRA Capability for Risk-Informed ISI

The BVPS-1 and BVPS-2 PRA models continue to be suitable for use in the RI-ISI application. This conclusion is based on:

- the PRA maintenance and update processes in place,
- the PRA technical capability evaluations that have been performed and are being planned, and
- the RI-ISI process considerations, as noted above, that demonstrate the relatively limited reliance of the process on PRA capability.

References

1. Westinghouse Electric Company, "Beaver Valley Power Station PRA Peer Review Report, Final Report," December 2002.
2. Nuclear Energy Institute, NEI 00-02, "Probabilistic Risk Assessment (PRA) Peer Review Process Guidance," Revision A3, March 2000.
3. Westinghouse Electric, WCAP-16464-NP, "Westinghouse Owner's Group Mitigating Systems Performance Index Cross Comparison," Revision 0, August 2005.
4. "Summary of Beaver Valley Power Station Unit 2 PRA Regulatory Guide 1.200 App. B / ASME PRA Standard 'Gap' Assessment," attached to LTR-RAM-I-08-016, January 2008.
5. American Society of Mechanical Engineers, ASME RA-Sb-2005, "Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications," New York, New York, December 2005.
6. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 1, January 2007.

7. "Focused Peer Review of the Human Reliability Analysis Against the ASME PRA Standard Requirements for the Beaver Valley Power Station Probabilistic Risk Assessment," attached to LTR-RAM-II-08-006, March 2008.

8. Nuclear Energy Institute, NEI 05-04, "Process for Performing Follow-On PRA Peer Reviews Using the ASME PRA Standard (Internal Events)," Revision 1 (Draft), November 2007.

9. "Fire PRA Peer Review Against the Fire PRA Standard Supporting Requirements From Section 4 of the ASME/ANS Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessments for Nuclear Power Plant Applications for the Beaver Valley Unit 1 Fire Probabilistic Risk Assessment," LTR-RAM-II-09-006, April 2009.

10. Nuclear Energy Institute, NEI 07-12, "Fire Probabilistic Risk Assessment (FPRA) Peer Review Process Guidelines," Draft Version E, November 2008.

11. Westinghouse Electric, WCAP-14572, "Westinghouse Owners Group Application of Risk-Informed Methods to Piping Inservice Inspection Topical Report," Revision 1-NP-A, February 1999, including:

    Supplement 1, "Westinghouse Structural Reliability and Risk Assessment (SRRA) Model for Piping Risk-Informed Inservice Inspection," Revision 1-NP-A, February 1999.

    Supplement 2, "Pressurized Water Reactor Owners Group Application of Risk Informed Methods to Piping Inservice Inspection Topical Report Clarifications," Revision 1-NP-A, September 2006.

12. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 1, November 2002.

Table 1 – Status of Open Gaps to Capability Category II of the ASME PRA Standard[2]

Facts and Observations	Description of Gap	Supporting Requirement	Current Status or Comment	Importance to Risk-Informed Inservice Inspection
IE-A6-01	There is no documentation of interviews of plant personnel (for example: operations, maintenance, engineering, safety analysis) to determine if potential initiating events have been overlooked. This is required to meet Capability Category II.	IE-A6	Open – Plan is to document interviews of plant personnel that determined if potential initiating events have been overlooked.	None. Capability Category I is met and appropriate for this application. This gap is a documentation consideration only.
IE-C9-01	Plant-specific information used in the assessment and quantification of recovery actions included in the support system initiating event analysis is not included in the support system notebooks. Analysis of the recovery actions should be consistent with the applicable requirements in the human reliability analysis.	IE-C9	Open – Will document the assessment and quantification of any recovery actions assumed in the support system initiating event analysis. If no recovery actions are used or modified, also note that in the documentation.	None. This gap is a documentation consideration only.
IE-C10-01	There is no comparison of the initiating event analysis with generic data sources or explanation of differences to provide a reasonableness check of the results.	IE-C10	Partially Resolved – A comparison was made between the BVPS initiating event data and the WOG initiating event database and NUREG/CR-5750 values. Any outliers have justification provided.	None. This gap is a documentation consideration only.
SC-A5-01	This supporting requirement requires that for sequences in which stable plant conditions would not be achieved by 24 hours using the modeled plant equipment and human actions, perform additional evaluation or modeling by using an appropriate technique. The makeup to refueling water storage tank (MU) top event for medium loss of coolant accident (LOCA) and small LOCA/general transient uses refueling water storage tank (RWST) makeup as part of the success path when recirculation has failed. While a mission time of 24 hours is assumed, the plant is not at	SC-A5	Open - Additional evaluations or modeling by using an appropriate technique will be performed for sequences in which stable plant conditions would not be achieved by 24 hours using the modeled plant equipment and human actions. For top event MU, document that the plant conditions reach acceptable stable values and that using the analyzed RWST makeup flow rate would not result in	None. Capability Category I is met and appropriate for this application. This gap is a documentation consideration only.

[2] The gap analysis is conducted independently of RI-ISI and is based on comparing the PRA model against the supporting requirements of ASME PRA standard at Capability Category II. Many of the identified gaps are not applicable to RI-ISI since in general Capability Category I is sufficient. For completeness, all current gaps that do not meet at least Capability Category II are identified in Table 1.

Facts and Observations	Description of Gap	Supporting Requirement	Current Status or Comment	Importance to Risk-Informed Inservice Inspection
	a safe stable state because another action is required for long term success. The RWST refill results in additional water to the containment which eventually will result in the design basis flooding level being exceeded and the potential for subsequent loss of instrumentation and control. The impact of continued RWST makeup and injection into containment needs to be discussed in relation to the achievement of a safe stable state where no additional operator actions are required. A similar situation exists for steam generator tube rupture (SGTR) and interfacing system LOCA (ISLOCA) where RWST refill is being used to maintain core cooling, but the justification for mission time of only 24 hours is not apparent given that the plant is not in a safe stable state by traditional definitions.		containment flooding issues that would impact any equipment or instrumentation important for mitigating the accident. Use the containment water level and volume SAMG CA-5 for guidance on what equipment and instrumentation could become submerged based on the RWST makeup flow rate.	
SC-A5-02	The success criteria for top event makeup to RWST given leakage through secondary (WM) for the SGTR states that 400 gallons per minute (gpm) makeup to the RWST is sufficient to maintain high head safety injection (HHSI) for reactor coolant system (RCS) inventory control at full RCS pressure, despite leakage through a ruptured steam generator (SG) tube. The maximum RCS inventory loss through a single SGTR is approximately 600 gpm if the primary side is at normal operating pressure and the secondary side of the SG is not depressurized. This is in excess of the 400 gpm makeup and therefore appears to invalidate the success criteria as stated. Also, if continued HHSI at full system pressure is required, SG overfill is likely to occur and the SG will be depressurized and the leakage through the ruptured tube will even be higher.	SC-A5	Open – Will provide justification for 400 gpm success criteria for top event WM to maintain HHSI for RCS inventory control at full RCS pressure despite leakage through a ruptured SG tube. Also noting that by the time makeup is required the RCS would not be at full RCS pressure due to the breach in the SG tube.	None. Capability Category I is met and appropriate for this application. This was judged not to impact PRA results and is not required to meet SC-A5. This is expected to be a clarification of the use of these success criteria and is therefore assigned a Level C. This gap is a documentation consideration only.

Facts and Observations	Description of Gap	Supporting Requirement	Current Status or Comment	Importance to Risk-Informed Inservice Inspection
SC-C1-02	The ASME PRA standard for SC-C1 requires that success criteria be documented in a manner that facilitates applications, upgrades, and peer reviews. The current state of the BVPS PRA success criteria is that the accident sequence success criteria are gathered in the success criteria notebook, but other success criteria are scattered throughout the PRA. Examples include the SW success criteria and ISLOCA success criteria for BVPS-1. It is recommended that FENOC consider gathering all success criteria in the success criteria notebook to facilitate future usage.	SC-C1, SY-C1	Open – Will gather all (system) success criteria in the success criteria Notebook to facilitate future usage.	None. This gap is a documentation consideration only.
SC-C2-01	No discussion of the limitations of the modular accident analysis program (MAAP) code for success criteria are provided in the success criteria notebook. Two known limitations are the use of MAAP for early phase large LOCAs and the use of MAAP for steam generator dryout assessments without benchmarking to design basis codes (for example, bleed and feed initiation). It was observed in the success criteria notebook that MAAP runs were made to justify only one accumulator (but that two of two intact accumulators appear to have been actually used as stated to be used in section 3.1 of the notebook). It is recommended that a discussion of MAAP limitations (similar to that provided in the EPRI assessment for MAAP 3) be documented or referenced in the success criteria notebook.	SC-C2	Open – Will add a discussion of MAAP limitations (similar to the EPRI assessment for MAAP 3) to be documented or referenced in the success criteria notebook. Also reference the MAAP users guide for additional info.	None. This gap is a documentation consideration only.
SY-A14-01	The draft revision 4 system notebooks for auxiliary feedwater, service water, component cooling water secondary side, component cooling water primary side, and main feedwater were reviewed. Discuss failure modes and contributors to system unavailability and unreliability that are excluded from the systems analysis. However, a SY-A14 criterion does not appear to have been applied consistently	SY-A12, SY-A14, SY-C1	Open – Will add a discussion for the excluded failure modes and contributors to system unavailability and unreliability. However, it is unlikely that these contributors will significantly impact PRA results.	None. This gap is a documentation consideration only.

Facts and Observations	Description of Gap	Supporting Requirement	Current Status or Comment	Importance to Risk-Informed Inservice Inspection
	throughout the analysis. The only exceptions found where the SY-A14 criteria are explicitly met is in the CCS notebook, section 14, c, assumption 2, and the AFW notebook section 14, c, assumption 3. In some instances, such as the CCP notebook section 14, c, assumption 1, there was no explanation given for why the contributor was not modeled.			
SY-C1-01	In providing the response to peer review DA-09, which deals with providing documentation of the common cause failure (CCF) groupings, FENOC noted that the systems analysis overview and guidance notebook provides the process used to identify CCF groupings. The response further suggests details of the common cause groups that were retained in the PRA system models and presented in appendix C of the BVPS-2 PRA system notebooks, under the common cause sections of the risk management software program (RISKMAN) System notebook files are adequately documented and can be found by knowledgeable personnel. The reviewer agrees that one can review Appendix C of the systems notebooks and see what the CCF groupings are and how the CCF probabilities were generated. The reviewer also agrees that high level guidance is provided in the systems analysis overview and guidance notebook. However, it appears a link between the two documents is missing. For example, the guidance states "When identical, non-diverse, and active components are used to provide redundancy, they should be considered for assignment to common cause groups, one group for each identical redundant component." When the systems notebook appendix C is reviewed, the components	SY-C1	Open – Will add brief summary of the CCF group selections, possibly as part of the system notebook, section 15 "Common Cause".	None. This gap is a documentation consideration only.

Facts and Observations	Description of Gap	Supporting Requirement	Current Status or Comment	Importance to Risk-Informed Inservice Inspection
	contained in the CCF group are clearly identified, but there is no documentation that states that those components are "identical, and/or non-diverse" or used to provide redundancy. Further examination of other sections of the system notebooks (such as section 3 "System Success Criteria", or section 6 "Operating Features", would lead a reviewer to find this type of information. But this documentation is not always intuitively obvious and makes peer review difficult at times.			
SY-C1-02	The BVPS-2 system notebooks have no indication of system engineering reviews. These reviews help ensure that systems are modeled in accordance with day-to-day plant operations and additionally expand the PRA knowledge of the system engineers.	SY-C1	Partially Resolved -- In the process of documenting system notebooks reviews by system engineering.	None. This gap is a documentation consideration only.
HR-B1-01	This is a carry-over from the HR-2 peer review. A generic error of omission term from the Pikard, Lowe, and Garrick, Inc. (PLG) database (ZHEO1A) was used for all misalignment human error probabilities without regard for procedural or operational failure barriers such as independent verification, peer checks, walkdowns, etc. However, plant-specific data was used for test and maintenance frequencies. Therefore, the overall misalignment errors were a hybrid of generic and plant-specific data. This was used for systems which are important to CDF (for example, auxiliary feedwater and safety injection).	HR-B1, HR-D2	Open – Will calculate specific misalignment error of omission failure probabilities for important systems using the EPRI human reliability analysis (HRA) calculator.	None. Capability Category I is met and appropriate for this application. Refer to the section "Assessment of PRA Capability Needed for Risk-Informed Inservice Inspection." It is not expected that the BVPS specific misalignment values will be significantly different from the generic values used.
HR-D3-01	While the discussion in the system notebooks (auxiliary feedwater, quench spray, and recirculation spray notebooks were reviewed) references the procedures, no documentation of quality of those procedures or administrative controls was found.	HR-D3	Open – Will confirm and document that the procedure quality is sufficient to support the crew response within the times assigned in the PRA evaluation.	None. Capability Category I is met and appropriate for this application. This gap is a documentation consideration only.

Facts and Observations	Description of Gap	Supporting Requirement	Current Status or Comment	Importance to Risk-Informed Inservice Inspection
HR-I1-01	The BVPS-2 system and data notebooks have been updated and exist in draft form, but there is no record of formal review and approval. Furthermore, only a subset of the total PRA notebooks have been updated for this revision of the PRA.	HR-I1, HR-I2	Partially Resolved – Complete the update of the PRA analysis and system notebooks with formal review and approval.	None. This gap is a documentation consideration only.
HR-I2-01	The BV human reliability analysis does document a process to perform a systematic search for dependent human actions credited on individual sequences. It is clear from the human action identifier sheets documented in the BVPS-2 human reliability analysis notebook that such an evaluation has been performed, but there is no evidence of the process documented in the human reliability analysis notebook. To be consistent with current human reliability analysis methods, there must be a systematic process to identify, assess and adjust dependencies between multiple human errors in the same sequence, including those in the initiating events.	HR-I2	Open – Will document the process used to perform a systematic search for dependent human actions credited on individual sequences.	None. This gap is a documentation consideration only.
HR-I2-02	There is no evidence in the human reliability analysis or success criteria notebooks that an operator review of the human reliability analysis has been performed.	HR-I2	Open - During the recent extended power uprate evaluation, plant operations did review the operator actions and timings. There are reports to document these reviews [See Note 2]. Furthermore, several operator action scenarios were evaluated using the plant simulator. The results of the review of operator actions will be incorporated into the human reliability analysis notebook or success criteria notebook.	None. This gap is a documentation consideration only.
HR-I3-01	The human reliability analysis notebook sporadically discusses assumptions and uncertainties. Per the clarification to Regulatory Guide 1.200, Revision 1, there is an increased	IE-D3, AS-C3, SC-C1, SC-C3,	Open – Will document all of the human reliability analysis assumptions and uncertainties into a new "Assumptions and	None. This gap is a documentation consideration only.

Facts and Observations	Description of Gap	Supporting Requirement	Current Status or Comment	Importance to Risk-Informed Inservice Inspection
	importance in the industry to identify assumptions and uncertainties in the PRA model. In reviewing the human reliability analysis notebook, it is difficult to locate the assumptions and uncertainties.	HR-I1, HR-I3, IF-F3, QU-F4, LE-F3, LE-G4	Uncertainties" section in the human reliability analysis notebook. Also, the quantification notebook lists an evaluation of the model uncertainties; however, a more comprehensive set of assumptions and uncertainties will be documented.	
DA-C4-01	A clear basis for the identification of events as failures is not included in the data analysis notebook. This basis could be used to distinguish between those degraded states for which a failure, as modeled in the PRA, would have occurred during the mission and those for which a failure would not have occurred (for example, slow pick-up to rated speed). It could not be determined from the data analysis notebook if any failures were screened out or if the maintenance rule maintenance preventable functional failures are used as the data source.	DA-C4	Partially Resolved – The methodology for a clear basis to identify events as failures was interpreted from NUREG/CR-2823 and is documented in a draft copy of the data analysis notebook. This provides the basis for guidance to distinguish between degraded states for which a failure is modeled in the PRA.	None. This gap is a documentation consideration only.
DA-C5-01	There is no listing or description in the data analysis notebook of repeated component failures that were counted as a single failure. Repeated component failures occurring within a short time interval should be counted as a single failure if there is a single, repetitive problem that causes the failures. In addition only one demand should be counted.	DA-C5	Open – Will document a listing or description of repeated component failures that were counted as a single failure in the data analysis notebook.	None. This gap is a documentation consideration only.
DA-C8-01	Plant records should be used and documented to determine the time that components are configured in their standby status. This is required to change DA-C8 from Capability Category I to III.	DA-C8	Open – Will use plant records to determine and document the time that components are configured in their standby status.	None. Capability Category I is met and appropriate for this application. This gap is a documentation consideration only.

Facts and Observations	Description of Gap	Supporting Requirement	Current Status or Comment	Importance to Risk-Informed Inservice Inspection
DA-C10-01	Decompose failure modes into sub-elements and count demands and failures individually in the sub-elements.	DA-C10	Open – If the component failure mode is decomposed into sub-elements that are fully tested, will review test procedures to ensure that tests that exercise specific sub-elements are used for their evaluation.	None. Capability Category I is met and appropriate for this application. Refer to the Section "Assessment of PRA Capability Needed for Risk-Informed Inservice Inspection." There are only a limited number of component failure modes that are decomposed into sub-elements, no significant impact is expected.
IF-A1a-01	It is not clear from the documentation that a comprehensive assessment has been conducted to finalize the combined rooms including propagation, barriers, etc. The internal flooding assessment is based on large flood areas but there is no description of the process used to define those areas with respect to flood propagation and barriers.	IF-A1a	Open - The internal flooding analysis gaps will be addressed during the internal flooding model update that is scheduled for 2010. This flooding analysis update will then be integrated into the level 1 internal events model update.	None. Capability Category I is met and appropriate for this application. In addition, the internal flooding PRA analysis was not used directly to support the development of the RI-ISI program, but instead was used as one source for identifying potential flooding sources and possible equipment affected by flooding. This gap is a documentation consideration only.
IF-A3-01	There is no evidence in the internal flooding notebook that it represents the current as-built, as-operated plant. Revision 4 documentation in another document may include the information to show that the internal flooding assessment is current, but it is not in this notebook. IF-A3-01 was written as a B level fact and observation to provide documentation that the internal flooding assessment still represents the as-built as-operated plant in 2007. This probably also applies to other PRA elements from the ASME PRA standard (for example, system analysis, success criteria, human reliability analysis, etc.) and should be addressed generically for the BVPS PRA. This would facilitate future reviews and development of PRA applications.	IF-A3	Open - The internal flooding analysis gaps will be addressed during the internal flooding model update that is scheduled for 2010. This flooding analysis update will then be integrated into the level 1 internal events model update.	None. The internal flooding PRA analysis was not used directly to support the development of the RI-ISI program, but instead was used as one source for identifying potential flooding sources and possible equipment affected by flooding. This gap is a documentation consideration only.

Facts and Observations	Description of Gap	Supporting Requirement	Current Status or Comment	Importance to Risk-Informed Inservice Inspection
IF-B1-01	The ASME PRA standard states "for each flood area, identify the potential sources of flooding." Section C3.1 identifies flood sources in each area but clear documentation of each source in an area is lacking. The standard expects a more systematic approach for identifying potential flood sources and then later screening them. The internal flooding assessment here includes initial screening without written justification. It is suggested that a complete discussion of potential sources be documented and used as the basis for screening potential sources.	IF-B1	Open - The internal flooding analysis gaps will be addressed during the internal flooding model update that is scheduled for 2010. This flooding analysis update will then be integrated into the level 1 internal events model update.	None. The internal flooding PRA analysis was not used directly to support the development of the RI-ISI program, but instead was used as one source for identifying potential flooding sources and possible equipment affected by flooding. This gap is a documentation consideration only.
IF-B1-02	Section C3.1 states that major flood sources were reviewed to identify potential flood locations. The ASME standard suggests that first you identify flooding areas then identify all flooding sources in that area. This method used for BVPS may have led to overlooking other sources of flooding within each area.	IF-B1	Open - The internal flooding analysis gaps will be addressed during the internal flooding model update that is scheduled for 2010. This flooding analysis update will then be integrated into the level 1 internal events model update.	None. The internal flooding PRA analysis was not used directly to support the development of the RI-ISI program, but instead was used as one source for identifying potential flooding sources and possible equipment affected by flooding. This gap is a documentation consideration only.
IF-B2-01	B-2 of the PRA standard requires "For each source of flooding, identify the flooding mechanisms that would result in a fluid release including failure models, human-induced mechanisms, and other events resulting in a release into the flood area." In addition, B-3 requires "For each source and its identified failure mechanism, identify the characteristic of release and the capacity of the source." Section C3.1 of the internal flooding notebook does not provide enough detail to judge whether this requirement is met. One example is that although a few human error induced floods (for example, testing or maintenance errors) were considered, there is no evidence of a systematic assessment of potential test and maintenance errors.	IF-B2, IF-B3	Open - The internal flooding analysis gaps will be addressed during the internal flooding model update that is scheduled for 2010. This flooding analysis update will then be integrated into the level 1 internal events model update.	None. The internal flooding PRA analysis was not used directly to support the development of the RI-ISI program, but instead was used as one source for identifying potential flooding sources and possible equipment affected by flooding. This gap is a documentation consideration only.

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IF-C2b-01	Section C3.1 does not have enough detail to show that the capacity of the drains and the amount of water retained by the sumps, berms, dikes, and curbs was estimated. The reviewer notes that it is likely that this was performed but there is no record of the assessment. The capacity of drains and the amount of water retained by sumps, etc. should be documented in the internal flooding notebook.	IF-C2b	Open - The internal flooding analysis gaps will be addressed during the internal flooding model update that is scheduled for 2010. This flooding analysis update will then be integrated into the level 1 internal events model update.	None. The internal flooding PRA analysis was not used directly to support the development of the RI-ISI program, but instead was used as one source for identifying potential flooding sources and possible equipment affected by flooding. This gap is a documentation consideration only.
IF-C3-01	The PRA standard states "for each SSCs identified in IF-C2c identify the susceptibility of each SSC in the flood area to flood-induced failure mechanism". Also, C3a states, "to determine susceptibility of SSC to flood-induced failure mechanism, take credit for the operability of SSC identified in IF-C2c with respect to internal flood impact only if supported by an appropriate combination of: 1) test or operational data, 2) engineering analysis, and 3) expert judgment." It is likely that flood-induced failure mechanisms were considered in the internal flooding assessment but are not identified in the internal flooding notebook. Section C3.1 does not provide enough detail on the impact of the flood on SSCs.	IF-C3, IF-C3a	Open - The internal flooding analysis gaps will be addressed during the internal flooding model update that is scheduled for 2010. This flooding analysis update will then be integrated into the level 1 internal events model update.	None. The internal flooding PRA analysis was not used directly to support the development of the RI-ISI program, but instead was used as one source for identifying potential flooding sources and possible equipment affected by flooding. This gap is a documentation consideration only.
IF-C3b-01	IF-C3b requires that all potential mechanisms that can create interconnections between flooding areas be considered for Capability Category II (CCII) and that barrier unavailability also be considered for Capability Category III (CCIII). There is no evidence in appendix C of the initiating events notebook that any mechanism other than open obvious pathways (for example, vents in doors, tunnels, etc.) were considered. This may be just a documentation issue for CCII. Also, the RI-ISI program did a comprehensive assessment of flooding potential for various break locations. A comparison should be	IF-C3b	Open - The internal flooding analysis gaps will be addressed during the internal flooding model update that is scheduled for 2010. This flooding analysis update will then be integrated into the level 1 internal events model update.	None. The internal flooding PRA analysis was not used directly to support the development of the RI-ISI program, but instead was used as one source for identifying potential flooding sources and possible equipment affected by flooding. This gap is a documentation consideration only.

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	performed between the RI-ISI flooding assessment and the PRA internal flooding assessment to ensure consistency. Note that upgrading to CCIII requires the additional consideration of barrier unavailability, for example due to maintenance activities or maintenance unavailability.			
IF-C3c-01	Develop engineering calculations for all flooding scenarios, not just the worst case scenarios. This is likely just a documentation issue, but since it is missing from the internal flooding notebook, IF-C3c is not met.	IF-C3c	Open - The internal flooding analysis gaps will be addressed during the internal flooding model update that is scheduled for 2010. This flooding analysis update will then be integrated into the level 1 internal events model update.	None. The internal flooding PRA analysis was not used directly to support the development of the RI-ISI program, but instead was used as one source for identifying potential flooding sources and possible equipment affected by flooding. This gap is a documentation consideration only.
IF-C4-01	The operator actions credited in the internal flooding assessment are based on detailed HRA assessments for two operator actions. Cues, procedures, etc. are detailed in the HRA assessment. It is not clear if these actions are also applied to scenarios other than those used to quantify the human error probability in the HRA notebook. In addition, there are a number of other instances in which the operators are assumed to be highly reliable. There is also no indication that these are validated by operator interviews. Cleaner documentation of the operator actions that are credited (as well as those not credited), and their basis, should be completed to assist in future reviews and for risk applications in which the performance of operators is important. Also a clear linkage between the internal flooding and HRA notebooks should be documented for the basis of the important HRA input and some of the operator actions to screen scenarios are based on highly reliable operator actions.	IF-C4	Open - The internal flooding analysis gaps will be addressed during the internal flooding model update that is scheduled for 2010. This flooding analysis update will then be integrated into the level 1 internal events model update.	None. The internal flooding PRA analysis was not used directly to support the development of the RI-ISI program, but instead was used as one source for identifying potential flooding sources and possible equipment affected by flooding. This gap is a documentation consideration only.

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IF-C4-02	IF-C4 requires the development of flood scenarios by examining the equipment and relevant plant features in the flood area and area in potential propagation paths, taking credit for appropriate flood mitigation systems or operator actions, and identifying susceptible SSCs. No flood scenarios are developed in the internal flooding notebook.	IF-C4	Open - The internal flooding analysis gaps will be addressed during the internal flooding model update that is scheduled for 2010. This flooding analysis update will then be integrated into the subsequent level 1 internal events model update.	None. The internal flooding PRA analysis was not used directly to support the development of the RI-ISI program, but instead was used as one source for identifying potential flooding sources and possible equipment affected by flooding. This gap is a documentation consideration only.
IF-C5-01	The screening methodology documented in Section C3.1 does not follow the systematic methodology described in the standard. For the internal flooding assessment, the screening is performed at the source and location level and, in some cases, without adequate basis as discussed in IF-B1-01. The method used in the internal flooding assessment may be technically adequate, if the basis is better documented, even though it does not meet the standard supporting requirements for C-5, C5a and C7.	IF-C5, IF-C5a, IF-C7, IF-D7	Open - The internal flooding analysis gaps will be addressed during the internal flooding model update that is scheduled for 2010. This flooding analysis update will then be integrated into the subsequent level 1 internal events model update.	None. The internal flooding PRA analysis was not used directly to support the development of the RI-ISI program, but instead was used as one source for identifying potential flooding sources and possible equipment affected by flooding. This gap is a documentation consideration only.
IF-D1-01	The FENOC response to DE-06 from the owners group peer review is incomplete. The fact and observation is concerned about the vintage of the data used to estimate pipe break frequencies and the FENOC response talks about walkdowns.	IF-D1	Open - The internal flooding analysis gaps will be addressed during the internal flooding model update that is scheduled for 2010. This flooding analysis update will then be integrated into the subsequent level 1 internal events model update.	None. The internal flooding PRA analysis was not used directly to support the development of the RI-ISI program, but instead was used as one source for identifying potential flooding sources and possible equipment affected by flooding. This gap is a documentation consideration only.
IF-D1-02	The internal flooding assessment does not rely on grouping of initiating events, sources, locations, etc. The screening methodology discussed in the internal flooding notebook and assessed under the IF-C-xx supporting requirements methodology resulted in only a handful of flooding events to be considered. These were individually assessed in the overall PRA quantification using RISKMAN. The methodology used may be technically adequate in spite of not meeting the ASME standard	IF-D1	Open - The internal flooding analysis gaps will be addressed during the internal flooding model update that is scheduled for 2010. This flooding analysis update will then be integrated into the subsequent level 1 internal events model update.	None. The internal flooding PRA analysis was not used directly to support the development of the RI-ISI program, but instead was used as one source for identifying potential flooding sources and possible equipment affected by flooding. This gap is a documentation consideration only.

Facts and Observations	Description of Gap	Supporting Requirement	Current Status or Comment	Importance to Risk-Informed Inservice Inspection
	supporting requirements for grouping if it can be justified that only a handful of events are important.			
IF-D5-01	The internal flooding pipe and tank break frequencies used in the internal flooding assessment are based on 1988 and 1990 data. The prior pipe break frequencies should be updated to reflect more recent experience and should include plant-specific experience. In estimating pipe break frequencies, it is recommended that experience with safety related vs. balance of plant piping be considered along with active pipe degradation mechanisms. Credit for condition monitoring programs should also be applied where applicable.	IF-D5, IF-D5a	Open - The internal flooding analysis gaps will be addressed during the internal flooding model update that is scheduled for 2010. This flooding analysis update will then be integrated into the subsequent level 1 internal events model update.	None. The internal flooding PRA analysis was not used directly to support the development of the RI-ISI program, but instead was used as one source for identifying potential flooding sources and possible equipment affected by flooding. In the PWROG RI-ISI methodology, piping failure probabilities are estimated using the Win-SRRA code for each segment in the scope of the RI-ISI program. Failure probability estimates include appropriate degradation mechanisms including active degradation mechanisms. Although, Beaver Valley should consider more recent pipe rupture data that considers an increased knowledge of pipe degradation mechanisms and considers plant aging concerns, updating is not expected to significantly impact the flooding results.
IF-D5-02	The initiating event frequency (IEF) for pipe breaks is based on a generic 80 percent capacity factor. There are two issues with this method: a) current capacity factors are typically greater than 80 percent so that the IEFs are slightly lower, and b) the method is inconsistent with the method used to calculate other IEFs. It is recommended that the calculation for internal flooding IEF be revised to be consistent with the method used for other IEFs.	IF-D5	Open - The internal flooding analysis gaps will be addressed during the internal flooding model update that is scheduled for 2010. This flooding analysis update will then be integrated into the subsequent level 1 internal events model update.	None. The internal flooding PRA analysis was not used directly to support the development of the RI-ISI program, but instead was used as one source for identifying potential flooding sources and possible equipment affected by flooding. In the PWROG RI-ISI methodology, piping failure probabilities are estimated using the Win-SRRA code for each segment in the scope of the RI-ISI program. Failure probability estimates include appropriate degradation mechanisms including active

Facts and Observations	Description of Gap	Supporting Requirement	Current Status or Comment	Importance to Risk-Informed Inservice Inspection
				degradation mechanisms. Although, Beaver Valley should consider current capacity factors and methodology, updating is not expected to significantly impact the flooding results.
IF-E1-01	The standard states "for each flood scenario, review the accident sequences for the associated plant-initiating event group to confirm applicability of other accident sequences model." A spot check was made to provide reasonable confidence that the overall results are correct. However, there is no record that each scenario was reviewed.	IF-E1	Open - The internal flooding analysis gaps will be addressed during the internal flooding model update that is scheduled for 2010. This flooding analysis update will then be integrated into the subsequent level 1 internal events model update.	None. The internal flooding PRA analysis was not used directly to support the development of the RI-ISI program, but instead was used as one source for identifying potential flooding sources and possible equipment affected by flooding. This gap is a documentation consideration only.
IF-F1-01	The internal flooding documentation does not include the results of the walkdowns performed during the original assessment. FENOC's response to the owners group peer review DE-4 indicates that the RI-ISI walkdowns are documented and cover the issues required for an internal flooding walkdown. To facilitate future maintenance and reviews of the internal flooding assessments, the use of the RI-ISI walkdowns for internal flooding should be documented in the internal flooding notebook and a direct reference to a retrievable copy the RI-ISI walkdowns should also be included.	IF-F1, SY-A4	Open - The internal flooding analysis gaps will be addressed during the internal flooding model update that is scheduled for 2010. This flooding analysis update will then be integrated into the subsequent level 1 internal events model update.	None. The internal flooding PRA analysis was not used directly to support the development of the RI-ISI program, but instead was used as one source for identifying potential flooding sources and possible equipment affected by flooding. This gap is a documentation consideration only.
IF-F1-02	If the current internal flooding methodology is retained, a comparison of the current methodology to the ASME standard is recommended to facilitate future reviews.	IF-F1	Open - The internal flooding analysis gaps will be addressed during the internal flooding model update that is scheduled for 2010. This flooding analysis update will then be integrated into the subsequent level 1 internal events model update.	None. The internal flooding PRA analysis was not used directly to support the development of the RI-ISI program, but instead was used as one source for identifying potential flooding sources and possible equipment affected by flooding. This gap is a documentation consideration only.
IF-F2-01	The documentation of the processes to identify flood areas, sources, pathways, scenarios, etc. are not clearly documented. For example, the	IF-F2	Open - The internal flooding analysis gaps will be addressed during the internal flooding model	None. The internal flooding PRA analysis was not used directly to support the development of the RI-ISI program,

Facts and Observations	Description of Gap	Supporting Requirement	Current Status or Comment	Importance to Risk-Informed Inservice Inspection
	rules used to screen out sources and areas are not defined and the bases for eliminating or justifying propagation pathways is either not clearly defined or not provided at all.		update that is scheduled for 2010. This flooding analysis update will then be integrated into the subsequent level 1 internal events model update.	but instead was used as one source for identifying potential flooding sources and possible equipment affected by flooding. This gap is a documentation consideration only.
IF-F2-02	The internal flooding notebook states that the annual frequency of a flood scenario in location X is Rx = Fi * fx,i * fs,x * fp,x and the equation used to quantify scenarios in which recovery actions can be included is Sx = Rx (Dx + Ix). However, the frequency is never quantified using these equations. This is confusing for a reviewer -- what is the purpose of these statements if they are not used or if they are used, an explanation is needed. Note 1 defines the variables.	IF-F2	Open - The internal flooding analysis gaps will be addressed during the internal flooding model update that is scheduled for 2010. This flooding analysis update will then be integrated into the subsequent level 1 internal events model update.	None. The internal flooding PRA analysis was not used directly to support the development of the RI-ISI program, but instead was used as one source for identifying potential flooding sources and possible equipment affected by flooding. This gap is a documentation consideration only.
QU-B9-01	Component boundary conditions are not well defined. The data analysis notebook, as well as several system notebooks (auxiliary feedwater and service water) were reviewed and there is no discussion of component boundary. There are assumptions made regarding system boundaries, but no discussion of component boundaries. As a result, module definitions can not be determined.	QU-B9	Open – Will provide a discussion of component boundaries in the data analysis or system notebooks. This may be addressed at a higher level by identifying typical groupings of components.	None. This gap is a documentation consideration only.
QU-D5a-01	The revision 3B quantification notebook and revision 4 initiating events notebook were reviewed. Significant contributors to core damage frequency have been identified, but there is no identification of SSCs and operator actions that contribute to initiating event frequencies and event mitigation.	QU-D5a	Open – Will identify significant SSCs and operator actions that contribute to initiating event frequencies and event mitigation in the quantification notebook.	None. Category I is met and appropriate for this application. This gap is a documentation consideration only.
QU-F4-01	The revision 3B quantification notebook, section 5 states that the PRA notebooks "include an estimation of the uncertainty introduced by the data used to quantify the PRA model...This	QU-F4, QU-E4, IE-D3	Open – Will characterize all major sources of uncertainty using WCAP-16304[3] or EPRI TR-1009652[4] guidance.	None. This gap is a documentation consideration only.

[3] Westinghouse Electric, WCAP 16304-P, "Strategy for Identification and Treatment for Modeling Uncertainties in PSAs Applications to LOCA and LOOP."
[4] Electric Power Research Institute (EPRI), Report 1009652, "'Guideline for the Treatment of Uncertainty in Risk-Informed Applications' Technical Basis Document."

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	uncertainty estimation does not, however, reflect possible effects on the results from other sources of uncertainty. Such sources may include such things as: optimism or pessimism in definitions of sequence, component, or human action success criteria; limitations in sequence models due to simplifications (for example, not modeling available systems or equipment) made to facilitate quantification; uncertainty in defining human response within the emergency procedures...; degree of completeness in selection of initiating events; assumptions regarding phenomenology or structures, systems, and components (SSC) behavior under accident conditions... While it is difficult to quantify the effects of such sources of uncertainty, it is important to recognize and evaluate them because there may be specific PRA applications where their effects may have a significant influence on the results." QU-F4 requires that these sources of uncertainty be characterized regardless of the difficulty of the evaluation. By Beaver Valley's own admission, it is important to recognize and evaluate them because there may be specific PRA applications where their effects may have a significant influence on the results. Furthermore, the documentation provided in chapter 5 of the quantification notebook makes a start at identifying the sources of model uncertainty. PWROG guidance suggests the number of identified sources of uncertainty typically is on the order of 50 items. It is also suggested that BVPS perform a more rigorous search to complete a fairly complete list of sources of uncertainty.			

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QU-F4-02	A detailed description of the risk management program (RISKMAN) quantification process is provided. However, the revision 3B quantification notebook does not discuss limitations in the methodology.	QU-F4	Open – Will add a brief discussion regarding the limitations of the RISKMAN methodology to the quantification notebook.	None. This gap is a documentation consideration only.
QU-F6-01	Beaver Valley does list important operator action basic events; however, there is no documented definition of "significant". The revision 3B quantification notebook lists top accident sequences but provides no definition of whether they are "significant" or not. The only discussion is that there is "no single sequence makes up a large fraction of the CDF". The quantification notebook states the following definition for important systems: "The system rankings for determining High Importance is based on having an F-V Importance greater than 5.0E-02 or a RAW greater than 10, while the Low Importance is based on having an F-V Importance less than 5.0E-03 and a RAW less than 2. Medium Importance systems are comprised of everything else in between these importance measures." This definition agrees with the Regulatory Guide 1.200 definition for "significant contributors." However, there is no documented justification (no reference to a standard definition, such as R.G. 1.200 or the EPRI PRA Applications Guide).	QU-F6	Open – Will define the term "significant" and add a discussion or reference to justify the risk-importance rankings for systems and basic events.	None. This gap is a documentation consideration only.
LE-C2a-01	LE-C2a is assigned a Capability Category I because BVPS 2 does not use operator actions post core damage. This is considered conservative treatment of operator actions following the onset of core damage. To meet Capability Category III for this supporting requirement, BVPS-2 level 2 analysis must contain realistic operator actions, based on	LE-C2a, LE-C2b, LE-C3, LE-C6	Open – Will include realistic operator actions as part of the level 2 analysis based on severe accident management guidelines, emergency operating procedures, and WCAP-16657-P[5].	None. Capability Category I is met and appropriate for this application. Refer to the Section "Assessment of PRA Capability Needed for Risk-Informed Inservice Inspection." Any credit for post core damage operator actions would only help to reduce the

[5] Westinghouse Electric, WCAP 16657-P, "SAMG Template for Level 2 PRA."

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	severe accident management guidelines (SAMGs), emergency operating procedures (EOPs), etc. such as WCAP-16657-P.			LERF.
LE-C2b-01	Only recovery of AC power after uncovery of top of active fuel is discussed in the Level 2 notebook. It is concluded that not enough time exists to assign a high success probability. No other recoveries are discussed.	LE-C2b	Open – Will discuss post core damage recoveries that could impact LERF such as restoring feedwater to a ruptured steam generator, using WCAP-16657 and WCAP-16341 as a reference.	None. Capability Category I is met and appropriate for this application. This gap is a documentation consideration only.
LE-C9a-01	Level 2 and LERF analysis stopped at containment failure and continued operation of equipment and operator actions were not modeled. Operation of mitigating systems after containment failure is not modeled either. Justify the lack of credit of equipment survivability.	LE-C9a, LE-C9b	Open – Will justify the lack of credit of equipment survivability and review NUREG/CR-6595[6] for guidance.	None. Capability Category I is met and appropriate for this application. This gap is a documentation consideration only.
LE-C10-01	SGTR and containment bypass did not take credit for scrubbing. WCAP-16657 suggests that scrubbing for tube rupture events can be credited by an operator action restart auxiliary feedwater to the ruptured steam generator.	LE-C10	Open – Will credit scrubbing for SGTR and containment bypass events based on WCAP-16657. The ASME standard recognizes scrubbing during SGTRs as a way to reduce LERF.	None. Capability Category I is met and appropriate for this application. Any credit for SGTR scrubbing would only help to reduce the LERF.
LE-D5-01	Beaver Valley Thermal Induced SGTR is based on a 1995 Fauske and Associates report and Westinghouse Calculation CN-RRA-02-38. Recent investigations suggest that these results may be too optimistic. A more reasonable approach may be implementing WCAP-16341, Simplified LERF Model, and characterizing the uncertainties based on that latest EPRI, PWROG, and NRC interactions.	LE-D5	Open – Will update the thermal induced SGTR model to incorporate new methodology from WCAP-16341.	None. Capability Category I is met and appropriate for this application. Refer to the Section "Assessment of PRA Capability Needed for Risk-Informed Inservice Inspection." This is not expected to have a significant impact on LERF.
LE-D6-01	The containment isolation analysis for BV2REV3b is based on a sub-atmospheric containment. BVPS-2 has been converted to atmospheric so this analysis must be revisited. BV1REV4 does account for the atmospheric containment conversion in the containment	LE-D6	Open – Will revise the BVPS-2 LERF notebook to account for the change to atmospheric containment.	None. Category I is met and appropriate for this application. This gap is a documentation consideration only.

[6] U.S. Nuclear Regulatory Commission, NUREG/CR-6595, "An Approach for Estimating the Frequencies of Various Containment Failure Modes and Bypass Events."

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	isolation notebook. The results of a similar assessment for BVPS-2 need to be incorporated in the LERF notebook.			
LE-E4-01	The BVPS-2 LERF model is quantified using RISKMAN. Only point-estimates for each top event are used and there are no uncertainty estimates or uncertainty propagation.	LE-E4	Open – Will develop database distributions for level 2 split fractions, so that a Monte Carlo quantification can be used for the LERF uncertainty propagation analysis. No significant impact to the LERF mean value is expected.	None. This gap is a documentation consideration only.
LE-F2-01	The PRA peer review team suggested in L2-02, using uncertainty analysis for the LERF top events to ensure that future applications are not affected by use of point estimates. This fact and observation was entered into FENOC's Corrective Action Program as CA 02-09043-26 to track and resolve the issues. The suggested PRA Peer Review Team resolution to this observation was not addressed in the BV2REV3B PRA model update, but will be evaluated sometime later in a future PRA model update. This update has not yet been completed. At the time, it was a "C" level fact and observation but the PRA standard raises the requirements for PRA quality and this fact and observation is now a "B" level.	LE-F2	Open – Will use the LERF uncertainty propagation analysis to identify key sources of uncertainty, then perform and document sensitivity studies for the significant LERF contributors. No significant impact to the LERF mean value is expected.	None. This gap is a documentation consideration only.
LE-G5-01	Limitations of the LERF analysis are identified throughout the BVPS-2 Level 2 notebook. However, they need to be gathered into a single location to facilitate future usage.	LE-G5	Open – Will document the limitations of the LERF analysis identified throughout the level 2 notebook into a single location to facilitate future usage.	None. This gap is a documentation consideration only.
HR-PR-001	BVPS does not have a written process for evaluating dependencies between multiple human error probabilities (HEPs) occurring in a single accident and does not provide a	HR-D5, HR-G7, HR-H3, HR-I1, HR-I2	Open – Will explicitly describe the process used to identify and evaluate the dependencies between multiple HEPs occurring	None. This gap is a documentation consideration only.

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	summary of those that were explicitly evaluated for dependencies and the associated levels of dependencies and joint HEPs.		in the same accident sequence.	
HR-PR-002	BVPS does not appear to have evaluated their HEPs for internal consistency consistent with the requirements of HR-G6 and does not have a documented process to do so.	HR-G6, HR-I2	Open – Will perform and document an explicit process for reviewing the HEPs for internal consistency with respect to scenario, context, procedures and timing.	None. This gap is a documentation consideration only.
HR-PR-003	The method for quantifying pre-initiator misalignment errors as described on page 8 of the "Beaver Valley Power Station Unit 2 PRA Notebook – Human Reliability Analysis," Revision 2, dated 10/01/07, relies on the use of a generic error of omission rate that does not reflect any detailed assessment of the human error probabilities. The process also does not consider the quality of plant-specific written procedures, administrative controls or the man-machine interface, and does not include an explicit assessment of the potential for recovery that specifically delineates which procedures and processes influence the potential for identification and recovery. Furthermore, the method for quantifying post-maintenance miscalibrations relies on a single generic error of omission rate. A complication in reviewing the pre-initiator human failure events (HFEs) was that the HRA notebook does not include a list of the pre-initiator HFEs or their probabilities. The system notebooks provide evidence of the search for and identification of misalignments but they do not present a list of such events or their probabilities.	HR-D2, HR-D3, HR-D4, HR-I1, HR-I2	Open – Use the EPRI HRA calculator to update and document the process for identification and quantification of pre-initiator HFEs.	None. This gap is a documentation consideration only.
HR-PR-005	The BVPS HRA is documented in the "Beaver Valley Power Station Unit 2 PRA Notebook – Human Reliability Analysis," Revision 2, dated	HR-I3	Open – Will add an "Assumptions" section to the HRA notebook and collect all of the high level	None. This gap is a documentation consideration only.

Facts and Observations	Description of Gap	Supporting Requirement	Current Status or Comment	Importance to Risk-Informed Inservice Inspection
	10/01/07. This notebook does not have an explicit assumptions section to identify and characterize assumptions. A review of this notebook revealed assumptions scattered throughout the text.		assumptions into it. Also, qualitatively characterize the potential impact of the assumptions and potential impact of alternate assumptions (if any) on the HRA analysis.	
HR-PR-007	In general, BVPS excludes virtually all miscalibration events based on the assumption that events related to instrument miscalibrations are captured in the equipment failure rate data and the on-line maintenance program precludes common-cause miscalibration by scheduling work on opposite trains in different weeks. Post-maintenance misalignments were excluded for normally operating system based on the assumption that misalignments on normally operating systems would be quickly detected and corrected. While these rules seem reasonable, they are applied to classes of maintenance and test activities to screen them from further consideration. This is sufficient for Capability Category I but not for Capability Category II.	HR-B1	Open – Will develop a table which lists the individual test, maintenance and calibration activities that were reviewed as potential pre-initiator human actions, and list the screening rule that was applied for each action that was screened from further consideration.	None. This gap is a documentation consideration only.

Notes for Table 1

1. **The variables used in the above equation [Cell IF-F2-02] for Rx are:**

Rx	=	annual frequency of a flood scenario in location x.
Fi	=	total annual frequency of the flood of any severity in building i.
fx,i	=	conditional frequency of the flood occurring in location x of building i, given that the flood has occurred in building i.
fs,x	=	severity factor; conditional frequency of the flood being of a severity to cause equipment failure.
fp,x	=	propagation factor; conditional frequency of the flood propagating to the adjacent locations, given that the flood occurred at location x with the severity specified to cause equipment failure (for localized cases, fp,x = 1.0).

The variables used in the above equation [Cell IF-F2-02] for Sx are:

Sx	=	the annual frequency of the scenario and recovery failure.
Rx	=	the annual frequency of the scenario before recovery.
Dx	=	the probability that timely detection of the flood fails. This includes consideration of detection capabilities (alarms, etc.), the likelihood of operator diagnosis, and time available.
Ix	=	the probability of not isolating or mitigating the flood prior to failing critical systems, given detection of the flood. This includes human actions.

2. **Reports are attached to the following FENOC letters:**

FENOC letter to NRC, January 25, 2006, Beaver Valley Power Station, Unit No. 1 and No. 2, BV-1 Docket No. 50-334, License No. DPR-66, BV-2 Docket No. 50-412, License No. NPF-73, Additional Information in Support of License Amendment Request Nos. 302 and 173.
[NRC Accession Number ML060330262]

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[NRC Accession Number ML060520569]