NRC FORM 464 Part I
(6-1998)

U.S. NUCLEAR REGULATORY COMMISSION

RESPONSE TO FREEDOM OF INFORMATION ACT (FOIA) / PRIVACY ACT (PA) REQUEST

FOIA/PA	RESPONSE NUMBER
2000-0186	4

RESPONSE TYPE: ✓ FINAL PARTIAL

REQUESTER: Paul Gunter

DATE: AUG 09 2000

PART I. -- INFORMATION RELEASED

No additional agency records subject to the request have been located.

Requested records are available through another public distribution program. See Comments section.

APPENDICES — Agency records subject to the request that are identified in the listed appendices are already available for public inspection and copying at the NRC Public Document Room.

✓ APPENDICES F, G — Agency records subject to the request that are identified in the listed appendices are being made available for public inspection and copying at the NRC Public Document Room.

Enclosed is information on how you may obtain access to and the charges for copying records located at the NRC Public Document Room, 2120 L Street, NW, Washington, DC.

✓ APPENDICES F, G — Agency records subject to the request are enclosed.

Records subject to the request that contain information originated by or of interest to another Federal agency have been referred to that agency (see comments section) for a disclosure determination and direct response to you.

We are continuing to process your request.

See Comments.

PART I.A -- FEES

AMOUNT *

$

* See comments for details

You will be billed by NRC for the amount listed.

You will receive a refund for the amount listed.

None. Minimum fee threshold not met.

✓ Fees waived.

PART I.B -- INFORMATION NOT LOCATED OR WITHHELD FROM DISCLOSURE

No agency records subject to the request have been located.

✓ Certain information in the requested records is being withheld from disclosure pursuant to the exemptions described in and for the reasons stated in Part II.

✓ This determination may be appealed within 30 days by writing to the FOIA/PA Officer, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001. Clearly state on the envelope and in the letter that it is a "FOIA/PA Appeal."

PART I.C COMMENTS (Use attached Comments continuation page if required)

SIGNATURE - FREEDOM OF INFORMATION ACT AND PRIVACY ACT OFFICER

Carol Ann Reed

NRC FORM 464 Part II (6-1998) U.S. NUCLEAR REGULATORY COMMISSION

RESPONSE TO FREEDOM OF INFORMATION ACT (FOIA) / PRIVACY ACT (PA) REQUEST

FOIA/PA 2000-0186

DATE AUG 0 9 2000

PART II.A -- APPLICABLE EXEMPTIONS

APPENDICES F

Records subject to the request that are described in the enclosed Appendices are being withheld in their entirety or in part under the Exemption No.(s) of the PA and/or the FOIA as indicated below (5 U.S.C. 552a and/or 5 U.S.C. 552(b)).

Exemption 1: The withheld information is properly classified pursuant to Executive Order 12958.

Exemption 2: The withheld information relates solely to the internal personnel rules and procedures of NRC.

Exemption 3: The withheld information is specifically exempted from public disclosure by statute indicated.

Sections 141-145 of the Atomic Energy Act, which prohibits the disclosure of Restricted Data or Formerly Restricted Data (42 U.S.C. 2161-2165).

Section 147 of the Atomic Energy Act, which prohibits the disclosure of Unclassified Safeguards Information (42 U.S.C. 2167).

41 U.S.C., Section 253(b), subsection (m)(1), prohibits the disclosure of contractor proposals in the possession and control of an executive agency to any person under section 552 of Title 5, U.S.C. (the FOIA), except when incorporated into the contract between the agency and the submitter of the proposal.

✓ Exemption 4: The withheld information is a trade secret or commercial or financial information that is being withheld for the reason(s) indicated.

✓ The information is considered to be confidential business (proprietary) information.

The information is considered to be proprietary because it concerns a licensee's or applicant's physical protection or material control and accounting program for special nuclear material pursuant to 10 CFR 2.790(d)(1).

The information was submitted by a foreign source and received in confidence pursuant to 10 CFR 2.790(d)(2).

Exemption 5: The withheld information consists of interagency or intraagency records that are not available through discovery during litigation. Applicable privileges:

Deliberative process: Disclosure of predecisional information would tend to inhibit the open and frank exchange of ideas essential to the deliberative process. Where records are withheld in their entirety, the facts are inextricably intertwined with the predecisional information. There also are no reasonably segregable factual portions because the release of the facts would permit an indirect inquiry into the predecisional process of the agency.

Attorney work-product privilege. (Documents prepared by an attorney in contemplation of litigation)

Attorney-client privilege. (Confidential communications between an attorney and his/her client)

Exemption 6: The withheld information is exempted from public disclosure because its disclosure would result in a clearly unwarranted invasion of personal privacy.

Exemption 7: The withheld information consists of records compiled for law enforcement purposes and is being withheld for the reason(s) indicated.

(A) Disclosure could reasonably be expected to interfere with an enforcement proceeding (e.g., it would reveal the scope, direction, and focus of enforcement efforts, and thus could possibly allow recipients to take action to shield potential wrongdoing or a violation of NRC requirements from investigators).

(C) Disclosure would constitute an unwarranted invasion of personal privacy.

(D) The information consists of names of individuals and other information the disclosure of which could reasonably be expected to reveal identities of confidential sources.

(E) Disclosure would reveal techniques and procedures for law enforcement investigations or prosecutions, or guidelines that could reasonably be expected to risk circumvention of the law.

(F) Disclosure could reasonably be expected to endanger the life or physical safety of an individual.

OTHER (Specify)

PART II.B -- DENYING OFFICIALS

Pursuant to 10 CFR 9.25(g), 9.25(h), and/or 9.65(b) of the U.S. Nuclear Regulatory Commission regulations, it has been determined that the information withheld is exempt from production or disclosure, and that its production or disclosure is contrary to the public interest. The person responsible for the denial are those officials identified below as denying officials and the FOIA/PA Officer for any denials that may be appealed to the Executive Director for Operations (EDO).

DENYING OFFICIAL	TITLE/OFFICE	RECORDS DENIED	APPELLATE OFFICIAL EDO	SECY	IG
Samuel J. Collins	Director, Office of Nuclear Reactor Regulation	Appendix F	✓		

Appeal must be made in writing within 30 days of receipt of this response. Appeals should be mailed to the FOIA/Privacy Act Officer, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, for action by the appropriate appellate official(s). You should clearly state on the envelope and letter that it is a "FOIA/PA Appeal."

APPENDIX F
RECORDS BEING WITHHELD IN PART

NO.	DATE	DESCRIPTION/(PAGECOUNT)/EXEMPTIONS
1.	10/05/95	Ltr to C. Carpenter, NRC from J.T. Beckham, Jr., Southern Nuclear Operating Co. Re: "Safety Assessment of BWR Reactor internals (BWRVIP-06)," EPRI TR-105707, September 1995. (2 pages PDR Acc. # 9510060367)
	10/03/95	Ltr to NRC from A. Kenny, EPRI, Re: Request for withholding of proprietary documents (7 pages PDR Acc. # 9510060371) **enclosed BWR Vessel and Internals Project Safety Assessment of BWR Reactor Internals BWRVIP-06 (155 pages) Ex. 4**
2.	09/28/95	Ltr to C. Carpenter, NRC from C. Terry, Niagara Mohawk Power Co. Re: "BWR Vessel and Inernals Project, BWR Reactor Pressure Vessel Shell Weld inspection recommendation (BWRVIP-05)," EPRI TR-105697, Sept. 1995 (2 pages PDR Acc. # 9510030130); 5/20/96 Ltr to Beckham from Sheron (2 pages) (Release)
		Letter to NRC from A. Kenny EPRI Re: Request for withholding of proprietary document (3 pages PDR Acc. # 9510030134) **enclosed BWR Vessel and Internals Project BWRVIP-05 (180 pages) Ex. 4** **The Second Request for Additional Information Regarding Report EPRI TR-105697 "BWR Vessel & Internals Project - BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations BWRVIP-05 September 1995 (33 pages) EX. 4**
3.	12/27/96	Ltr to C. Carpenter, NRC from C. Terry, Niagara Mohawk Power Co. Re: "BWR Vessel and Internals Project: Top Guide Inspection and Flaw Evaluation Guideline BWRVIP-26" (1 page PDR Acc. # 9701020293) **enclosed EPRI Report TR-107285 (77 pages) Ex. 4**

FOIA No. 2000-186

APPENDIX F
RECORDS BEING WITHHELD IN PART

NO.	DATE	DESCRIPTION/(PAGECOUNT)/EXEMPTIONS
4.	12/27/96	Ltr to C. Carpenter, NRC from C. Terry, Niagara Mohawk Power Co. Re: "BWRVIP - Top Guide Inspection and Flaw Evaluation Guideline BWRVIP-26, EPRI Report TR-107285, December, 1996 (2 pages PDR Acc. # 9701020293)
	12/13/96	Ltr to NRC from A. Kenny, EPRI Re: Request withholding of proprietary documents TR-107285 (4 pages PDR Acc. # 9701020297) **enclosed BWRVIP - Quantitative Safety Assessment of BWR Reactor Internals BWRVIP-09 (227 pages) Ex. 4**
5.	12/23/96	Ltr to C. Carpenter, NRC from C. Terrry, Niagara Mohawk Power Co. Re: "BWRVIP - Assessment of BWR Jet Pump Riser Elbow to Thermal Sleeve Weld Cracking BWRVIP-28 (2 pages PDR Acc. # 9701030006)
		Non proprietary version of BWRVIP EPRI Report TR-107667 (12 pages PDR Acc. # 9701030009) **enclosed proprietary version BWRVIP Assessment of BWR Jet Pump Riser Elbow to Thermal Sleeve Weld Cracking BWRVIP-28 (27 pages)** Ex. 4
6.	12/20/96	Letter to C. Carpenter, NRC from R. Dyle, Southern Nuclear Operating Co. Re: BWRVIP Response to NRC Request for additional information on BWRVIP-06 (1 page PDR Acc. # 9612240201) **enclosed BWRVIP Response to NRC requests for additional information (RAI) (8 pages) Ex. 4**

FOIA No. 2000-186

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NO.	DATE	DESCRIPTION/(PAGECOUNT)/EXEMPTIONS
7.	11/27/96	Ltr to C. Carpenter, NRC from C. Terry, Niagara Mohawk Power Co. Re: "BWR Vessel and Internals project, Roll/Expansion Repair of Control Rod Drive and In-Core Instrument Penetrations in BWR Vessels (BWRVIP-17)," EPRI Report TR-106712, November 1996 (2 pages PDR Acc. # 9612020131)
	11/27/96	Ltr to NRC from A. Kenny, EPRI Re: Request for withholding of proprietary document (3 pages PDR Acc. # 9612020132)
	11/96	Non-proprietary BWR Vessel and Internals Project Roll/Epansion Repair of Control Rod Drive and In-Core Instrument Penetrations in BWR Vessels BWRVIP-17(10 pages PDR Acc. # 9612020148) **enclosed Proprietary BWR Vessel and Internals Project Roll/Expansion Repair of Control Rod Drive and In-Core Instrument Penetrations inBWR Vessels BWRVIP-17 (62 pages) Ex. 4**
8.	08/12/96	Ltr to C. Carpenter, NRC from J.T. Beckman, Southern Nuclear Operating Co. Re: Non-Proprietary versionof BWR Vessel and Internals Project, BWR Core Spray Internals Inspection and Flaw Evaluation Guidelines BWRVIP-18 (4 pages release) **enclosed BWR Core Spray Internals Inspection and Flaw Evaluation Guidelines BWRVIP-18 (11 pages)**

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APPENDIX F
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NO.	DATE	DESCRIPTION/(PAGECOUNT)/EXEMPTIONS
9.	07/26/96	Ltr to C. Carpenter, NRC from C. Terry, Niagara Mohawk Power Co. Re: "BWR Vessel and Internals Project, BWR Core Spray Internals Inspection and Flaw Evaluation Guidelines (BWRVIP-18)," EPRI Report TR-106740, July 1996 (2 pages PDR Acc. # 9607300115)
	07/24/96	Ltr to NRC from A. Kenny, EPRI Re: Request for Withholding of Proprietary Document (4 pages PDR Acc. # 9607300128) **enclosed BWR Vessel and Internals Project BWR Core Spray Internals Inspection and Flaw Evaluation Guidelines BWRVIP-18 (105 pages) Ex. 4**
10.	06/24/96	Ltr to B. Sheron, NRC from R. Dyle, Southern Nuclear Operating Co. Re: BWRVIP Response to NRC requests for additional Information on BWRVIP-05 (2 pages PDR Acc.# 9606270004) **enclosed EPRI Report on TR-105697 "BWR Vessel and Internals Project - BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations BWRVIP-05, " September 1995 (35 pages) Ex. 4**
11.	12/06/99	Ltr to C Carpenter, NRC from V Wagoner, Carolina Power & Light Co re: BWRVIP Response to NRC Request for Additional Information on BWRVIP-55 (1 page) (Release) enclosed **BWRVIP Response to NRC Request for Additional Information on BWRVIPi Lower Plenum Repair Design Criteria BWRVIP-55 (4 pages) EX. 4**

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12.	04/16/96	Ltr to C. Carpenter, NRC from J. T. Beckham, Jr. Re: "BWR Vessel and Internals Project, Reactor Pressure Vessel and Internals Examination Guidelines BWRVIP-03, Section 6A, Standards for Visual Inspection of Core Spray Piping, Spargers, and Associated Components," EPRI Report TR-105696, February 1996.(2 pages PDR Acc. # 9604180166)
	04/16/96	Ltr to NRC from A. Kenny, EPRI Re: Request for withholding of Proprietary Documents for TR-105696, "BWR Vessel and Internals Project, Reactor Pressure Vessel and Internals Examination Guidelines BWRVIP-03, Section 6A, Standards for Visual Inspection of Core Spray Piping, Spargers, and Associated Components," EPRI Report, February 1996 (3 pages PDR Acc. # 9604180190) **enclosed BWRVIP-03 TR-105696 6A Standards for Visual Inspection of Core Spray Piping, Spargers, and Associated Components (13 pages) Ex. 4**
13.	04/02/96	Ltr to J. Beckham, Southern Nuclear Operating Co. from B. Sheron, NRC Re: BWR Owners' Group, RAI for Topical Report EPRI TR-105697, "BWR Vessel and Internals Projet, BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations BWRVIP-05" (1 page release) **enclosed RAI regarding EPRI Report TR-105697 "BWR Vessel and Internals Projet, BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations BWRVIP-05" (3 pages) Ex. 4**

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NO.	DATE	DESCRIPTION/(PAGECOUNT)/EXEMPTIONS
14.	03/18/96	Ltr to C. Carpenter, NRC from J.T. Beckham, Jr., Southern Nuclear Operating Co. Re: "BWR Vessel and Internals project, Evaluation of Crack Growth in BWR Stainless Steel RPV Internals BWRVIP-14, EPRI Report TR-105873, March 1996 (1 page PDR Acc. # 9603200399)
	03/14/96	Ltr to NRC from M. Edwards, EPRI Re: Request for Withholding of Proprietary Document (4 pages PDR Acc. # 9603200406) **enclosed Evaluation of Crack Growth in BWR Stainless Steel RPV Internals BWRVIP-14 (341 pages) Ex. 4**
15.	12/06/99	Ltr to C Carpenter, NRC from V Wagoner, Carolina Power & Light Co. re: BWRVIP Response to NRC Request for Additional Information on BWRVIP-53 (1 page) (Release) enclosed **BWRVIP Response to NRC Request for Additional Information on BWRVIP Standby Liquid Control Line Repair Design Criteria BWRVIP-53 (7 pages) EX. 4**
16.	12/18/97	Ltr to C. Carpenter, NRC from C. Terry, Niagara Mohawk Power Co. Re: BWRVIP response to NRC request for additional information on BWRVIP-05 (1 page PDR Acc. # 9712240085) **enclosed EPRI Report (108 pages) Ex. 4**

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17.	12/15/97	Ltr to C. Terry, Niagara Moahawk Power Co. from C. Carpenter, NRC Re: Review of "BWRVIP, Internal Core Spray Piping and Sparger Replacement Design Criteria BWRVIP-16 enclosed Memorandum to J. Coley, NRC/RII from W. Bateman, NRC/HQ Re: NUREG-0313 Category E Weldments and the Inservice Inspection Program (3 pages Release) **enclosed Proprietary request for additional information regarding EPRI Topical Report TR-106712, "BWRVIP, Internal Core Spray Piping and Sparger Replacement Design Criteria BWRVIP-16" (2 pages)**
18.	12/14/97	Ltr to C. Terry, Niagara Mohawk Power Co from C. Carpenter, NRC Re: Review of BWRVIP - Technical Basis for Part Circumference Weld Overlay Repair of Vessel Internal Core Spray Piping BWRVIP-34 (3 pages release) **enclosed EPRI Topical Report TR-108198 (3 pages) Ex. 4**
19.	12/11/97	Ltr to C. Carpenter, NRC from C. Terry, Niagara Mohawk Power Co. Re: Transmittal of "BWR Vessel and Internals Project: BWR LPCI Coupling Inspection and Flaw Evaluation Guidelines (BWRVIP-42)," EPRI Report TR-108726, December 1997 (1 page PDR Acc. # 9712170167) **enclosed BWR Vessel and Internals Project LPCI Coupling Inspection and Flaw Evaluation Guidelines BWRVIP-42 (96 pages) Ex. 4**

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NO.	DATE	DESCRIPTION/(PAGECOUNT)/EXEMPTIONS
20.	12/09/97	Memo to E. Sullivan, NRC from C. Carpenter, NRC Re: Meeting summary for November 5 and 6, 1997, Meetings with Boiling Water Reactors Vessel and Internals Project Technical Chairs, regarding BWRVIP-07 Safety Evaluation Report and to discuss issues related to BWR licensee vessel surveillance programs, relative to monitoring radiation embrittlement at BWR facilities (4 pages release) **enclosed BWRVIP's November 5, 1997, presentation viewgraphs (54 pages) Ex. 4**
21.	12/08/97	Ltr to C. Carpenter, NRC from V. Wagoner, Carolina Power and Light Re: BWRVIP Response to NRC RAI on BWRVIP-28 (1 page PDR Acc. # 9712100171) **enclosed BWRVIP-28 EPRI Topical Report T107667(37 pages) Ex. 4**
22.	11/26/97	Ltr to C. Carpenter, NRC from V. Wagoner, Carolina Power & Light Re: Response to NRC staff safety evaluation of the BWR vessel and internals projects BWRVIP-07 report (2 pages PDR Acc. # 9712010268) **enclosed Response to BWRVIP-07SE (22 pages) Ex. 4**
23.	11/26/97	Ltr to C. Carpenter, NRC from V. Wagoner, BWRVIP, Re: Response to NRC Staff Safety Evaluation of the BWR Vessel and Internals Project BWRVIP-07 Report (2 pages PDR Acc. # 9712010268) **enclosed Response to BWRVIP-07SE (22 pages) Ex. 4**

APPENDIX F
RECORDS BEING WITHHELD IN PART (con't)

NO.	DATE	DESCRIPTION/(PAGECOUNT)/EXEMPTIONS
24.	10/27/97	Ltr to C. Carpenter, NRC from C. Terry, Niagara Mohawk Power Co. Re: Proprietary version of "BWR Vessel and Internals Project, Underwater Weld Repair of Nickel Alloy Reactor Vessel Internals (BWRVIP-44," EPRI Report TR-108708, September 1997 (1 page PDR Acc. # 9711030055)
	10/27/97	Ltr to NRC from A. Kenny, EPRI Re: Request for withholding of proprietary document: "BWR Vessel and Internals Project, Underwater Weld Repair of Nickel Alloy Reactor Vessel Internals BWRVIP-44 (3 pages PDR Acc. # 9711030075) **enclosed BWR Vessel and Internals Project - Underwater Weld Repair of Nickel Alloy Reactor Vessel Internals BWRVIP-44 (99 pages) Ex. 4**
25.	10/27/97	Ltr to C. Carpenter, NRC from C. Terry, Niagara Mohawk Power Co. Re: Proprietary version of "BWR Vessel and Internals Project, Weldability of Irradiated LWR Structural Components (BWRVIP-45)," (1 page PDR Acc. # 9710300135)
	10/27/97	Ltr to NRC from A. Kenny, EPRI Re: Request for withholding of proprietary document: "BWR Vessel and Internals Project, Weldability of Irradiated LWR Structural Components BWRVIP-45 (3 pages PDR Acc. # 9710300140) **enclosed EPRI Report TR-108707, September 1997 (128 pages) Ex. 4**

APPENDIX F
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NO.	DATE	DESCRIPTION/(PAGECOUNT)/EXEMPTIONS
26.	10/15/97	Ltr to C. Carpenter, NRC from C. Terry, Niagara Mohawk Power Co. Re: "BWR Vessel and Internals Project, BWR Jet Pump Assembly Inspection and Flaw Evaluation Guidelines BWRVIP-41 (1 page PDR Acc. # 9710170021) Ltr to NRC from M. Fox, EPRI Power Research Inst. Re: TR-108728 Report "BWRVIP: BWR jet Pump Assembly Inspection and Flaw Evaluation Guidelines BWRVIP-41" (4 pages PDR Acc. # 9710170025) **enclosed BWR Vessel and Internals Project - BWR Jet Pump Assembly Inspection and Flaw Evaluation Guidelines BWRVIP-41 (174 pages) Ex. 4**
27.	10/08/97	Ltr to C. Carpenter, NRC from V. Wagoner, Carolina Power & Light Co. Re: BWRVIP Response to NRC request for additional information on BWRVIP-18 (2 pages PDR Acc. # 9710090293) **enclosed Response to NRC request for additional information regarding proprietary topical report EPRI TR-106740 (16 pages) Ex. 4**
28.	09/15/97	Ltr to C. Carpenter, NRC from C. Terry, Niagara Mohawk Power Co, Re: "BWRVIP - BWR Shroud Support Inspection and Flaw Evaluation Guidelines BWRVIP-38," EPRI Report TR-108823, September 1997 (1 page PDR Acc. # 9709170225) Ltr to NRC from M. Fox, EPRI Re: TR-108823 "BWRVIP: BWR Shround Support Inspection and Flaw Evaluation Guidelines BWRVIP-38, September, 1997 (4 pages PDR Acc. # 9709170227) **enclosed EPRI Report - BWR Shroud Support Inspection and Flaw Evaluation Guidelines BWRVIP-38 (122 pages) Ex. 4**

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NO.	DATE	DESCRIPTION/(PAGECOUNT)/EXEMPTIONS
29.	09/15/97	Ltr to C. Terry, Niagara Mohawk Power Co. from G. Lainas, NRC Re: Transmittal of NRC staff's safety evaluation of the BWR vessel and internals project BWRVIP-07 report (5 pages PDR Acc. # 9709240193); Safety Evaluation of EPRI Topical Report TR-105747 "BWR Vessel and Internals Project Guidelines for Reinspection of BWR core Shrouds (20 pages PDR Acc. # 9709240202) **enclosed BWR Vessel and internals project BWRVIP-07 (65 pages) Ex. 4**
30.	08/14/97	Ltr to C. Terry, Niagara Mohawk Power Co. from B. Sheron, NRC Re: Transmittal of NRC Staff's independent assessment of the Boiling Water Reactor Vessel and Internals Project BWRVIP-05 Report and Proprietary Request for additional information (4 pages release) **Enclosure 1: Independent Assessment by the Office of Nuclear Reactor Regulation related to the review of the Topical Report by the Boiling Water Reactor Vessel and Internals Project: "BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations" BWRVIP-05 (62 pages) (Release) Enclosure 2: Request for Additional Information Regarding Report EPRI TR-105697 (8 pages) EX. 4**
31.	07/28/97	Ltr to C. Carpenter, NRC from V. Wagoner, Carolina Power & Light Co. Re: BWRVIP Response to NRC Request for additional information on BWRVIP-14 (1 page PDR Acc. # 9707310036) **enclosed Responses to NRC request for additional Information on BWRVIP Report BWRVIP-14 on Evaluation of Crack Growth in RPV Intervals (53 pages) Ex. 4**
32.	07/07/97	Ltr to C. Carpenter, NRC from C. Terry, Niagara Mohawk Power Co. Re: Figures for BWRVIP Response to NRC Request for Additional Information on BWRVIP-03 (1 page PDR Acc. # 9707110153) **enclosed Figures for BWRVIP Response to NRC Request for Additional Information on BWRVIP-03 (17 pages) Ex. 4**

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33.	06/30/97	Ltr to C. Carpenter, NRC from C. Terry, Niagara Mohawk Power Co. Re: BWRVIP Response to NRC Request for additional Information on BWRVIP-03 (1 page PDR Acc. # 9707030004) **enclosed BWRVIP Response to "Proprietary Request for additional information regarding proprietary Topical Report EPRI TR-105696 (12 pages) Ex. 4**
34.	06/16/97	Ltr to C. Carpenter, NRC from C. Terry, Niagara Mohawk Power Co. Re: Proprietary version of "Configurations of Safety-Related BWR Reactor Inernals (BWRVIP-15)", March 1996 (1 page PDR Acc. # 9706230332)
	06/16/97	Ltr to NRC from M. Fox, EPRI Re: TR-106368 "BWRVIP Vessel and Internals Project: Configurations of Safety-Related BWR Reactor Internals (BWRVIP-15)", March 1996 (3 pages PDR Acc. # 9706230339) **enclosed BWR Vessel and Internals project Configurations of Safety-Related BWR Reactor Internals (BWRVIP-15) (411 pages) Ex. 4**
35.	06/13/97	Ltr to C. Carpenter, NRC from C. Terry, Niagara Mohawk Power Co. Re: Response to requests for additional information regarding BWRVIP recommendations for BWR Reactor Pressure Vessel Shell Weld Inspections (2 pages PDR Acc. # 9706180413) **enclosed EPRI Report (41 pages) Ex. 4**
36.	06/04/97	Ltr to C. Carpenter, NRC from C. Terry, Niagara Mohawk Power Co Re: Detailed Programming Information for VIPER (1 page PDR Acc. # 9706060278) **enclosed VIPER Computer Program Detailed Programming Information (16 pages) Ex. 4**
37.	05/27/97	Memo to J. Strosnider, NRC from C. Carpenter, NRC Re: April 29-30, 1997 meeting with Boiling Water Reactors Vessel & Internals Project BWRVIP regarding BWRVIP-06 (71 pages PDR Acc. # 9705300055) **enclosed Quantitative Safety Assessment (46 pages) Ex. 4**

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38.	05/22/97	Ltr to C. Carpenter, NRC from C. Terry, Niagara Mohawk Power Co. Re: "BWR Vessel and Internals Project Technical Basis for Part Circumference Weld Overlay Repair of Vessel Internal Core Spray Piping BWRVIP-34, May, 1997 (1 page PDR Acc. # 9705300146)
	05/21/97	Ltr to NRC from A. Kenny, EPRI Re: TR-108198 Report, "BWR Vessel and Internals Project Technical Basis for Part Circumference Weld Overlay Repair of Vessel Internal Core Spray Piping BWRVIP-34 (4 pages PDR Acc. # 9705300149) **enclosed EPRI Report TR-108198 Technical Basis for Part Circumference Weld Overlay Repair of Vessel Internal Core Spray Piping BWRVIP-34 (192 pages) Ex. 4**
39.	05/16/97	Ltr to C. Carpenter, NRC from P. C. Riccardella, Structural Integrity Assoc. Inc. Re: Transmittal of BWRVIP VIPER Computer Code (1 page release) **enclosed Printout from VIPER computer code (19 pages) Ex. 4**
40.	04/25/97	Ltr to C. Carpenter, NRC from C. Terry, Niagara Mohawk Power Co. Re: "BWRVIP Standby Liquid Control System/Core Plate Inspection and Flaw Evaluation Guidelines BWRVIP-27", April 1997 (5 pages PDR Acc. # 9705010093)
		Non-proprietary version of BWRVIP - BWR Standby Liquid Control System/Core Plate Inspection and Flaw Evaluation Guidelines BWRVIP-27(7 pages PDR Acc. # 9705010101) **enclosed BWRVIP - BWR Standby Liquid Control System/Core Plate Inspection and Flaw Evaluation Guidelines BWRVIP-27 (46 pages) Ex. 4**

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41.	04/24/97	Ltr to C. Terry, Niagara Mohawk Power Co. from C. Carpenter, NRC Re: BWRVIP - Assessment of BWR Jet Pump Riser Elbow to Thermal Sleeve Weld Cracking BWRVIP-28 (2 pages release) **enclosed Request for additional information regarding BWRVIP-28 EPRI Report TR-107667 (5 pages) Ex. 4**
42.	03/18/97	Ltr to C. Carpenter, NRC from C. Terry, Niagara Mohawk Power Co. Re: "BWRVIP Vessel and Internals Project Internal Core Spray Piping and Sparger Replacement Design Criteria BWRVIP-16 (1 page PDR Acc. # 9703260101)
	03/21/97	Ltr to NRC from A. Kenny, EPRI Re: Request for withholding of proprietary documents; TR-106708, "BWRVIP Core Spray Piping and Sparger Replacement Design Criteria BWRVIP-16 (3 pages PDR Acc. # 9703260107) **enclosed EPRI Report TR-106708, March 1997 (51 pages) Ex. 4**
43.	03/14/97	Ltr to C. Terry, Niagara Mohawk Power Co. from C. Carpenter, NRC Re: Proprietary request for additional information - Review of BWR Vessel and Internals Project Reports, "Core Plate Inspection and Flaw Evaluation Guideline," and "Top Guide Inspection and Flaw Evaluation Guideline" (2 pages release) **enclosed EPRI Report TR-107284 (5 pages) Ex. 4**

APPENDIX F
RECORDS BEING WITHHELD IN PART

NO.	DATE	DESCRIPTION/(PAGECOUNT)/EXEMPTIONS
44.	03/12/97	Ltr to C. Terry, Niagara Mohawk Power Co. from C. Carpenter, NRC Re: Proprietary Request for Additional Information - Review of BWR Vessel and Internals Project Reports, "BWR Vessel and Internals Project, Reactor Pressure Vessel and Internals Examination Guidelines" BWRVIP-03 (2 pages release) **enclosed Proprietary Topical Report EPRI TR-105696 (3 pages) Ex. 4**
45.	02/24/97	Ltr to C. Carpenter, NRC from B. McLeod, BWRVIP Re: BWRVIP Response to NRC Request for Additional Information on BWRVIP-19 (1 page PDR Acc. # 9702270196) **enclosed BWRVIP response to the Requests for additional information on BWRVIP-19 (2 pages) Ex. 4**
46.	01/22/97	Ltr to C. Terry, Niagara Moahawk Power Co. from C. Carpenter, NRC Re: Review of "BWR Core Spray Internals Inspecton and Flaw Evaluation Guidelines BWRVIP-18," and "Core Spray Piping and Sparger Repair Design Criteria BWRVIP-19 (1 page realease) **enclosed Request for additional information regarding proprietary Topical Report EPRI TR-105873, "BWR Vessel and Internals Project, BWR Core Spray Internals Inspection and Flaw Evaluation Guidelines BWRVIP-18 (5 pages)** EX4

APPENDIX F
RECORDS BEING WITHHELD IN PART (con't)

NO.	DATE	DESCRIPTION/(PAGECOUNT)/EXEMPTIONS
47.	12/31/98	Ltr to C. Carpenter, NRC from C. Terry, Niagara Mohawk Power Co. Re: "BWR Vessel and Internals Project: Technical Basis for Inspection Relief for BWR Internal Components with Hydrogen Injection BWRVIP-62 (1 page PDR Acc. # 9901050050)
	12/21/98	Ltr to NRC from A. Kenny, EPRI Re: TR-108705, "BWRVIP Technical Basis for Inspection Relief for BWR Internal Components with Hydrogen Injection BWRVIP-62, December 1998 (3 pages PDR Acc. # 9901050055) **enclosed EPRI Report TR-108705, December 1998 (127 pages) Ex. 4**
48.	12/23/98	Ltr to C. Carpenter, NRC from C. Terry, Niagara Mohawk Power Co. Re: "BWR Vessel and Internals Project: Evaluation of Crack Growth in BWR Nickel Base Austenitic Alloys in RPV Internals BWRVIP-59 (1 page PDR Acc. # 9812290036)
	12/17/98	Ltr to NRC from A. Kenny, EPRI Re: "BWR Vessel and Internals project, Evaluation of Crack Growth in BWR Nickel Base Austenitic Alloys in RPV Internals BWRVIP-59 (3 pages release) **enclosed EPRI Report TR-108710, December 1998 (292 pages) Ex. 4**

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RECORDS BEING WITHHELD IN PART

NO.	DATE	DESCRIPTION/(PAGECOUNT)/EXEMPTIONS
49.	12/22/98	Ltr to C. Carpenter, NRC from C. Terry, Niagara Mohawk Power Co. Re: "BWR Vessel and Internals Projec: CRD Internal Access Weld Repair BWRVIP-58," EPRI Report TR-108703, December 1998 (1 page release)
	11/13/98	Ltr to NRC from A. Kenny, EPRI Re: TR-108703 "BWRVIP Vessel and Internals Project: CRD Internal Access Weld Repair BWRVIP-58, December, 1998 (3 pages PDR Acc. # 9812280134) **enclosed EPRI Report on BWRVIP CRD Internal Access Weld Repair BWRVIP-58 Ex. 4**
50.	12/16/98	Ltr to C. Carpenter, NRC from C. Terry, Niagara Mohawk Power Co. Re: "BWR Vessel and Internals Project Instrument penetration Repair Design Criteria (BWRVIP-57)," EPRI Report TR-1087121, December, 1998 (1 page PDR Acc. # 98121800167)
	11/09/98	Ltr to NRC from A. Kenny, EPRI Re: TR-108721 "BWRVIP Vessel and Internals Project: Instrument Penetration Repair Design Criteria (BWRVIP-57)," December 1998 (3 pages PDR Acc. # 9812180169) **enclosed BWR Vessel and Internals Project Instrument Penetration Repair Design Criteria BWRVIP-57 (52 pages) EX 4**
51.	12/15/98	Ltr to C. Carpenter, NRC from V. Wagoner, Carolina Power & Light Co. Re: Response to NRC request for additional information on BWRVIP-05 dated June 8, 1998 (1 page PDR Acc. # 9812170067) **enclosed BWRVIP-05 - Response to NRC Axial weld RAI (20 pages) Ex. 4**

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NO.	DATE	DESCRIPTION/(PAGECOUNT)/EXEMPTIONS
52.	11/24/98	Ltr to C. Carpenter, NRC from C. Terry, Niagara Mohawk Power Co. Re: BWRVIP Response to NRC Safety Evaluation of BWRVIP-14 (1 page PDR Acc. # 9811250229) **enclosed EPRI Report on Response to NRC Safety Evaluation of the BWR Vessel and Internals Project BWRVIP-14 Report TR-105873 (14 pages) Ex. 4**
53.	11/24/98	Ltr to C. Carpenter, NRC from V. Wagoner, Carolina Power & Light Co. Re: BWRVIP Response to NRC request for Additional Information on BWRVIP-38 (2 pages release) **enclosed Response to reqeusted information regarding BWRVIP-38 (6 pages) Ex. 4**
54.	11/16/98	Ltr to C. Carpenter, NRC from C. Terry, Niagara Mohawk Power Co Re: "BWRVIP: LPCI Coupling Repair Design Criteria BWRVIP-56," EPRI Report TR-108717, November 1998 (1 page PDR Acc. # 9811200237) Ltr to Document Control Desk from A Kenny (3 pages PDR Acc. # 9811200237) Non-proprietary BWRVIP: LPCI Coupling Repair Design Criteria BWRVIP-56 (11 pages) **enclosed BWRVIP - LPCI Coupling Repair Design Criteria BWRVIP-56 (63 pages) Ex. 4**

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NO.	DATE	DESCRIPTION/(PAGECOUNT)/EXEMPTIONS
55.	09/22/98	Ltr to C. Carpenter, NRC from C. Terry, Niagara Mohawk Power Co. Re: "BWRVIP Lower Plenum Repair Design Criteria BWRVIP-55" EPRI Report TR-108719, September 1998 (1 page PDR Acc. # 9810020127) Ltr to NRC from A. Kenny, EPRI Re: TR-108719 "BWRVIP: Lower Plenum Repair Design Criteria BWRVIP-55, September, 1998 (3 pages PDR Acc. # 9810020136) **enclosed BWRVIP Lower Plenum Repair Design Criteria BWRVIP-55 (75 pages) Ex. 4**
56.	09/08/98	Letter to C. Carpenter, NRC from V. Wagoner, Carolina Power and Light Re: BWRVIP Response to NRC Request for Additional Information on BWRVIP-42 (1 page PDR Acc. # 9809100223) **enclosed EPRI Topical Report TR-108726 (4 pages) Ex. 4**
57.	07/13/98	Ltr to C. Carpenter, NRC from C. Terry, Niagara Mohawk Power Co. Re: BWRVIP Response to NRC RAI on BWRVIP-27 (1 page PDR Acc. # 9807200072) **enclosed EPRI Report - Proprietary request for additional information regarding EPRI Topical Report TR-106712 (4 pages) Ex. 4**

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NO.	DATE	DESCRIPTION/(PAGECOUNT)/EXEMPTIONS
58.	07/02/98	Ltr to C. Carpenter, NRC from C. Terry, Niagara Mohawk Power Co. Re: "BWRVIP: Standby Liquid Control Line Repair Design Criteria BWRVIP-53," EPRI Report TR-108716, July 1998 (1 page PDR Acc. # 9807090071)
	06/16/98	Ltr to NRC from A. Kenny, EPRI Re: TR-108716 " BWRVIP: Standby Liquid Control Line Repair Design Criteria BWRVIP-53, July 1998 (4 pages PDR Acc. # 9807090073) **enclosed EPRI Report: Standby Liquid Control Line Repair Design Criteria BWRVIP-53 (87 pages)** Ex 4
59.	06/26/98	Ltr to C. Carpenter, NRC from C. Terry, Niagara Mohawk Power Co. Re: "BWRVIP: Shroud Support and Vessel Bracket Repair Design Criteria BWRVIP-52" EPRI Report TR-108720, June 1998 (1 page PDR Acc. # 9806300339)
	06/16/98	Ltr to NRC from A. Kenny, EPRI Re: TR-108720 "BWRVIP: Shroud Support and Vessel Bracket Repair Criteria BWRVIP-52, June 1998 (4 pages PDR Acc. # 9806300341) **enclosed EPRI Report on BWRVIP Shroud Support and Vessel Bracket Repair Design Criteria BWRVIP-52 (59 pages)** Ex 4

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NO.	DATE	DESCRIPTION/(PAGECOUNT)/EXEMPTIONS
60.	05/14/98	Ltr to C. Carpenter, NRC from V. Wagoner, Carolina Power & Light Co. Re: BWRVIP: Top Guide/Core Plate Repair Design Criteria BWRVIP-50 (1 page PDR Acc. # 9805210238)
	04/27/98	Ltr to NRC from M. Fox, EPRI Re: TR-108722 "BWRVIP Vessel and Internals Project: Top Guide/Core Plate Repair Design Criteria BWRVIP-50 (4 pages PDR Acc. # 9805210244) **enclosed EPRI Report for Top Guide/Core Plate Repair Design Criteria BWRVIP-50 (50 pages) Ex. 4**
61.	05/14/98	Ltr to C. Carpenter, NRC from C. Terry, Niagara Mohawk Power Co. Re: "BWR Vessel and Internals Project: Jet Pump Repair Design Criteria BWRVIP-51" (1 page PDR Acc. # 9805210208)
	04/27/98	Ltr to NRC from M. Fox, EPRI Re: TR-108718 "BWRVIP: Jet Pump Repair Design Criteria BWRVIP-51, May, 1998 (4 pages PDR Acc. # 9805210216) **enclosed EPRI Report onJet Pump Repair Design Criteria BWRVIP-51 (81 pages) Ex.4**
62.	04/14/98	Ltr to C. Terry, Niagara Mohawk Power Co. from C. Carpenter, NRC Re: Proprietary request for additional information - Review of "BWR Vessel and Internals Project, BWR LPCI Coupling Inspection and Flaw Evaluation Guidelines BWRVIP-42" (2 pages release) **enclosed EPRI Topical Report TR-108726, "BWR Vessel and Internals Project, BWR LPCI Coupling Inspection and Flaw Evaluation Guidelines BWRVIP-42" (1 page) Ex. 4**

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NO.	DATE	DESCRIPTION/(PAGECOUNT)/EXEMPTIONS
63.	03/30/98	Ltr to C. Carpenter, NRC from V. Wagoner, BWRVIP Re: BWRVIP Response to NRC RAI on BWRVIP-34 (2 pages PDR Acc.# 9804060298) **enclosed EPRI Topical Report TR-10898, "BWR Vessel and Internals Project, Technical Basis for part Circumference Weld Overlay Repair of Vessel Internal Core Spray Piping BWRVIP-34) (17 pages) Ex. 4**
64.	03/27/98	Ltr to C. Carpenter, NRC from V. Wagoner, Carolina Power and Light Re: BWRVIP Response to NRC RAI on BWRVIP-16 (2 pages PDR Acc. # 9804060311) **enclosed Proprietary response to RAI regarding EPRI Topical Report TR-106712 (12 pages) Ex. 4**
65.	03/13/98	Ltr to C. Carpenter, NRC from C. Terry, Niagara Mohawk Power Co. Re: "BWR Vessel and Internals Project: Instrument Penetration Inspection and Flaw Evaluation Guidelines BWRVIP-49 (2 pages PDR Acc. # 9803190146)
	02/19/98	Ltr to NRC from M. Fox, EPRI Re: TR-108695 "BWRVIP BWR Instrument Penetration Inspection and Flaw Evaluation Guidelines BWRVIP-49," March 1998 (3 pages PDR Acc. # 9803190161) **enclosed EPRI Report TR-108695 (39 pages) Ex. 4**
66.	01/12/98	Ltr to C. Carpenter, NRC from V. Wagoner, Carolina Power & Light Co. Re: Missing Page from BWRVIP response to NRC request for additional information on BWRVIP-05 (1 page PDR Acc. # 9801150240) **enclosed Insertion A-11 for the response to BWRVIP-05 dated, December 18, 1997 (1 page) Ex. 4**

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NO.	DATE	DESCRIPTION/(PAGECOUNT)/EXEMPTIONS
67.	01/11/99	Ltr to C. Carpenter, NRC from C. Terry, Niagara Mohawk Power Co. Re: Project No. 704 - BWRVIP response to NRC Safety Evaluation of BWRVIP-18 (2 pages PDR Acc. # 9901140122) **enclosed BWRVIP Response to issues in NRC Safety Evaluation of BWRVIP-18 (9 pages) Ex. 4**
68.	01/29/99	Ltr to C. Carpenter, NRC from C. Terry, Niagara Mohawk Power Co. Re: Transmittal of "BWR Vessel and Internals Induction Heating Stress Improvement Effectiveness on Crack Growth in Operating Plants BWRVIP-1, January 1999 (1 page release) **enclosed EPRI Report - Induction Heating Stress Improvement Effectiveness on Crack Growth in Operating Plants BWRVIP-61 (53 pages) Ex. 4**
69.	02/12/99	Ltr to C. Terry, Niagara Mohawk Power Co from C. Carpenter, NRC Re: Proprietary request for additional information - Review of "BWR Vessel and Internals Project, Jet Pump Assemble Inspection and Flaw Evaluation guidelines (3 pages PDR Acc. # 9902180165) **enclosed RAI Safety Evaluation related to BWRVIP-41 (3 pages) Ex. 4**
70.	3/16/99	Ltr to C. Carpenter, NRC from C. Terry, Niagara Mohawk Power Co. Re: "BWR Vessel and Internals Project, Review of Test Data for Irradiated Stainless Steel Components BWRVIP-66 (1 page PDR Acc. # 9903290131)
	03/05/99	Ltr to NRC from M. Fox, EPRI Re: TR-112611, "BWRVIP - Review of Test Data for Irradiated Stainless Steel Components BWRVIP-66 (3 pages PDR Acc. # 9903180358) **enclosed BWRVIP - Review of Test Data for Irradiated Stainless Steel Components BWRVIP-66 (38 pages) Ex. 4**

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NO.	DATE	DESCRIPTION/(PAGECOUNT)/EXEMPTIONS
71.	03/30/99	Ltr to C. Carpenter, NRC from C. Terry, Niagara Mohawk Power Co. Re: Response to Safety Evaluation of "BWR Vessel and Internals Project: Reactor Pressure Vessel and Internals Examination Guidelines BWRVIP-03," EPRI Report TR-105696, October 1995 2 pages release) **enclosed BWRVIP Response to Issues in NRC Safety Evaluation of "BWR Vessel and Internals Project: Reactor Pressure Vessel and Internals Examination Guidelines BWRVIP-03," (6 pages) Ex. 4**
72.	03/30/99	Ltr to C. Carpenter, NRC from C. Terry, Niagara Mohawk Power Co. Re: "BWR Vesel and Internals Project: Reactor Pressure Vessel and Internals Examination Guidelines BWRVIP-03, Rev. 1, "EPRI Report TR-105696R1, March 1999 (2 pages release) **enclosed BWR Vessel and Internals Project: Reactor Pressure Vessel and Internals Examination Guidelines BWRVIP-03 Revision 1 (337 pages) Ex. 4**
73.	04/07/99	Ltr to C. Terry, Niagara Mohawk Power Co. from C. Carpenter, NRC Re: Proprietary RAI - Review of "BWRVIP - Standby Liquid Control Line Repair Design Criteria BWRVIP - 53 (3 pages PDR Acc. # 9904130085) **enclosed Proprietary RAI related to "Standby Liquid Control Line Repair Design Criteria BWRVIP-53" (3 pages)** EX4
74.	04/07/99	Ltr to C. Terry, Niagara Mohawk Power Co. from C. Carpenter, NRC Re: "BWRVIP, Top Guide/Core Plate Repair Design Criteria BWRIP-50 (3 pages PDR Acc. # 9904130049) **enclosed RAI Safety Evaluation related to BWRVIP-50 "BWR Top Guide/Core Plate Repair Design Criteria" (2 pages)** EX4

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NO.	DATE	DESCRIPTION/(PAGECOUNT)/EXEMPTIONS
75.	04/08/99	Ltr to C. Terry, Niagara Mohawk Power Co. from C. Carpenter, NRC Re: Review of "BWR Vessel and Internals Project, Jet Pump Repair Design Criteria BWRVIP-51 (3 pages PDR Acc. # 9904140175) **enclosed Request for additional information related to BWR Vessel and Internals Project, Jet Pump Repari Design Criteria BWRVIP-51 (1 page) Ex. 4**
76.	04/29/99	Ltr to C. Terry, Niagara Mohawk Power Co. from J. Strosnider, NRC Re: Staff's comments on the BWR Vessel and Internals Project's Response to the NRC's request for additional information on the inspection of BWR Axial Shell welds (3 pages PDR Acc. # 9905060026) **enclosed Staff's comments on BWRVIP response to NRC RAI on axial welds (5 pages) Ex. 4**
77.	05/10/99	Ltr to C. Carpenter, NRC from C. Terry, Niagara Mohawk Power Co. Re: Project No. 704 - BWRVIP Response to Item 15 in NRC RAI on BWRVIP-38 (2 pages PDR Acc. # 9905140198) **enclosed Response to request for additional information regarding proprietary topical report EPRI TR-108823 (3 pages) Ex. 4**
78.	06/30/99	Ltr to NRC from M. Fox, EPRI Re: TR-113170, "BWRVIP Vessel and Internals Project, shroud vertical weld inspection and evaluation guidelines (BWRVIP-63), " June 1999 (3 pages PDR Acc. # 9907080221) **enclosed BWR Vessel and Internals Project shroud Vertical Weld Inspection and Evaluation Guidelines BWRVIP-63 (49 pages) Ex. 4**

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NO.	DATE	DESCRIPTION/(PAGECOUNT)/EXEMPTIONS
79.	08/04/99	Ltr to C. Carpenter, NRC from V. Wagoner, BWRVIP, Re: BWRVIP Response to NRC RAI on BWRVIP-41 (1 page PDR Acc. # 9908060230) **enclosed BWRVIP Response to RAI Safety Evaluation to BWRVIP-41 (9 pages) Ex. 4**
80.	08/05/99	Ltr to C. Carpenter, NRC from V. Wagoner, BWRVIP Re: BWRVIP Response to NRC Safety Evaluation on BWRVIP-26 (1 page PDR Acc. # 9908090118) **enclosed BWRVIP response to NRC Safety evaluation of BWRVIP-26 (2 pages) Ex. 4**
81.	08/13/99	Ltr to C. Terry, Niagara Mohwak Power Co from C. Carpenter Re: Proprietary request for additional information - Review of EPRI Topical Reports TR-10872, TR-108719, and TR-108721 (2 pages release) **enclosed Proprietary request for additional information - Review of EPRI Topical Reports (2 pages) Ex. 4**
82.	12/06/99	Ltr to C Carpenter, NRC from V Wagoner, Carolina Power & Light Co re: BWRVIP Response to NRC Request for Additional Information on BWRVIP-52 (1 page) (Release) enclosed **BWRVIP Response to NRC Request for Additional Information on Shroud Support Repair Design Criteria BWRVIP-52 (2 pages) EX. 4**

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NO.	DATE	DESCRIPTION/(PAGECOUNT)/EXEMPTIONS
83.	09/21/99	Ltr to C. Carpenter, NRC from C. Terry, Niagara Mohawk Power Co. Re: Project 704 - "BWRVIP Vessel and Internals Project, BWR Reactor Pressure Vessel Inspection and Flaw Evaluation Guidelines BWRVIP-74," EPRI Report TR-113596, September 1999 (1 page PDR Acc. # 9909240153)
	09/14/99	Ltr to NRC from M. Fox, EPRI, Re: TR-113596, "BWR Vessel and Internals Project, BWR Reactor Pressure Vessel Inspection and Flaw Evaluation guidelines BWRVIP-74," September 1999 (3 pages PDR Acc. # 9909240156) **enclosed BWR Vessel and Internals Project BWR Reactor Pressure Vessel Inspection and Flaw Evaluation Guidelines BWRVIP-74 (126 pages) Ex. 4**
84.	10/21/99	Ltr to C. Carpenter, NRC from C. Terry, Niagara Mohawk Power Co. Re: Project No. 704 - BWRVIP Response to NRC Safety Evaluation of BWRVIP-42 (1 page PDR Acc. # 9910270139) **enclosed BWRVIP Response to NRC Safety Evaluation on BWRVIP-42 (3 pages) Ex. 4**
85.	11/04/99	Memo to PWR MRP Reactor Internals Issue Task Group from T. Mulford, EPRI Re: PWR MRP use of BWRVIP EVT-1 Information (2 pages release) **enclosed BWRVIP: Reactor Pressure Vessel and Internals Examination Guidelines BWRVIP-03, Revision 1(13 pages) Ex. 4**
86.	11/12/99	Ltr to C. Carpenter, NRC from V. Wagoner, Carolina Power & Light Co. Re: Response to NRC Comments on BWRVIP Analyses of BWR Reactor Pressure Vessel Axial Shell Welds (1 page release) **enclosed BWRVIP-05 Resolution of Axial Weld Issue (7 pages) Ex. 4**

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NO.	DATE	DESCRIPTION/(PAGECOUNT)/EXEMPTIONS
87.	11/29/99	Ltr to C. Terry, Niagara Mohawk Power Co. from C. Carpenter, NRC Re: ON EPRI TR-108710, "Evaluation of Crack Growth in BWR Nickel Base Austenitic Alloys in RPV Internals BWRVIP-59 (2 pages release) **enclosed Requests for additional information on EPRI TR-108710, Evaluation of Crack Growth in BWR Nickel Base Austenitic Alloys in RPV Internals BWRVIP-59 (4 pages)** Ex 4
88.	12/06/99	Ltr to C Carpenter, NRC from V Wagoner, Carolina Power & Light Co Re: BWRVIP Response to NRC Safety Evaluation on BWRVIP-16 and BWRVIP-19 (1 page) (Release) enclosing **BWRVIP Response to NRC Safety Evaluation of BWRVIP Core Spray Piping and Sparger Replacement Design Criteria BWRVIP-16 and BWRVIP Core Spray Piping and Sparger Repair Design Criteria BWRVIP-19 (5 pages) EX. 4**
89.	12/06/99	Ltr to C. Carpenter, NRC from V. Wagoner, Carolina Power & Light Re: BWRVIP Response to NRC request for additional Information on BWRVIP-57 (1 page release) **enclosed BWRVIP Response to NRC request for Additional Information on Instrument Penetration Repair Design Criteria BWRVIP-57 (5 pages) Ex. 4**
90.	12/06/99	Ltr to C. Carpenter, NRC from V. Wagoner, Carolina Power & Light Co. Re: BWRVIP Response to NRC Request for Additional Information on BWRVIP-50 (1 page release) **enclosed BWRVIP Response to NRC Request for Additional Information on BWRVIP Top Guide/Core Plate Repair Design Criteria (7 pages) Ex. 4**

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NO.	DATE	DESCRIPTION/(PAGECOUNT)/EXEMPTIONS
91.	12/06/99	Ltr to C. Carpenter, NRC from V. Wagoner, Carolina Power & Light Co. Re: BWRVIP Response to NRC Request for Additional Information on BWRVIP-51 (1 page release) **enclosed BWRVIP Response to Request for additional Information related to BWRVIP Jet Pump Repair Design Criteria BWRVIP-51 (5 pages) Ex. 4**

APPENDIX G
RECORDS BEING RELEASED IN THEIR ENTIRETY

NO.	DATE	DESCRIPTION/(PAGE COUNT)
1.	12/23/97	Ltr to NRC from C Terry, BWRVIP enclosing BWR Vessel and Internals Project, Update of Bounding Assessment of BWR/2-6 Reactor Pressure Vessel Integrity Issues (BWRVIP-46) (53 pages)
2.	9/15/99	Ltr to R Dalton from J Strosnider, Subject: Authorization of Task Order 5, "Review of EPRI Topical Report TR-108705, BWR Vessel and Internals project - Technical Basis for Inspection Relief for BWR Internal Components with Hydrogen Injection (BWRVIP-62) Under JCN J-2410 (7 pages)
3.	12/3/99	Ltr to C Terry from J Strosnider, Subject: Final Safety Evaluation of Proprietary Report TR 105873 BWR Vessel and Internals Project, Evaluation of Crack Growth in BWR Stainless Steel Internals (BWRVIP-14) (TAC No. M94975) (11 pages)

97-958

BWRVIP

BWR Vessel & Internals Project — *Issue Management and Resolution*

December 23, 1997

Document Control Desk
U. S. Nuclear Regulatory Commission
11555 Rockville Pike
Rockville, MD 20852-2738

Attention: C. E. Carpenter

Subject: "BWR Vessel and Internals Project, Update of Bounding Assessment of BWR/2-6 Reactor Pressure Vessel Integrity Issues (BWRVIP-46)," EPRI Report TR-109727, December 1997.

Reference: "BWR Vessel and Internals Project, Bounding Assessment of BWR/2-6 Reactor Pressure Vessel Integrity Issues (BWRVIP-08NP)," EPRI Report TR-105908NP, November 1995.

Enclosed are 10 copies of the document "BWR Vessel and Internals Project, Update of Bounding Assessment of BWR/2-6 Reactor Pressure Vessel Integrity Issues (BWRVIP-46)," EPRI Report TR-109727, December 1997.

The enclosed document is provided as an updated generic BWR response to Generic Letter 92-01, Revision 1, Supplement 1. The BWRVIP-08NP document referenced above provided an initial response in November 1995 to the issues identified in the Generic Letter supplement. Since that time, additional data has been retrieved by owners groups, nuclear steam supply system vendors and the BWRVIP. The enclosed document examines these new data and considers the impact on the results in BWRVIP-08NP for each U.S. BWR vessel if bounding chemistries are assumed for vessel beltline welds.

The BWRVIP utilities have approved transmittal of the enclosed report as a generic response that addresses all U.S. BWR/2-6 plants. However, this transmittal does not represent a commitment for any utility. Each utility must formally endorse the BWRVIP position for it to become that utility's position.

If you have any questions on this subject please call Dana Covill of GPU Nuclear, BWRVIP Assessment Committee Technical Chairman, at (973) 316-7525.

Sincerely,

9712290087 971223
PDR TOPRP EXIEPRI
C PDR

Carl Terry

240022

Carl Terry
Niagara Mohawk Power Company
Chairman, BWR Vessel and Internals Project

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Reply To: Carl Terry, BWRVIP Chairman, Niagara Mohawk Power Company, P. O. Box 63, Lycoming, NY 13093 • Phone: (315) 349-7283 • Fax: (315) 349-4753

EPRI

Keywords:
Boiling water reactor
Reactor pressure vessel
Safety assessment
Stress corrosion cracking
Vessel and internals
Welds

EPRI TR-109727
December 1997

BWR Vessel and Internals Project

Update of Bounding Assessment of BWR/2-6 Reactor Pressure Vessel Integrity Issues (BWRVIP-46)

Prepared by
ATI Consulting

F115

REPORT SUMMARY

BWR Vessel and Internals Project

Update of Bounding Assessment of BWR/2-6 Reactor Pressure Vessel Integrity Issues (BWRVIP-46)

The Boiling Water Reactor Vessel and Internals Project (BWRVIP), formed in June 1994, is an association of utilities focused exclusively on BWR vessel and internals issues. This report provides an updated response to NRC Generic Letter 92-01, Revision 1, Supplement 1, that requested all U.S. light water reactor owners to review all available data relevant to reactor vessel integrity issues, establish best estimate chemistries and assess the impact to vessel integrity issues of interest. This report provides an updated assessment based on additional data that has become available since the publication of the previous BWRVIP response in November 1995 (BWRVIP-08NP).

INTEREST CATEGORIES

Piping, reactor vessel and internals
Licensing and safety assessment

KEYWORDS

Boiling water reactor
Reactor pressure vessel
Safety Assessment
Stress Corrosion Cracking
Vessel and Internals
Welds

BACKGROUND The NRC issued Generic Letter 92-01, Revision 1, Supplement 1, in May 1995 because previous responses to Generic Letter 92-01 may not have considered all relevant data in evaluating reactor pressure vessel integrity. The BWRVIP prepared a response to Supplement 1 for all U. S. BWR/2-6 plants that was documented in BWRVIP-08NP in November 1995. At the time the BWRVIP-08NP report was prepared, all relevant data for BWR vessel beltline welds had
not yet been compiled or evaluated. The BWRVIP-08NP report indicated that the BWRVIP would continue to work with other industry groups in longer term efforts to address the reactor vessel integrity issues identified in the Generic Letter.

OBJECTIVE To provide an updated generic BWR response to NRC Generic Letter 92-01, Revision 1, Supplement 1.

APPROACH A bounding approach for weld chemistries was used in BWRVIP-08NP to evaluate the effects of weld chemistry variability. However, not all the weld data for the BWR vessel beltline welds was available at the time and a complete evaluation could not be performed in the previous study. Since then, additional data has been retrieved by the Combustion Engineering (CE) Owners Group for CE fabricated welds, by the BWRVIP and General Electric for CB&I fabricated welds in BWR vessels, and by Framatome Technologies for welds fabricated by Babcock & Wilcox. The current study examines these new data and considers the impact on the results in BWRVIP-08NP for each U.S. BWR vessel if bounding chemistries are assumed for vessel beltline welds.

RESULTS The focus of the current and previous studies was to determine the effects of weld chemistry variability on upper shelf energy (USE) evaluations and pressure-temperature (P-T) curves. Based on the use of bounding chemistries in a previous equivalent margin analysis, it was concluded that the issue of weld chemistry variability has no impact on BWR USE evaluations. In this study, P-T curves were evaluated by examining weld heats in the irradiated beltline region for each vessel. Chemistries of the beltline weld heats were determined and bounding values were assumed for each weld heat to project embrittlement levels in the

welds. These values were compared to the adjusted reference temperatures of the limiting beltline materials, and these updated results were compared to previously calculated values in the BWRVIP-08NP report. The results demonstrate that there is no impact on the P-T curves due to chemistry variability for the BWR vessels.

EPRI PERSPECTIVE Searches of vessel fabrication and test records revealed additional data for the BWR vessel beltline welds. All existing weld data has now been retrieved for the BWR vessel beltline weld materials, and the data has been reviewed for assessment of BWR vessel integrity concerns. The additional chemistry data from these searches were evaluated to determine the impact on the BWR vessel integrity issues and to update the previous evaluation. These new data do not affect the previous conclusion that the equivalent margin analysis for all US BWR/2-6 vessels is bounding for USE in beltline welds, and that the weld copper content has no impact on the USE adequacy in BWRs. This update of the bounding approach for BWR vessel beltline materials considers the effects of including the new weld chemistry data on the calculation of BWR P-T limit curves. The results of this study demonstrate that the current BWR P-T curves meet or exceed the conservatism inherent in the applicable requirements.

PROJECT
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BWR Vessel and Internals Project

Update of Bounding Assessment of BWR/2-6 Reactor Pressure Vessel Integrity Issues (BWRVIP-46)

TR-109727
Research Project B301-34

Final Report, December 1997

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EXECUTIVE SUMMARY

Generic Letter 92-01, Revision 1, Supplement 1 requests US light water reactor owners to review all available data relevant to reactor vessel integrity issues, establish best estimate chemistries and assess the impact to vessel integrity issues of interest, including pressurized thermal shock (PTS), upper shelf energy (USE), low temperature overpressurization (LTOP), and pressure-temperature (P-T) limits. A previous study (BWRVIP-08NP) evaluated all available data prior to November 1995, examined the concern of weld chemistry variability, and considered the impact of assuming bounding chemistries.

Due to the nature of BWR operating characteristics, the PTS and LTOP issues do not apply, so the focus of the current and previous studies was to determine the effects of weld chemistry variability on USE evaluations and P-T curves. USE has already been evaluated for welds and plates of all US BWR/2-6 plants by the BWR Owners' Group in 1993, by means of an equivalent margin analysis (EMA). Based on the use of bounding chemistries in the equivalent margin analysis, it was concluded that the issue of weld chemistry variability has no impact on BWR USE evaluations.

In evaluating P-T curves, the industry could not retrieve and evaluate all relevant data by the November 1995 date. Since then, additional data has been retrieved by the CE Owners Group on CE fabricated welds, and by ATI Consulting and General Electric on CB&I fabricated welds in BWR vessels, and by Framatome Technologies for welds fabricated by Babcock & Wilcox. The current study examines these new data and considers the impact on current P-T limits for each U.S. BWR vessel if bounding chemistries are assumed for vessel beltline welds.

Weld heats in the irradiated beltline region were examined for each vessel. Chemistries of the beltline weld heats were determined and bounding values were assumed for each weld heat to project embrittlement levels in the welds. These values were compared to the adjusted reference temperatures of the limiting beltline materials, and these updated results were compared to previously calculated values in the BWRVIP-08NP report. The results demonstrate that there is no impact on the current P-T curves due to chemistry variability for the BWR vessels.

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1

INTRODUCTION

Uncertainties in vessel weld chemistry, primarily copper (Cu), prompted the NRC to issue Supplement 1 to Generic Letter 92-01, Revision 1 (92-01). Supplement 1 requests US light water reactor owners to review all available data relevant to reactor vessel integrity issues. The integrity issues of interest are pressurized thermal shock (PTS), upper shelf energy (USE), low temperature overpressurization (LTOP), and pressure-temperature (P-T) limits. PTS and LTOP are not vessel integrity issues for the BWRs, due to the characteristics of operating with a steam region in the vessel, and are not addressed here.

The BWRVIP prepared a response to Supplement 1 for all US BWR/2-6 plants in 1995. The results of this evaluation were documented in BWRVIP-08NP [1]. This prior evaluation used a bounding approach with known material chemistry information from the NRC's Reactor Vessel Integrity Database (RVID) and from other industry sources available to the BWRVIP. At the time the BWRVIP-08NP report was prepared, all relevant data for BWR vessel beltline welds had not yet been compiled and evaluated.

New data for BWR vessel beltline welds has become available within the past year. In particular, the Combustion Engineering Owners Group Weld Database was released in June 1997. This database contains all known data for CE fabricated welds in PWR and BWR vessels [2]. Furthermore, a recent thorough search of the Chicago Bridge & Iron quality assurance records by ATI Consulting and GE yielded additional weld data for the BWR vessels fabricated by CB&I. Framatome Technologies has also performed additional analyses of electro-slag welds (ESW) in BWR vessels [3,4], and Linde 80 welds [8]. The additional chemistry data from these searches were evaluated to determine the impact on the BWR vessel integrity issues and to update the previous evaluation. This report is a supplement to the BWRVIP-08NP report [1]; it is structured similarly to present the new data and to permit ease of review of updated bounding adjusted reference temperatures (ART). This update of the bounding approach for BWR vessel beltline materials considers the effects of including the new weld chemistry data on the calculation of BWR P-T limit curves. These new data do not affect the previous conclusion that the equivalent margin analysis for all US BWR/2-6 vessels is bounding for USE in beltline welds, and that the weld Cu content has no impact on the USE adequacy in BWRs.

1.1 Generic Letter 92-01 Supplement 1

NRC Generic Letter 92-01, Supplement 1 was issued May 19, 1995, and contained the following requests:

1. Provide a description of actions taken or planned to locate all data relevant to the determination of RPV integrity.
2. Provide an assessment of any change in best-estimate chemistry based on considerations of all relevant data.
3. Provide a determination of the need for the use of the ratio procedure in accordance with the established Position 2.1 of Regulatory Guide 1.99, Revision 2 for those licensees that use surveillance data to provide a basis for RPV integrity evaluation.
4. Provide a written report providing the results of any necessary revisions to the evaluation of RPV integrity.

The BWRVIP response to item 1 was submitted August 10, 1995 [5]. That response included commitments to review available industry databases and to pursue the availability of additional vessel data with vessel fabricators. At that time, not all data had been gathered and reviewed for assessment of BWR vessel integrity concerns. The current report is part of the longer term effort that was undertaken by the industry and the BWRVIP to locate all relevant materials data from the vessel fabricators, and establish consistent methods for handling the data in evaluating BWR vessel integrity.

1.2 Technical Approach for BWR Issues

As discussed in [5], the BWRVIP response to Supplement 1, items 2-4, addressed three issues:

- Impact of chemistry variability on USE equivalent margin analyses (EMA)
- Impact of weld chemistry variability on P-T curves
- Use of Position 2.1 ratio procedure in RG 1.99, Revision 2 [6].

The impact of chemistry variability on USE equivalent margin analyses and the use of Position 2.1 ratio procedure in RG 1.99, Revision 2 are described in the BWRVIP-08NP report [1]. These analyses are still bounding and are unaffected by the new data which has been identified.

The impact of weld chemistry variability on P-T curves is described below.

1.3 Applicability of this Supplemental Report

This report is a supplement to the BWRVIP-08NP report [1]; it is structured similarly to present the new data and to permit ease of review of updated bounding adjusted reference temperatures (ART). Limiting beltline weld chemistries are given in Table 2-1, and those limiting chemistries are contrasted with limiting beltline materials in Table 2-2. For welds with Cu chemistry variability >0.05%, the recalculated results for ART are given in Appendix A. For those heats with new chemistry data, the individual data measurements are given in Appendix B.

The revised summary of vessel weld chemistry ranges is given in Table 2-1. This table replaces Table 3-1 in the BWRVIP-08NP report [1]. The updated results showing impact of weld chemistry variability on P-T curves are shown in Table 2-2. This table replaces Table 3-2 in the BWRVIP-08NP report [1]. The details of the plant assessments of weld variability impact are given in Appendix A. This Appendix documents the basis for updated bounding ART values using the new data, and it replaces the previous Appendix B in the BWRVIP-08NP report [1]. The updated values based on the evaluation of new data are shown as the shaded entries in Table 2-1 and 2-2, and in the tables in Appendix A.

If the range of observed copper is less than 0.05 wt%, then no further analysis was performed since it is within the variability limit used as the basis for Reg. Guide 1.99, Revision 2. If the copper range exceeds 0.05 wt%, then combination of the worst case copper and the worst case nickel was used to calculate the bounding ART value for the particular welds, and these are values compared to the current limiting values used to calculate P-T curves.

2

IMPACT ON PRESSURE-TEMPERATURE CURVES

A bounding approach for weld chemistries was used in the BWRVIP-08NP report [1] to evaluate the effects of weld chemistry variability on P-T curves. However, not all the weld data for the BWR vessel beltline welds was available at the time, and a complete evaluation could not be performed in the previous study. These weld data have now been obtained from the vessel fabricators through extensive data searches, and the impact of the additional weld chemistry data is examined here.

In cases where the beltline welds are the limiting material for the P-T curves, weld chemistry variability could impact the period of validity of the curves in the long term, or impact the margin in the current curves in the near term. Fortunately, many plants do not have P-T curves based on a limiting beltline weld with significant chemistry variation, because the beltline welds are not subject to chemistry variability or the beltline welds are not the limiting material for the P-T curves. In plants where beltline welds with chemistry variability are limiting, the bounding copper and nickel chemistry is assumed and the increase in adjusted reference temperature (ART) is calculated for the current P-T curve period of validity, or for the projected 32 EFPY fluence level.

For those few plants where P-T curves are impacted in the near term, the magnitude of that impact is estimated in terms of change in the ART. The corresponding impact on the ASME Code Appendix G margin is discussed in Reference 1.

2.1 Plants with Potential Impact on P-T Curves

Variations in weld chemistry could mean that the limiting beltline ART, and thus the P-T curves, could be made more conservative by considering the weld chemistry variability. The process of evaluating P-T curve impact is shown in Figure 2-1. For each plant identified in Table 2-1 as having potentially significant weld chemistry variability (> .05 wt%), a worst case bounding chemistry analysis was performed to calculate the highest weld ART values. Shaded entries represent new or updated values. The results are summarized in Appendix A for vessels with weld chemistry variability. Chemistry ranges have been updated using new data. For each plant evaluated in this report, the limiting beltline material (plate or weld) is identified, and its associated ART is calculated for the noted EFPY. Those welds with chemistry variability are evaluated assuming the highest independent values of Cu and Ni to determine bounding ART values. Worst case bounding ARTs are then compared with

the limiting beltline ART to determine the impact the bounding chemistry would have, if any, in terms of upward shift (in °F) of the current beltline P-T curve.

It is noted that Combustion Engineering (CE) and Babcock & Wilcox (B&W) used copper coated weld wire for the BWR submerged arc weld (SAW) processes. Chicago Bridge & Iron (CB&I), and later CBI Nuclear (CBIN), produced low copper SAW and shielded metal arc (SMAW) seam welds with relatively little chemistry variability. Additional weld chemistry data discovered at CB&I confirmed that the copper variability in these weld heats is low (< 0.05%) as shown in Appendix B. Measured copper and nickel chemistry data for the limiting BWR vessel beltline welds are shown in Appendix B as obtained from CB&I data searches. Data was also obtained from the CEOG database [2], and the relevant CE weld chemistry data are also included in Appendix B for the limiting BWR vessel welds with high Cu variability.

The results given in Appendix A (for vessels with copper variability > 0.05%) show the individual plant assessments of weld variability impact. Additional data from the CEOG database shows an increase in the upper bound Cu and Ni contents for some limiting welds such as heat no. W5214 and heat nos. 13253/12008. Other welds, such as heat nos. 27204/12008, show a decrease in the upper bound Cu and Ni from the conservative values that were previously assumed. This contributes to the reduction in impact on the P-T curves for the Cooper, Pilgrim, and FitzPatrick vessels. The recalculated weld ART values summarized in Table 2-2 shows the impact of assuming bounding chemistries on the current beltline P-T curve for each vessel. Shaded entries indicate updated values from the BWRVIP-08NP report [1].

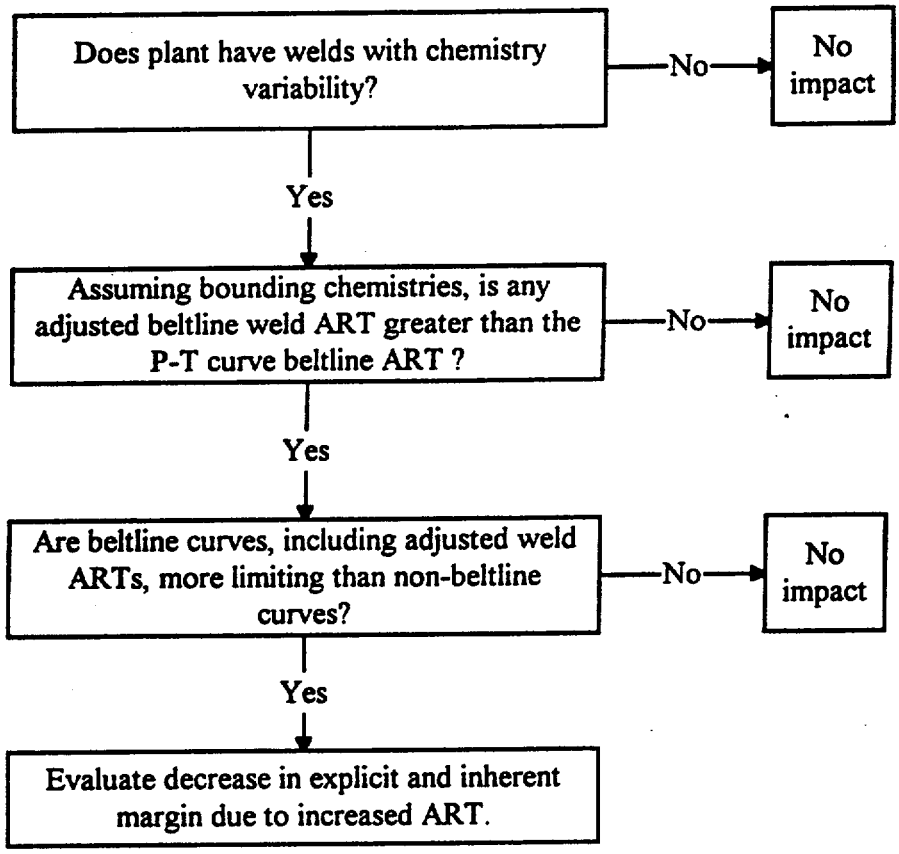

Figure 2-1. Evaluation Process for P-T Curve Impact

Table 2-1 : Vessel Weld Chemistry Ranges

Plant	Beltline Weld Fabricator	Limiting Weld Heat Number	Limiting Weld Cu Range,%	Limiting Weld Ni Range,%	Cu Variability >.05% ?
Browns Ferry 1	B&W	406L44	.22-.32	.58-.59	Yes
Browns Ferry 2	B&W	ESW (bound)	.10-.30	.18-.41	Yes
Browns Ferry 3	B&W	ESW (bound)	.10-.30	.18-.41	Yes
Brunswick 1*	CB&I	1P4218	.05-.06	.83-.89	No
Brunswick 2*	CB&I	S3986	.05-.06	.81-1.07	No
Clinton	CBIN	76492	.10	1.08	No
Cooper*	CE	27204/12008	.19-.27	.97-.99	Yes
Dresden 2	B&W	ESW (bound)	.10-.30	.18-.41	Yes
Dresden 3	B&W	299L44	.29-.35	.59-.68	Yes
Duane Arnold*	CB&I	432Z0471	.03	.91	No
Fermi 2	CE	13253/12008	.11-.31	.72-.92	Yes
FitzPatrick	CE	27204/12008	.19-.27	.97-.99	Yes
Grand Gulf 1	CBIN	627260	.06	1.08	No
Hatch 1	CE	90099	.09-.30	.06	Yes
Hatch 2	CE	10137	.10-.28	.02-.15	Yes
Hope Creek 1	IHI	D53040	.05-.10	.41-.68	No
LaSalle 1	CE	1P3571	.17-.53	.51-.88	Yes
LaSalle 2*	CBIN	3P4966	.02-.03	.80-1.07	No
Limerick 1*	CB&I	640892	.09	1.00	No
Limerick 2*	CB&I	640892	.09	1.00	No
Millstone 1*	CE	W5214	.12-.37	.63-1.29	Yes
Monticello*	CB&I	Unknown	0.1	.99	No
Nine Mile Point 1*	CE	1248;86054B**	.19-.24	.04-.06	No
Nine Mile Point 2*	CBIN	4P7216	.03-.06	.75-.94	No
Oyster Creek*	CE	86054B	.19-.24	.04-.05	No
Peach Bottom 2*	B&W	ESW (bound)	.21	.21	No
Peach Bottom 2*	CB&I	S3986	.05-.06	.81-1.07	No
Peach Bottom 3*	B&W	ESW (bound)	.21	.21	No
Peach Bottom 3*	CB&I	1P4217	.09-.11	.91-.96	No
Perry	CBIN	627260	.06	1.08	No
Pilgrim	CE	27204/12008	.19-.27	.97-.99	Yes
Quad Cities 1	B&W	ESW (bound)	.10-.30	.18-.41	Yes
Quad Cities 2	B&W	ESW (bound)	.10-.30	.18-.41	Yes
River Bend	CBIN	5P6756	.08-.09	.92-.96	No
Susquehanna 1*	CB&I	494K2351	.04	1.10	No
Susquehanna 2*	CB&I	624263	.06	.89	No
Vermont Yankee*	CB&I	Unknown	.04	1.00	No
WNP-2 (Hanford)*	CBIN	05P018	.09	.90	No

*Vessels limited by base metal for P-T curves

**Equivalent beltline weld heats

Table 2-2: Impact of Weld Chemistry Variability on P-T Curves

Plant	Assumed EFPY for P-T Curve	Limiting Beltline Material Heat for P-T Curve	P-T Curve Max. ART (°F)	Limiting Beltline Weld Heat	ART for Bounding Weld Chemistry (°F)	Impact on P/T Curves (°F)
Browns Ferry 1	12	406L44	92.3	406L44	88.6	--
Browns Ferry 2	12	ESW (estimate)	80.7	ESW (bound)	59.8	--
Browns Ferry 3	12	ESW (estimate)	80.7	ESW (bound)	80.9	--
Cooper	18	Plate C2274-1	102	27204/12008	105.4	3
Dresden 2*	22	299L44	97.4	ESW (bound)	86.1	--
Dresden 3*	22	299L44	97.4	299L44	97.4	--
Fermi 2	32	13253/12008	70.1	13253/12008	70.0	--
FitzPatrick	16	27204/12008	116	27204/12008	89.2	--
Hatch 1	32	90099	163.9	90099	123.2	--
Hatch 2	32	10137	71.9	10137	63.2	--
LaSalle 1	16	1P3571	53	1P3571	49.9	--
Millstone 1	32	Plate C1079-1	137	W5214	111.0	--
Pilgrim	32	27204/12008	116	27204/12008	105.6	--
Quad Cities 1*	22	299L44	97.4	ESW (bound)	85.6	--
Quad Cities 2*	22	299L44	97.4	ESW (bound)	95.9	--

* All Dresden and Quad Cities units use the same P-T curves based on limiting weld material and fluence from Dresden 3

3

CONCLUSIONS

Searches of vessel fabrication and test records revealed additional data for the BWR vessel beltline welds. All existing weld data have now been retrieved for the BWR vessel beltline weld materials, and the data has been reviewed for assessment of BWR vessel integrity concerns. The impact of weld chemistry variability on P-T curves was evaluated by a screening process which involved several steps (Figure 2-1). In the first step, 21 plants were identified as having copper variability $\leq$ .05%, and therefore are not affected by chemistry variability. For the remaining 15 plants, the previous values for weld chemistry ranges and ART values were updated, and 14 of the vessels were found to have P-T curves that would be conservative even assuming the most bounding beltline weld chemistry data. The one remaining plant showed a decrease in the potential impact on P-T curves to only 3°F by using the new data. This small potential impact has no significant effect on the explicit safety factor in the Appendix G P-T curves for this plant. This study demonstrates that the current BWR P-T curves meet or exceed the conservatism inherent in the Appendix G margin requirements.

4

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4. Campbell, C. A. "Evaluation of RT_{NDT}, USE, and Chemical Composition of Core Region Electroslag Welds for Quad Cities Units 1 and 2," Framatome Technologies, BAW-2259, January 1996.

5. Letter, dated August 10, 1995, J. T. Beckham, Jr., BWRVIP Chairman to USNRC, "BWRVIP Response to NRC Generic Letter 92-01, Revision 1, Supplement 1, Reactor Vessel Structural Integrity."

6. Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," U.S. Nuclear Regulatory Commission, May 1988.

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A

PLANT ASSESSMENTS OF WELD VARIABILITY IMPACT

In order to determine the impact of weld chemistry variability on P-T curves, an evaluation of ART changes was made for plants where a beltline weld is limiting or could potentially become limiting due to chemistry variability. The process for evaluation followed these steps:

1. Identify those plants where weld chemistry variability (Cu > 0.05%) is observed to occur based on the existing weld data, including any new weld chemistry data.

   (Plants identified in Step 1 were further evaluated for ART changes)

2. For the affected plants, determine weld heat numbers and examine the range of copper and nickel chemistry data for the specific vessel beltline welds.

3. Perform the ART evaluation for the EFPY to which the P-T curves are valid, or for the assumed end-of-life (32 EFPY) fluence.

4. Identify the limiting material, and its ART, for the corresponding EFPY above.

5. Recalculate ART for welds with high ART and for welds subject to chemistry variability:
   - Select maximum Cu
   - Select maximum Ni (these are not necessarily paired chemistry values)
   - Use the latest calculated fluence based on revised fluence
   - Use improved initial RT_{NDT} values where appropriate
   - Include surveillance adjustment (Position 2.1) and reduced margin term where appropriate

6. Compare recalculated weld ART with limiting P-T curve ART to determine bounding impact due to chemistry variability.

It should be noted that not every weld is evaluated; if it was clear based on an initial screening of fluence, chemistry, initial RT_{NDT}, etc. that a weld's ART would be considerably lower than the limiting ART, it was not evaluated.

It should also be noted that in the process of identifying bounding chemistries from all the available data, in some cases a lower bounding copper content was used when actual data could reduce the previously-assumed bounding chemistry value.

Weld Variability Impact

Plant: **Browns Ferry 1**

EFPY: **12**

Limiting Beltline Material:

Material Type	Heat Number	Initial RT_{NDT} (°F)	% Cu	% Ni	RG 1.99, Rev. 2 CF (°F)	P-T Curve ART (°F)
SAW	406L44	20	0.31	0.59	196.7	92.3

Beltline Weld Materials:

Material Type	Heat Number	Initial RT_{NDT} (°F)	% Cu	% Ni	RG 1.99, Rev. 2 CF (°F)	P-T Curve ART (°F)	Bounding % Cu	Bounding % Ni	Bounding CF (°F)	Bounding ART (°F)	Impact on P-T Curves (°F)
ESW	Various	10	0.28	0.35	154.5	69.7	0.30	0.41	168.4	67.7	--
SAW	406L44	20	0.31	0.59	196.7	92.3	0.32	0.59	200.7	88.6	--

Notes: Curve ART values include $\sigma_i = 10^\circ$ F in the Margin term which is not included in the bounding ART value

Weld Variability Impact

Plant: **Browns Ferry 2**

EFPY: **12**

Limiting Beltline Material:

Material Type	Heat Number	Initial RT_{NDT} (°F)	% Cu	% Ni	RG 1.99, Rev. 2 CF (°F)	P-T Curve ART (°F)
ESW	Various	10	0.28	0.35	154.5	80.7

Beltline Weld Materials:

Material Type	Heat Number	Initial RT_{NDT} (°F)	% Cu	% Ni	RG 1.99, Rev. 2 CF (°F)	P-T Curve ART (°F)	Bounding % Cu	Bounding % Ni	Bounding CF (°F)	Bounding ART (°F)	Impact on P-T Curves (°F)
ESW	Various	10	0.28	0.35	154.5	80.7	0.30	0.41	168.4	59.8	--

Notes: Curve ART values include $\sigma_i = 10^\circ$ F in margin term, not included in bounding ART value. Curve ART based on 32 EFPY ¼T fluence = 7.4×10^{17}. Bounding ART based on latest fluence calculation of 4.2×10^{17}.

Weld Variability Impact

Plant: **Browns Ferry 3**

EFPY: **12**

Limiting Beltline Material:

Material Type	Heat Number	Initial RT_{NDT} (°F)	% Cu	% Ni	RG 1.99, Rev. 2 CF (°F)	P-T Curve ART (°F)
ESW	Various	10	0.28	0.35	154.5	80.7

Beltline Weld Materials:

Material Type	Heat Number	Initial RT_{NDT} (°F)	% Cu	% Ni	RG 1.99, Rev. 2 CF (°F)	P-T Curve ART (°F)	Bounding % Cu	Bounding % Ni	Bounding CF (°F)	Bounding ART (°F)	Impact on P-T Curves (°F)
ESW	Various	10	0.28	0.35	154.5	80.7	0.30	0.41	168.4	80.9	--

Note: Curve ART values include σ_i = 10°F in the margin term, which is not included in the bounding ART value.

Weld Variability Impact

Plant: Cooper

EFPY: 18

Limiting Beltline Material:

Material Type	Heat Number	Initial RT_{NDT} (°F)	% Cu	% Ni	RG 1.99, Rev. 2 CF (°F)	P-T Curve ART (°F)
Plate	C2274-1	14	0.2	0.68	153	102

Beltline Weld Materials:

Material Type	Heat Number	Initial RT_{NDT} (°F)	% Cu	% Ni	RG 1.99, Rev. 2 CF (°F)	P-T Curve ART (°F)	Bounding % Cu	Bounding % Ni	Bounding CF (°F)	Bounding ART (°F)	Impact on P-T Curves (°F)
SAW	12420	-50	0.35	1.0	272	99.3	0.22	1.04	372.1	77.2	--
SAW	27204/ 12008	-50	0.19	0.97	215.7	80.0	0.27	0.99	388.4	105.4	3
SAW	21935	-50	0.20	0.71	178.7	67.3	0.35	0.9	402.7	85.4	--

Notes: Bounding ARTs consider the latest fluences, which are location dependent. Welds 12420 and 21935 have 32 EFPY 1/4T fluences of 7.5×10^{17} n/cm², while 27204/12008 has 1.1×10^{18} n/cm². Weld CFs multiplied by Pos. 2.1 ratio surveillance adjustment (SA = 1.57) for bounding analysis of welds. Since the surveillance adjustment is used, the Margin term is 28°F (1σ). Evaluation shows no significant decrease in safety factor of P-T curves for 3°F increase in ART for limiting beltline material.

Weld Variability Impact

Plant: **Dresden 2**

EFPY: **22**

Limiting Beltline Material:

Material Type	Heat Number	Initial RT_{NDT} (°F)	% Cu	% Ni	RG 1.99, Rev. 2 CF (°F)	P-T Curve ART (°F)
SAW	299L44	-5	0.35	0.68	224	97.4

Beltline Weld Materials:

Material Type	Heat Number	Initial RT_{NDT} (°F)	% Cu	% Ni	RG 1.99, Rev. 2 CF (°F)	P-T Curve ART (°F)	Bounding % Cu	Bounding % Ni	Bounding CF (°F)	Bounding ART (°F)	Impact on P-T Curves (°F)
ESW	Various	23	0.24	0.36	139	78.7	0.30	0.41	168.4	86.1	--
SAW	71249	10	0.26	0.61	182	66.8	0.26	0.62	183.2	67.1	--

Note: 97.4°F curve ART developed for composite curve applicable to all Dresden and Quad Cities plants, with the Dresden 3 fluence and limiting materials used. Plant-specific bounding ART for welds uses 32 EFPY 1/4T fluence = 2.5×10^{17} n/cm² [7]. The initial RT_{NDT} for ESW = 23°F with σ_i = 13°F based on new data from BAW-2258 [3]. Revised ART for SAW includes σ_i = 20°F [7].

Weld Variability Impact

Plant: **Dresden 3**

EFPY: **22**

Limiting Beltline Material:

Material Type	Heat Number	Initial RT_{NDT} (°F)	% Cu	% Ni	RG 1.99, Rev. 2 CF (°F)	P-T Curve ART (°F)
SAW	299L44	-5	0.35	0.68	224	97.4

Beltline Weld Materials:

Material Type	Heat Number	Initial RT_{NDT} (°F)	% Cu	% Ni	RG 1.99, Rev. 2 CF (°F)	P-T Curve ART (°F)	Bounding % Cu	Bounding % Ni	Bounding CF (°F)	Bounding ART (°F)	Impact on P-T Curves (°F)
ESW	Various	23	0.24	0.36	139	87.4	0.30	0.41	168.4	96.9	--
SAW	299L44	-5	0.35	0.68	224	97.4	0.35	0.68	224	97.4	--

Note: The 32 EFPY 1/4T fluence = 3.5×10^{17} n/cm² [7]. The initial RT_{NDT} for ESW = 23°F with σ_i = 13°F based on new data from BAW-2258 [3]. Revised ART for SAW includes σ_i = 20°F [7].

Weld Variability Impact

Plant: **Fermi 2**

EFPY: **32 (10% power uprate included)**

Limiting Beltline Material:

Material Type	Heat Number	Initial RT_{NDT} (°F)	% Cu	% Ni	RG 1.99, Rev. 2 CF (°F)	P-T Curve ART (°F)
SAW	13253/12008	-44	0.26	0.87	224	70.1

Beltline Weld Materials:

Material Type	Heat Number	Initial RT_{NDT} (°F)	% Cu	% Ni	RG 1.99, Rev. 2 CF (°F)	P-T Curve ART (°F)	Bounding % Cu	Bounding % Ni	Bounding CF (°F)	Bounding ART (°F)	Impact on P-T Curves (°F)
SAW	13253/ 12008	-50	0.26	0.87	224	70.1*	0.31	0.92	247.2	70.0	--
SAW	33A277	-50	0.32	0.50	188.5	47.8	0.32	0.23	158	31.8	--
SAW	10137	-50	0.23	1.0	236	67.2	0.27	0.18	132.5	18.7	--

* P-T Curve ART does not reflect lower initial RT_{NDT} value. Note: Curve ART for 13253/12008 was based on analytical estimates of Cu and Ni. 32 EFPY 1/4T fluence = 4.0×10^{17} n/cm^2. The initial RT_{NDT} for weld heat 13253/12008 = -50°F (from GE estimation method) using 3 Charpy test data points exceeding 50 ft-lbs at a test temperature of +10°F.

Weld Variability Impact

Plant: **FitzPatrick**

EFPY: **16**

Limiting Beltline Material:

Material Type	Heat Number	Initial RT_{NDT} (°F)	% Cu	% Ni	RG 1.99, Rev. 2 CF (°F)	P-T Curve ART (°F)
SAW	27204/12008	-22	0.25	0.99	241	116

Beltline Weld Materials:

Material Type	Heat Number	Initial RT_{NDT} (°F)	% Cu	% Ni	RG 1.99, Rev. 2 CF (°F)	P-T Curve ART (°F)	Bounding % Cu	Bounding % Ni	Bounding CF (°F)	Bounding ART (°F)	Impact on P-T Curves (°F)
SAW	27204/ 12008	-50	0.25	0.99	241	116*	0.27	0.99	247.4	89.2	--
SAW	305414	-50	0.33	0.59	204	84.1	0.35	0.61	213.5	87.7	--
SAW	13253/ 12008	-50	0.26	0.87	224	91.8	0.31	0.92	247.2	89.1	--

* P-T Curve ART does not reflect lower initial RT_{NDT} value. Note: Curve ART for 13253/12008 was based on analytical estimates of Cu and Ni. The 16 EFPY 1/4T fluence = 6.5×10^{17} n/cm². The initial RT_{NDT} for weld heat 27204/12008 = -50°F (from GE estimation method) using 3 Charpy test data points exceeding 50 ft-lbs at a test temperature of +10°F.

Weld Variability Impact

Plant: **Hatch 1**

EFPY: **32 (10% power uprate included)**

Limiting Beltline Material:

Material Type	Heat Number	Initial RT_{NDT} (°F)	% Cu	% Ni	RG 1.99, Rev. 2 CF (°F)	P-T Curve ART (°F)
SAW	90099	-10	0.17	1.0	207	163.9

Beltline Weld Materials:

Material Type	Heat Number	Initial RT_{NDT} (°F)	% Cu	% Ni	RG 1.99, Rev. 2 CF (°F)	P-T Curve ART (°F)	Bounding % Cu	Bounding % Ni	Bounding CF (°F)	Bounding ART (°F)	Impact on P-T Curves (°F)
SAW	90099	-10	0.17	1.0	207	163.9	0.30	0.06	135.5	123.2	--
SAW	33A277	-50	0.23	1.0	236	140.4	0.32	0.23	158	95.9	--
SAW	IP2809	-50	0.28	0.76	211.8	126.6	0.24	0.77	201.1	120.5	--
SAW	13253	-50	0.27	0.74	206.4	120.6	0.27	0.74	206.4	120.6	--

Notes: Curve Ni values at 1.0% were default values. Previous Cu value for IP2809 was based on surveillance weld data. Information now indicates that surveillance weld is 1P3571. Heat 90099 bare wire data shows 0.06% Ni.

Weld Variability Impact

Plant: **Hatch 2**

EFPY: **32 (10% power uprate included)**

Limiting Beltline Material:

Material Type	Heat Number	Initial RT_{NDT} (°F)	% Cu	% Ni	RG 1.99, Rev. 2 CF (°F)	P-T Curve ART (°F)
SAW	10137	-50	0.23	0.50	154.5	71.9

Beltline Weld Materials:

Material Type	Heat Number	Initial RT_{NDT} (°F)	% Cu	% Ni	RG 1.99, Rev. 2 CF (°F)	P-T Curve ART (°F)	Bounding % Cu	Bounding % Ni	Bounding CF (°F)	Bounding ART (°F)	Impact on P-T Curves (°F)
SAW	10137	-50	0.23	0.50	154.5	71.9	0.28	0.15	134.0	63.2	--

Note: Curve value of Ni was a conservative estimate, given that Ni was not a required element for this particular weld.

Weld Variability Impact

Plant: **LaSalle 1**

EFPY: **16**

Limiting Beltline Material:

Material Type	Heat Number	Initial RT_{NDT} (°F)	% Cu	% Ni	RG 1.99, Rev. 2 CF (°F)	P-T Curve ART (°F)
SAW	1P3571	-30	0.37	0.75	241	53

Beltline Weld Materials:

Material Type	Heat Number	Initial RT_{NDT} (°F)	% Cu	% Ni	RG 1.99, Rev. 2 CF (°F)	P-T Curve ART (°F)	Bounding % Cu	Bounding % Ni	Bounding CF (°F)	Bounding ART (°F)	Impact on P-T Curves (°F)
SAW	1P3571	-30	0.37	0.75	241	53	0.53	0.88	272	49.9	--

Notes: Bounding ART based on fluence from surveillance capsule results (32 EFPY 1/4T fluence = 3.1×10^{17} n/cm^2). Fluence used for P-T curve ART was higher (32 EFPY 1/4T fluence = 4.0×10^{17} n/cm^2). Cu exceeds RG 1.99, Rev. 2 Table values. Default chemistry (0.35% Cu, 1.0% Ni, CF = 272) is more limiting than the highest CF Table value at 0.88% Ni (0.40 Cu, 0.88 Ni, CF = 269.4). Default chemistry was used.

Weld Variability Impact

Plant: **Millstone 1**

EFPY: **32**

Limiting Beltline Material:

Material Type	Heat Number	Initial RT_{NDT} (°F)	% Cu	% Ni	RG 1.99, Rev. 2 CF (°F)	P-T Curve ART (°F)
Plate	C1079-1	26	0.19	0.51	219.3	137

Beltline Weld Materials:

Material Type	Heat Number	Initial RT_{NDT} (°F)	% Cu	% Ni	RG 1.99, Rev. 2 CF (°F)	P-T Curve ART (°F)	Bounding % Cu	Bounding % Ni	Bounding CF (°F)	Bounding ART (°F)	Impact on P-T Curves (°F)
SAW	W5214	-20	0.26	1.2	276	126.5	0.37	1.29	236	111.0	--
SAW	34B009	-50	0.18	1.03	216.4	70.9	0.23	1.03	182.4	57.5	--

Note: The plate CF includes a Position 2.1 ratio surveillance adjustment factor of 1.66 (CF = 132.1 * 1.66). The 32 EFPY fluence = 1.1×10^{18} n/cm^2. Bounding ART includes ratio surveillance adjustment of 0.76 applied to beltline welds based on credible surveillance weld data (CF = 311*0.76 for weld heat W5214, and CF = 240*0.76 for weld heat 34B009).

Weld Variability Impact

Plant: **Pilgrim**

EFPY: **32 (1/4 T f = 9.3 x 10^{17} at weld)**

Limiting Beltline Material:

Material Type	Heat Number	Initial RT_{NDT} (°F)	% Cu	% Ni	RG 1.99, Rev. 2 CF (°F)	P-T Curve ART (°F)
SAW	27204/12008	-50	0.35	1.0	272	116

Beltline Weld Materials:

Material Type	Heat Number	Initial RT_{NDT} (°F)	% Cu	% Ni	RG 1.99, Rev. 2 CF (°F)	P-T Curve ART (°F)	Bounding % Cu	Bounding % Ni	Bounding CF (°F)	Bounding ART (°F)	Impact on P-T Curves (°F)
SAW	27204/ 12008	-50	0.35	1.0	272	116	0.27	0.99	247.4	105.6	--
SAW	21935	-50	--	--	--	--	0.35	0.9	256.5	109.7	--
SAW	27204	-50	--	--	--	--	0.32	1.0	263	111.9	--
SAW	13253	-50	--	--	--	--	0.27	0.74	206.4	89.1	--

Notes: As-welded chemistries were not known, so the default values of 0.35% Cu and 1.0% Ni were used for all welds in preparing the P-T curves.

Weld Variability Impact

Plant: **Quad Cities 1**

EFPY: **22**

Limiting Beltline Material:

Material Type	Heat Number	Initial RT_{NDT} (°F)	% Cu	% Ni	RG 1.99, Rev. 2 CF (°F)	P-T Curve ART (°F)
SAW	299L44	-5	0.35	0.68	224	97.4

Beltline Weld Materials:

Material Type	Heat Number	Initial RT_{NDT} (°F)	% Cu	% Ni	RG 1.99, Rev. 2 CF (°F)	P-T Curve ART (°F)	Bounding % Cu	Bounding % Ni	Bounding CF (°F)	Bounding ART (°F)	Impact on P-T Curves (°F)
ESW	Various	23	0.24	0.36	139	78.0	0.30	0.41	168.4	85.6	--
SAW	72445	-5	0.21	0.59	162	66.9	0.22	0.60	167	68.2	--
SAW	406L44	-10	0.31	0.59	197	75.3	0.32	0.59	200.7	76.5	--

Note: 97.4°F curve ART developed for composite curve applicable to all Dresden and Quad Cities plants, with the Dresden 3 fluence and limiting materials bounding. Plant-specific bounding ART for welds uses 32 EFPY 1/4T fluence = 2.4×10^{17} n/cm^2 [7]. Initial RT_{NDT} for ESW = 23°F with σ_i = 13°F based on new data from BAW-2259 [4]. Revised ART for SAW includes σ_i = 20°F [7].

Weld Variability Impact

Plant: **Quad Cities 2**

EFPY: **22**

Limiting Beltline Material:

Material Type	Heat Number	Initial RT_{NDT} (°F)	% Cu	% Ni	RG 1.99, Rev. 2 CF (°F)	P-T Curve ART (°F)
SAW	299L44	-5	0.35	0.68	224	97.4

Beltline Weld Materials:

Material Type	Heat Number	Initial RT_{NDT} (°F)	% Cu	% Ni	RG 1.99, Rev. 2 CF (°F)	P-T Curve ART (°F)	Bounding % Cu	Bounding % Ni	Bounding CF (°F)	Bounding ART (°F)	Impact on P-T Curves (°F)
ESW	Various	23	0.24	0.36	139	86.3	0.30	0.41	168.4	95.9	--

Note: 97.4°F curve ART developed for composite curve applicable to all Dresden and Quad Cities plants, with the Dresden 3 fluence and limiting materials bounding. Plant-specific bounding ART for weld uses 32 EFPY 1/4T fluence = 2.3×10^{17} n/cm^2 [7]. Initial RT_{NDT} for ESW = 23°F with σ_i = 13°F based on new data from BAW-2259 [4]. Revised ART for SAW includes σ_i = 20°F [7].

B

SUMMARY OF PROPERTIES FOR WELD WIRE HEATS

Table B-1: Summary of Properties for Weld Wire Heat No. 1P4217

Heat No.	Flux Type	Flux Lot	%Cu	%Ni	Source
1P4217	Linde 124	3478	0.1	0.96	CB&I CTR*
1P4217	Linde 124	3478	0.1	0.91	CB&I CTR*
1P4217	Linde 124	3929	0.11	0.93	.CB&I CTR
1P4217	Linde 124	3929	0.11	0.96	CB&I, WQ
1P4217			0.09	0.95	CE CTR*

Table B-2: Summary of Properties for Weld Wire Heat No. 5P6756

Heat No.	Flux Type	Flux Lot	%Cu	%Ni	Source
5P6756			0.08	0.96	CB&I CTR
5P6756	Unknown		0.08	0.96	CB&I, WQ*
5P6756	Linde 124	342	0.08	0.93	CB&I,WQ
5P6756	Linde 124	342	0.09	0.92	CB&I,WQ

* Additional CB&I weld data from searches through 11/97.

Table B-3: Summary of Properties for Weld Wire Heat No. S3986

Heat No.	Flux Type	Flux Lot	%Cu	%Ni	Source
S3986			0.03	1.07	ADCOM CMTR*
S3986	Linde 124		0.055	0.96	Brunswick 1, SC
S3986	Linde 124		0.051	0.98	Brunswick 1, SC
S3986	Linde 124	934	0.05	0.96	CB&I CTR
S3986	Linde 124	934	0.05	0.92	CB&I CTR
S3986	Linde 124	934	0.06	0.9	CB&I CTR
S3986	Linde 124	934	0.06	0.81	CB&I CTR
S3986	Linde 124	934	0.06	0.97	CB&I CTR
S3986			0.09	0.97	CB&I CTR*
S3986			0.03	1.07	CB&I CTR*
S3986	Linde 124	934	0.05	0.96	CB&I PT#200
S3986	Linde 124	934	0.055	0.97	D C Cook 2, SC
S3986	Linde 124	934	0.051	0.93	Trojan, SC
S3986	Linde 124	934	0.06	0.97	Trojan, SC

Table B-4: Summary of Properties for Weld Wire Heat No. 4P7216

Heat No.	Flux Type	Flux Lot	%Cu	%Ni	Source
4P7216	Linde 124	751	0.06	0.85	CB&I,WQ
4P7216	Linde 124	751	0.04	0.83	CB&I,WQ
4P7216	Linde 124	156	0.03	0.75	CB&I,WQ
4P7216	Linde 124	156	0.03	0.81	CB&I,WQ
4P7216	Linde 124	156	0.03	0.79	CBIN CTR*
4P7216			0.05	0.94	QATR*

* Additional CB&I weld data from searches through 11/97.

Table B-5: Summary of Properties for Weld Wire Heat No. 3P4966

Heat No.	Flux Type	Flux Lot	%Cu	%Ni	Source
3P4966	Linde 124	342	0.03	0.93	CB&I CTR*
3P4966	Linde 124	342	0.023	0.9	CB&I CTR*
3P4966	Linde 124	331	0.023	1.07	CB&I CTR*
3P4966	Linde 124	331	0.025	0.94	CB&I CTR*
3P4966	Linde 124	1214	0.02	0.92	CB&I CTR*
3P4966	Linde 124	1214	0.02	0.8	CB&I CTR*
3P4966	Linde 124	1214	0.03	0.9	CB&I,WQ
3P4966	Linde 124	1214	0.03	0.88	CB&I,WQ
3P4966			0.03	0.96	RACO CTR*

Table B-6: Summary of Properties for Weld Wire Heat No. 27204/12008**

Heat No.	Flux Type	Flux Lot	%Cu	%Ni	Source
27204/12008	Linde 1092	3692		0.98	CE D-LOG
27204/12008	Linde 1092	3724	0.27	0.99	CE D-LOG
27204/12008	Linde 1092	3714	0.19	0.97	WCAP-10754

* Additional CB&I weld data from searches through 11/97.

** Data obtained from CEOG database [2]

Table B-7: Summary of Properties for Weld Wire Heat No. 86054B**

Heat No.	Flux Type	Flux Lot	%Cu	%Ni	Source
86054B	Linde 1092		0.23	0.05	CE D-LOG
86054B	ARCOS B5		0.24	0.05	CE D-LOG
86054B	ARCOS B5	3K2F		0.04	CE D-LOG
86054B	ARCOS B5	4E5F	0.19	0.05	CE D-LOG

Table B-8: Summary of Properties for Weld Wire Heat No. W5214**

Heat No.	Flux Type	Flux Lot	%Cu	%Ni	Source
W5214	Linde 1092	3576		0.99	CE D-LOG
W5214	Linde 1092	3617	0.33	0.63	WCAP-10304
W5214	Linde 1092	3617	0.34	0.66	WCAP-10304
W5214	Linde 1092	3617	0.35	0.69	WCAP-10304
W5214	Linde 1092	3617	0.26		RPVDATA
W5214	Linde 1092	3692	0.154	0.99	Westinghouse
W5214	Linde 1092	3692	0.163	0.9	Westinghouse
W5214	Linde 1092	3692	0.152	1.08	Westinghouse
W5214	Linde 1092	3692	0.166	1.00	Westinghouse
W5214	Linde 1092	3576		0.96	CE D-LOG
W5214	Linde 1092	3576		0.92	CE D-LOG
W5214	Linde 1092	3576		1.12	CE D-LOG

** Data obtained from CEOG database [2]

W5214	Linde 1092	3576		0.97	CE D-LOG
W5214	Linde 1092	3576		1.05	CE D-LOG
W5214	Linde 1092	3600	0.18		SWRI-02-5212
W5214	Linde 1092	3600	0.19		SWRI-02-5212
W5214	Linde 1092	3600	0.20		SWRI-02-5212
W5214	Linde 1092	3600	0.22		SWRI-02-5212
W5214	Linde 1092	3600	0.20	0.69	SWRI-17-2108
W5214	Linde 1092	3600	0.12	1.00	SWRI-17-2108
W5214	Linde 1092	3600	0.23	1.02	SWRI-17-2108
W5214	Linde 1092	3600	0.20	1.06	SWRI-17-2108
W5214	Linde 1092	3600		0.72	CE D-LOG
W5214	Linde 1092	3600		0.76	CE D-LOG
W5214	Linde 1092	3600		0.77	CE D-LOG
W5214	Linde 1092	3600		0.81	CE D-LOG
W5214	Linde 1092	3600		0.81	CE D-LOG
W5214	Linde 1092	3600		0.81	CE D-LOG
W5214	Linde 1092	3600		0.96	CE D-LOG
W5214	Linde 1092	3600		0.96	CE D-LOG
W5214	Linde 1092	3600		0.97	CE D-LOG
W5214	Linde 1092	3600		0.98	CE D-LOG
W5214	Linde 1092	3600		0.98	CE D-LOG
W5214	Linde 1092	3600		0.99	CE D-LOG

W5214	Linde 1092	3600		1.00	CE D-LOG
W5214	Linde 1092	3600		1.01	CE D-LOG
W5214	Linde 1092	3600		1.01	CE D-LOG
W5214	Linde 1092	3600		1.03	CE D-LOG
W5214	Linde 1092	3600		1.03	CE D-LOG
W5214	Linde 1092	3600		1.05	CE D-LOG
W5214	Linde 1092	3600		1.06	CE D-LOG
W5214	Linde 1092	3600		1.08	CE D-LOG
W5214	Linde 1092	3600	0.20		CE D-LOG
W5214	Linde 1092	3600			CE D-LOG
W5214	Linde 1092	3600		0.87	CE D-LOG
W5214	Linde 1092	3600		0.99	CE D-LOG
W5214	Linde 1092	3600		1.07	CE D-LOG
W5214	Linde 1092	3600	0.20	1.15	WCAP-7323
W5214	Linde 1092	3617	0.278	1.059	CPCo
W5214	Linde 1092	3617	0.281	1.066	CPCo
W5214	Linde 1092	3617	0.292	1.127	CPCo
W5214	Linde 1092	3617	0.367	1.154	CPCo
W5214	Linde 1092	3617	0.291	1.156	CPCo
W5214	Linde 1092	3617	0.27	1.16	CE/CPCO
W5214	Linde 1092	3617	0.29	1.18	CE/CPCO
W5214	Linde 1092	3617	0.365	1.193	CPCo

W5214	1092	3617	0.34	1.23	CPCo
W5214	1092	3617	0.37	1.23	CE/CPCO
W5214	1092	3617	0.37	1.29	CE/CPCO
W5214	1092	3617	0.22	0.96	CPCo
W5214	1092	3617	0.239	0.96	CPCo
W5214	1092	3617	0.237	1.024	CPCo
W5214	1092	3617	0.228	1.107	CPCo
W5214	1092	3617	0.22	1.11	CE/CPCO
W5214	1092	3617	0.233	1.149	CPCo
W5214	1092	3617	0.34	1.15	CE/CPCO
W5214	1092	3617	0.22	1.18	CE/CPCO
W5214	1092	3617	0.353	1.203	CPCo
W5214	1092	3617	0.359	1.204	CPCo
W5214	1092	3617	0.22	1.22	CE/CPCO
W5214	1092	3617	0.35	1.29	CE/CPCO
W5214	1092	3617	0.34	0.78	CPCo
W5214	1092	3617	0.31	1.003	CPCo
W5214	1092	3617	0.31	1.006	CPCo
W5214	1092	3617	0.30	1.05	CE/CPCO
W5214	1092	3617	0.33	1.09	CE/CPCO
W5214	1092	3617	0.266	1.09	CPCo
W5214	1092	3617	0.341	1.093	CPCo

W5214	1092	3617	0.25	1.10	CE/CPCO
W5214	1092	3617	0.266	1.104	CPCo
W5214	1092	3617	0.328	1.116	CPCo
W5214	1092	3692	0.15	1.02	WCAP-11815
W5214	1092	3692	0.166	1.21	WCAP-11815
W5214	1092	3692	0.16	1.06	Westinghouse
W5214	1092	3692	0.15	1.11	Westinghouse
W5214	1092	3692	0.15	1.09	Westinghouse

Table B-9: Summary of Properties for Weld Wire Heat No. 1P3571**

Heat No.	Flux Type	Flux Lot	%Cu	%Ni	Source
1P3571	1092	3958	0.4	0.82	CE D-LOG
1P3571	1092	3958	0.17	0.51	
1P3571	1092	3958	0.15	0.54	
1P3571	1092	3958	0.17	0.61	
1P3571	1092	3958	0.17	0.64	
1P3571	1092	3958	0.18	0.67	
1P3571	1092	3958	0.19	0.67	
1P3571	1092	3958	0.19	0.67	
1P3571	1092	3958	0.186	0.689	
1P3571	1092	3958	0.2	0.7	
1P3571	1092	3958	0.19	0.71	

1P3571	1092	3958	0.172	0.717	
1P3571	1092	3958	0.34	0.72	
1P3571	1092	3958	0.22	0.73	
1P3571	1092	3958	0.191	0.734	WCAP-14280
1P3571	1092	3958	0.066	0.736	
1P3571	1092	3958	0.24	0.74	
1P3571	1092	3958	0.182	0.742	
1P3571	1092	3958	0.354	0.742	
1P3571	1092	3958	0.207	0.769	
1P3571	1092	3958	0.2	0.77	
1P3571	1092	3958	0.43	0.78	
1P3571	1092	3958	0.23	0.79	
1P3571	1092	3958	0.209	0.795	
1P3571	1092	3958	0.22	0.8	
1P3571	1092	3958	0.434	0.8	
1P3571	1092	3958	0.196	0.803	
1P3571	1092	3958	0.214	0.816	
1P3571	1092	3958	0.223	0.871	
1P3571	1092	3958	0.25	0.66	BCL-585-21
1P3571	1092	3958	0.25	0.7	BCL-585-21
1P3571	1092	3958	0.33	0.7	BCL-585-21
1P3571	1092	3958	0.33	0.71	BCL-585-21

1P3571	1092	3958	0.356	0.728	WCAP-12819
1P3571	1092	3958	0.432	0.745	WCAP-12819
1P3571	1092	3958	0.365	0.78	CE D-LOG
1P3571	1092	3958	0.34	0.73	CE D-LOG
1P3571	1092	3958	0.3	0.76	CE D-LOG
1P3571	1092	3958	0.35	0.76	CE D-LOG
1P3571	1092	3958	0.33	0.77	CE D-LOG
1P3571	1092	3958	0.31	0.78	CE D-LOG
1P3571	1092	3958	0.32	0.78	CE D-LOG
1P3571	1092	3958	0.32	0.78	CE D-LOG
1P3571	1092	3958	0.32	0.78	CE D-LOG
1P3571	1092	3958	0.33	0.78	CE D-LOG
1P3571	1092	3958	0.37	0.8	CE D-LOG
1P3571	1092	3958	0.38	0.8	CE D-LOG
1P3571	1092	3958	0.52	0.8	CE D-LOG
1P3571	1092	3958	0.53	0.81	CE D-LOG
1P3571	1092	3958	0.21	0.88	CE D-LOG
1P3571	80	344	0.22	0.67	CE D-LOG
1P3571	1092	3958	0.2	0.73	
1P3571	1092	3958	0.22	0.73	
1P3571	1092	3958	0.2	0.74	
1P3571	1092	3958	0.2	0.75	

1P3571	Linde 1092	3958	0.22	0.75	
1P3571	Linde 1092	3958	0.2	0.76	
1P3571	Linde 1092	3958	0.22	0.79	
1P3571	Linde 1092	3958	0.21	0.8	
1P3571	Linde 1092	3958	0.21	0.8	
1P3571	Linde 1092	3958	0.22	0.8	
1P3571	Linde 1092	3958	0.23	0.82	
1P3571	Linde 1092	3958	0.22	0.83	
1P3571	Linde 1092	3958	0.21	0.78	CE D-LOG
1P3571	Linde 1092	3958		0.76	NEDC-30997
1P3571	Linde 1092	3958	0.28	0.76	NEDC-30997
1P3571	Linde 1092	3958	0.28	0.76	NEDC-30997
1P3571	Linde 1092	3958	0.32	0.82	NE-B1100691-01
1P3571	Linde 1092	3958	0.32	0.87	NE-B1100691-01
1P3571	Linde 1092	3958	0.32	0.87	NE-B1100691-01
1P3571	Linde 1092	3958	0.37	0.75	CE D-LOG

Table B-10: Summary of Properties for Weld Wire Heat No. IP2809

Heat No.	Flux Type	Flux Lot	%Cu	%Ni	Source
IP2809	Linde 1092	3854	0.07	0.74	RPVDATA
IP2809	Linde 1092	3854	0.24	0.77	BGE**

** Data obtained from CEOG database [2]

Table B-11: Summary of Properties for Weld Wire Heat No. 13253/12008**

Heat No.	Flux Type	Flux Lot	%Cu	%Ni	Source
13253/12008	Linde 1092	3774	0.31	0.72	
13253/12008	Linde 1092	3774	0.31	0.72	MDE-49-0386
13253/12008	Linde 1092	3791	0.11	0.72	CE D-LOG
13253/12008	Linde 1092	3791	0.15	0.72	CE D-LOG
13253/12008	Linde 1092	3791	0.12	0.73	CE D-LOG
13253/12008	Linde 1092	3791	0.11	0.74	CE D-LOG
13253/12008	Linde 1092	3791	0.11	0.74	CE D-LOG
13253/12008	Linde 1092	3791	0.11	0.74	CE D-LOG
13253/12008	Linde 1092	3791	0.11	0.74	CE D-LOG
13253/12008	Linde 1092	3791	0.11	0.75	CE D-LOG
13253/12008	Linde 1092	3791	0.11	0.75	CE D-LOG
13253/12008	Linde 1092	3791	0.11	0.75	CE D-LOG
13253/12008	Linde 1092	3791	0.11	0.76	CE D-LOG
13253/12008	Linde 1092	3791	0.22	0.76	CE D-LOG
13253/12008	Linde 1092	3791	0.22	0.76	CE D-LOG
13253/12008	Linde 1092	3791	0.21	0.77	CE D-LOG
13253/12008	Linde 1092	3791	0.22	0.77	CE D-LOG
13253/12008	Linde 1092	3791	0.22	0.77	CE D-LOG

** Data obtained from CEOG database [2]

13253/12008	Linde 1092	3791	0.22	0.78	CE D-LOG
13253/12008	Linde 1092	3791	0.22	0.78	CE D-LOG
13253/12008	Linde 1092	3791	0.22	0.78	CE D-LOG
13253/12008	Linde 1092	3791	0.22	0.78	CE D-LOG
13253/12008	Linde 1092	3791	0.2	0.79	CE D-LOG
13253/12008	Linde 1092	3791	0.2	0.86	CE D-LOG
13253/12008	Linde 1092	3791	0.2	0.86	CE D-LOG
13253/12008	Linde 1092	3791	0.2	0.87	CE D-LOG
13253/12008	Linde 1092	3791	0.2	0.87	CE D-LOG
13253/12008	Linde 1092	3791	0.22	0.87	CE D-LOG
13253/12008	Linde 1092	3791	0.2	0.88	CE D-LOG
13253/12008	Linde 1092	3791	0.21	0.88	CE D-LOG
13253/12008	Linde 1092	3791	0.21	0.88	CE D-LOG
13253/12008	Linde 1092	3791	0.21	0.88	CE D-LOG
13253/12008	Linde 1092	3791	0.22	0.88	CE D-LOG
13253/12008	Linde 1092	3791	0.22	0.88	CE D-LOG
13253/12008	Linde 1092	3791	0.22	0.88	CE D-LOG
13253/12008	Linde 1092	3791	0.2	0.89	CE D-LOG
13253/12008	Linde 1092	3791	0.2	0.89	CE D-LOG
13253/12008	Linde 1092	3791	0.21	0.89	CE D-LOG
13253/12008	Linde 1092	3791	0.21	0.89	CE D-LOG
13253/12008	Linde 1092	3791	0.21	0.89	CE D-LOG

13253/12008	Linde 1092	3791	0.21	0.89	CE D-LOG
13253/12008	Linde 1092	3791	0.21	0.89	CE D-LOG
13253/12008	Linde 1092	3791	0.21	0.89	CE D-LOG
13253/12008	Linde 1092	3791	0.22	0.89	CE D-LOG
13253/12008	Linde 1092	3791	0.22	0.89	CE D-LOG
13253/12008	Linde 1092	3791	0.2	0.9	CE D-LOG
13253/12008	Linde 1092	3791	0.2	0.9	CE D-LOG
13253/12008	Linde 1092	3791	0.2	0.9	CE D-LOG
13253/12008	Linde 1092	3791	0.2	0.9	CE D-LOG
13253/12008	Linde 1092	3791	0.21	0.9	CE D-LOG
13253/12008	Linde 1092	3791	0.21	0.9	CE D-LOG
13253/12008	Linde 1092	3791	0.21	0.9	CE D-LOG
13253/12008	Linde 1092	3791	0.22	0.9	CE D-LOG
13253/12008	Linde 1092	3791	0.22	0.9	CE D-LOG
13253/12008	Linde 1092	3791	0.2	0.91	CE D-LOG
13253/12008	Linde 1092	3791	0.2	0.91	CE D-LOG
13253/12008	Linde 1092	3791	0.2	0.91	CE D-LOG
13253/12008	Linde 1092	3791	0.2	0.91	CE D-LOG
13253/12008	Linde 1092	3791	0.21	0.91	CE D-LOG
13253/12008	Linde 1092	3791	0.2	0.92	CE D-LOG
13253/12008	Linde 1092	3791	0.21	0.92	CE D-LOG
13253/12008	Linde 1092	3791; 3833	0.24	0.875	CE D-LOG

Table B-12: Summary of Properties for Weld Wire Heat No. 10137**

Heat No.	Flux Type	Flux Lot	%Cu	%Ni	Source
10137	Linde 0091	3999	0.22	0.02	CE D-LOG
10137	Linde 0091	3999	0.28	0.02	BAW-2220
10137	Linde 0091	3999	0.18	0.05	BAW-2220
10137	Linde 0091	3999	0.2	0.05	BAW-2220
10137	Linde 0091	3999	0.2	0.05	BAW-2220
10137	Linde 0091	3999	0.19	0.06	BAW-2220
10137	Linde 0091	3999	0.2	0.07	BAW-2220
10137	Linde 0091	3999	0.19	0.08	BAW-2220
10137	Linde 0091	3999	0.21	0.08	BAW-2220
10137	Linde 0091	3999	0.2	0.09	BAW-2220
10137	Linde 0091	3999	0.18	0.11	BAW-2220
10137	Linde 0091	3999	0.19		BAW-2220
10137	Linde 0091	3999	0.2		BAW-2220
10137	Linde 0091	3999	0.23		CE D-LOG
10137	Linde 0091	3999	0.23	0.04	BAW-2220
10137	Linde 0091	3999	0.19	0.05	BAW-2220
10137	Linde 0091	3999	0.24	0.05	SWRI-7524
10137	Linde 0091	3999	0.26	0.059	BAW-2220
10137	Linde 0091	3999	0.24	0.06	SWRI-7524
10137	Linde 0091	3999	0.18	0.07	BAW-2220
10137	Linde 0091	3999	0.24	0.07	BAW-2220
10137	Linde 0091	3999	0.23	0.073	BAW-2220
10137	Linde 0091	3999	0.26	0.077	BAW-2220

** Data obtained from CEOG database [2]

10137	Linde 0091	3999	0.18	0.08	BAW-2220
10137	Linde 0091	3999	0.19	0.09	BAW-2220
10137	Linde 0091	3999	0.23	0.09	BAW-2220
10137	Linde 0091	3999	0.19	0.15	BAW-2220
10137	Linde 0091	3999	0.2	0.04	CE D-LOG
10137	Linde 0091	3999	0.18		BAW-2220
10137	Linde 0091	3999	0.24		BAW-2220
10137	Linde 0091	3999	0.18	0.05	CE D-LOG
10137	Linde 0091	3999	0.18	0.05	CE D-LOG
10137	Linde 0091	3999	0.18	0.05	CE D-LOG
10137	Linde 0091	3999	0.18	0.05	CE D-LOG
10137	Linde 0091	3999	0.22	0.05	CE D-LOG
10137	Linde 0091	3999	0.17	0.06	CE D-LOG
10137	Linde 0091	3999	0.17	0.06	CE D-LOG
10137	Linde 0091	3999	0.18	0.06	CE D-LOG
10137	Linde 0091	3999	0.19	0.06	CE D-LOG
10137	Linde 0091	3999	0.19	0.06	CE D-LOG
10137	Linde 0091	3999	0.21	0.06	CE D-LOG
10137	Linde 0091	3999	0.23	0.06	CE D-LOG
10137	Linde 0091	3999	0.24	0.06	CE D-LOG
10137	Linde 0091	3999	0.2	0.07	CE D-LOG
10137	Linde 0091	3999	0.24	0.07	CE D-LOG
10137	Linde 0091	3999	0.25	0.07	CE D-LOG
10137	Linde 0091	3999	0.27	0.07	CE D-LOG
10137	Linde 0091	3999	0.17	0.09	CE D-LOG
10137	Linde 0091	3999	0.215	0.02	CE D-LOG

Table B-13: Summary of Properties for Weld Wire Heat No. 90099**

Heat No.	Flux Type	Flux Lot	%Cu	%Ni	Source
90099	Linde 0091	3458	0.25		CE D-LOG
90099	Linde 0091	3922	0.09		CE D-LOG
90099	Linde 0091	3950	0.19		CE D-LOG
90099	Linde 0091	3977	0.17		CE D-LOG
90099	Linde 0091	3977	0.18		CE D-LOG
90099	Linde 0091	3999	0.30		CE D-LOG
90099	Linde 0091	3458	0.22		CE D-LOG
90099	Linde 0091	3950	0.17		CE D-LOG
90099				0.06	Bare Wire

** Data obtained from CEOG database [2]

About the BWR Vessel and Internals Project

The BWR Vessel and Internals Project (BWRVIP) is an association of utilities owning and operating boiling water reactors. The project is focused exclusively on reactor vessel and vessel internals issues in operating plants. Objectives of the BWRVIP are to lead the BWR industry toward generic resolution of vessel and internals integrity and operability issues; to identify or develop generic, cost-effective strategies from which each operating plant will select the most appropriate alternative; to serve as the focal point for the regulatory interface with the industry on BWR vessel and internals integrity and operability issues; and to share information among members. EPRI manages the technical program on behalf of the utility members of the BWRVIP.

EPRI

Electricity is increasingly recognized as a key to societal progress throughout the world, driving economic prosperity and improving the quality of life. The Electric Power Research Institute delivers the science and technology to make the generation, delivery, and use of electricity affordable, efficient, and environmentally sound.

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September 15, 1999

Ms. Roberta Dalton, Program Manager
Work-for-Others - Argonne Group
Chicago Operations Office
U.S. Department of Energy
9800 South Cass Avenue
Argonne, Illinois 60439

SUBJECT: AUTHORIZATION OF TASK ORDER 5, "REVIEW OF EPRI TOPICAL REPORT TR-108705, "BWR VESSEL AND INTERNALS PROJECT - TECHNICAL BASIS FOR INSPECTION RELIEF FOR BWR INTERNAL COMPONENTS WITH HYDROGEN INJECTION (BWRVIP-62)" UNDER JCN J-2410

Dear Ms. Dalton:

The enclosed NRC Form 173, Standard Order for DOE Work, is submitted in accordance with Section III.B.2 of the DOE/NRC Memorandum of Understanding dated February 24, 1978. Please indicate DOE acceptance by signing and returning the enclosed NRC Form 173 as stated on the form.

If you have any questions concerning acceptance of this order, please contact Mr. Bernard L. Grenier, Program Manager on 301/415-2726. Thank you for your assistance.

Sincerely,

original signed by Richard Wessman for:
Jack R. Strosnider, Director
Division of Engineering
Office of Nuclear Reactor Regulation

Enclosure:
NRC Form 173

cc: R. Combs, ANL
Y. Y. Liu, ANL
W. J. Shack, ANL

DISTRIBUTION

Official Contract File (Original Concurrence Copy) (7-D-26), Section 2 and Task Order 5
CECarpenter, KRWichman, RAHermann

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BLGrenier	CECarpenter	KRWichman	WHBateman	RHWessman	JRStrosnider	TDChampion
09/9/99	09/8/99	09/8/99	09/10/99	09/10/99	09/10/99	9/14/99 09/15/99

6/2

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

September 15, 1999

Ms. Roberta Dalton, Program Manager
Work-for-Others - Argonne Group
Chicago Operations Office
U.S. Department of Energy
9800 South Cass Avenue
Argonne, Illinois 60439

SUBJECT: AUTHORIZATION OF TASK ORDER 5, "REVIEW OF EPRI TOPICAL REPORT TR-108705, "BWR VESSEL AND INTERNALS PROJECT - TECHNICAL BASIS FOR INSPECTION RELIEF FOR BWR INTERNAL COMPONENTS WITH HYDROGEN INJECTION (BWRVIP-62)" UNDER JCN J-2410

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for Jack R. Strosnider, Director
Division of Engineering
Office of Nuclear Reactor Regulation

Enclosure:
NRC Form 173

cc: R. Combs, ANL
Y. Y., Liu, ANL
W. J. Shack,

NRC FORM 173
(4-95)
NRC MD 11.7

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD ORDER FOR DOE WORK (SOEW)

ORDER NUMBER: 20-99-162

DATE OF ISSUANCE: Sept 15, 1999

This agreement is entered into pursuant to the authority of the Energy Reorganization Act of 1974, as amended (42 U.S.C. 5801 et seq.). This work will be performed in accordance with NRC Management Directive 11.7.

ISSUED TO: *(DOE OFFICE)* **CHICAGO OPERATIONS OFFICE**	**ISSUED BY:** *(NRC OFFICE)* Office of Nuclear Reactor Regulation Division of Engineering	**ACCOUNTING CITATION** APPROPRIATION SYMBOL: 31X0200.920 B&R NUMBER: 920-15-101-130
PERFORMING ORGANIZATION Argonne National Laboratory Argonne, Illinois	**DISTRIBUTION OF OUTGOING** *(NRC SIGNED)*: Official Contract File	JOB CODE: J-2410-9 BOC CODE: 253D
JOB CODE TITLE Technical Assistance in Support of the Division of Engineering Review of Operating Reactors Licensing Actions, Related licensing activities and Topical Reports	**DISTRIBUTION OF INCOMING** *(DOE SIGNED)*: Official Contract File See distribution in Remarks section below.	**PROJECT PERFORM. PERIOD** BEGIN: 07/15/96 END: 03/31/00 **THIS FY FUNDING PERIOD** FROM: 09/13/99 TO: 02/25/00

OBLIGATION AVAILABILITY PROVIDED BY:

A. THIS ORDER	$53,200
B. TOTAL OF ORDERS PLACED PRIOR TO THIS ORDER WITH THE PERFORMING ORGANIZATION UNDER THIS JOB CODE FOR THIS FISCAL YEAR.	$93,700
C. TOTAL ORDERS TO DATE FOR THIS JOB CODE FOR THIS FISCAL YEAR. (TOTAL A & B)	$146,900

STANDARD TERMS AND CONDITIONS ARE PART OF THIS ORDER UNLESS OTHERWISE NOTED *(See NRC Management Directive 11.7)*.

ATTACHMENTS	SECURITY
THE FOLLOWING ATTACHMENTS ARE HEREBY MADE A PART OF THIS ORDER ☐ STATEMENT OF WORK ☐ ADDITIONAL TERMS AND CONDITIONS ☐ OTHER *(Specify)*	☐ WORK ON THIS ORDER INVOLVES CLASSIFIED INFORMATION. NRC FORM 187 IS ATTACHED. ☐ WORK ON THIS ORDER INVOLVES SENSITIVE UNCLASSIFIED, UNCLASSIFIED SAFEGUARDS INFORMATION, OR UNESCORTED ACCESS TO PROTECTED AND VITAL AREAS OF NUCLEAR POWER PLANTS. NRC FORM 187 IS ATTACHED.
FEE BILLABLE UNDER 10 CFR PART 170 ☐ YES ☒ NO	☒ WORK ON THIS ORDER IS UNCLASSIFIED AND NOT SENSITIVE.

REMARKS: *(At a minimum, reference the approved proposal (NRC Form 189) by number and date. See further instructions on the reverse side.)*

This order provides funds for completion of work on Task Order 5 which is attached; this work falls within the scope of the ANL proposal signed on May 20, 1996. The period of performance is extended to March 31, 2000.

Task Order 5 Authorized Cost Ceiling: $53,200
Task Order Period of Performance: 09/13/99 to 02/25/00

Project Officer: Bernard L. Grenier

After acceptance please mail the original to the NRC Office of the Controller, ATTN: M. Givvines, T-9E-2, and mail copies to the Office of Nuclear Reactor Regulation, ATTN: B. L. Grenier, O-07-D-26 and T. D. Champion, O-05-A-2.

CERTIFICATION OF FUNDS

This certifies that funds in the amount cited in Block A are available in the current Fiscal Year allotment for work authorized by this SOEW.

FUNDS CERTIFICATION OFFICIAL *(Typed Name)*	SIGNATURE	DATE
Tanya D. Champion	Tanya D. Champion	9/15/99

SIGNATURES

NRC ISSUING AUTHORITY *(Typed Name and Title)*	SIGNATURE	DATE
Jack R. Strosnider, Director, Div. of Engineering, NRR	for JRS	9/10/99
DOE ACCEPTING ORGANIZATION *(Typed Name and Title)*	SIGNATURE	DATE

PRINTED ON RECYCLED PAPER

F182

Statement of Work for
Task Order 5 Under JCN J-2410

Title: Review of EPRI Topical Report TR-108705, "BWR Vessel and Internals Project - Technical Basis for Inspection Relief for BWR Internal Components with Hydrogen Injection (BWRVIP-62)"

Technical Monitor: C. E. Carpenter, 301-415-2169; E-mail: CEC@NRC.GOV

TAC Number: MA4468

BACKGROUND

BWR availability has been negatively impacted by the IGSCC of austenitic stainless steel piping and, more recently, reactor pressure vessel (RPV) internal components. Regular inspection is necessary for BWR piping to provide adequate assurance of structural integrity of affected piping systems. Similar inspections may be required for RPV internal components. However, due to the difficulty and expense of RPV internals inspections, it is clearly desirable to demonstrate that fewer inspections are necessary when suitable RPV internals IGSCC mitigation steps are taken.

Usage of a hydrogen water chemistry (HWC) process for the mitigation of intergranular stress corrosion cracking (IGSCC) of RPV internals has been developed along two parallel paths to provide RPV internals IGSCC mitigation. The first mitigation technique, moderate HWC (HWC-M), involves higher hydrogen injection rates in the feedwater compared to the recirculation piping. The second mitigation technique, noble metal chemical application (NMCA), involves the continuous injection of a small amount of hydrogen to give a hydrogen to oxygen molar ratio greater than 2 plus an occasional batch injection of noble metal compounds that act as catalysts for the various recombination reactions.

OBJECTIVE

The objective of this report is to obtain technical expertise from the Argonne National Laboratory (ANL) to assist the staff in determining the safety adequacy of the methodologies contained within the above referenced topical report for demonstrating that either HWC-M or NMCA provides IGSCC mitigation of BWR internals, and that inspection relief is justified for BWR internals at plants that have effectively implemented HWC-M or NMCA through the use of crack growth rate factors of improvement (FOIs).

TECHNICAL AND OTHER SPECIAL QUALIFICATION REQUIRED

Key Personnel: Dr. Bill J. Shack

WORK REQUIREMENTS

Tasks	Completion Schedule
1. Evaluate BWRVIP Topical Report TR-108705, "BWR Vessel and Internals Project - Technical Basis for Inspection Relief for BWR Internal Components with Hydrogen Injection (BWRVIP-62)," using the best engineering judgment and results of ongoing NRC environmental cracking RES program and other related research programs. Specifically:	October 22, 1999

Tasks	Completion Schedule
o Determine the validity and applicability of the BWRVIP's proposed technical basis for HWC mitigation, including the roles of ionic impurities in the BWR coolant and of the oxidizing environment in the BWR, and the effect of irradiation.	
o Determine the validity and applicability of the BWRVIP / GE / Harwell radiolysis and ECP model.	
o Determine the validity and applicability of the HWC effectiveness assessment, including the validity of monitoring secondary parameters to determine the effectiveness of HWC and/or NMCA in BWRS; and, the BWRVIP crack growth modeling for stainless steel and nickel-base alloys.	
o Determine the validity and applicability of the submitted technical basis for the proposed inspection relief.	
o Identify areas in TR -108710 that need additional or clarifying information and prepare a technical letter report.	
2. Identify, to the extent possible, alternative approaches to that proposed.	November 14, 1999
3. Review and evaluate for acceptability the responses from the RAIs and prepare a technical letter report.	Three weeks after receipt of responses.
4. Prepare for and travel to NRC Headquarters to present the results of the review to date and discuss issues identified in the review, after coordination with the Technical Monitor and acknowledgment in writing by the Project Officer. BWRVIP members may be present at the meeting for consultation. Prepare a trip report.	One week after the meeting.
5. Review and evaluate any additional information resulting from the meeting and prepare a Technical Letter Report.	
a. Draft.	Three weeks after completion of Task 4, or three weeks after receipt of any additional information.
b. Incorporate NRC comments and prepare the final TLR.	Two weeks after receipt of NRC comments.

LEVEL OF EFFORT

The estimated level of effort in professional staff hours apportioned among the tasks are as follows:

Tasks	Labor Category		Level of Effort (days)
1.	Senior Materials Engineer (IGSCC)		15.0
	Senior Mechanical Engineer (Fracture Mechanics)		6.0
	Senior Metallurgist (Statistics)		6.0
		Subtotal	27.0
2.	Senior Materials Engineer (IGSCC)		2.5
	Senior Mechanical Engineer (Fracture Mechanics)		2.0
		Subtotal	4.5
3.	Senior Materials Engineer (IGSCC)		3.0
	Senior Mechanical Engineer (Fracture Mechanics)		2.0
	Senior Metallurgist (Statistics)		2.0
		Subtotal	7.0
4.	Senior Materials Engineer (IGSCC)		2.0
5.	Senior Materials Engineer (IGSCC)		6.0
	Senior Mechanical Engineer (Fracture Mechanics)		2.5
	Senior Metallurgist (Statistics)		2.5
		Subtotal	11.0
	Project Management		3.5
		Total	55.0

PERIOD OF PERFORMANCE

The period of performance is September 13, 1999 to February 25, 2000.

DELIVERABLES

Technical Reporting Requirements

NOTE: All reports are to be submitted electronically to the Technical Monitor with a copy provided to the Project Officer. Documents are to be prepared in WordPerfect 8.0, or using compatible software.

1. At the completion of Task 1, submit a technical letter report that contains a preliminary assessment of the Topical Report to include:

   a. the results with supporting information for determining the validity and applicability of the BWRVIP's proposed technical basis for HWC mitigation, including the roles of ionic impurities in the BWR coolant and of the oxidizing environment in the BWR, and the effect of irradiation,

   b. the results with supporting information for determining the validity and applicability of the BWRVIP / GE / Harwell radiolysis and ECP model,

Technical Reporting Requirements (Continued)

c. the results with supporting information for determining the validity and applicability of the HWC effectiveness assessment, including the validity of monitoring secondary parameters to determine the effectiveness of HWC and/or NMCA in BWRS; and, the BWRVIP crack growth modeling for stainless steel and nickel-base alloys, and

d. the results with supporting information for determining the validity and applicability of the submitted technical basis for the proposed inspection relief.

f. identify specific areas in TR-108705 that need additional or clarifying information and provide a request for additional information.

2. At the completion of Task 2, submit a technical letter report that contains any alternative approaches to that proposed.

3. At the completion of Task 3, submit a Technical Letter Report that contains a brief one- or two-page summary of the results and assessment of the review of the RAIs.

4. At the completion of Task 4, submit a trip report that contains a brief summary of major highlights of the meeting and major discussion points, decisions made and agreements reached, open issues, list of attendees, and action items. For action items, identify who is responsible for follow-up, who will resolve or provide information, and when will the action items be completed.

5. At the completion of Tasks 5, submit a Technical Letter Report, draft and final as appropriate, that contains a summary of the work performance, discussion of the results of the review efforts, the details of the specific review that was performed, findings, conclusions, recommendations with bases on the acceptability or non-acceptability of using the proposed technical basis for inspection relief for BWR internal components with hydrogen injection, and any proposed alternate approaches that could justify inspection relief other than that proposed.

NOTE: Care should be taken to avoid the inclusion of any proprietary information in the report. The final report should be submitted in original, three hard copies in addition to the electronic copy.

MEETINGS AND TRAVEL

One one-person, two-day trip to the NRC headquarters in Rockville, Maryland.

NRC-FURNISHED MATERIALS

A copy of the EPRI Topical Report TR-108705, "BWR Vessel and Internals Project - Technical Basis for Inspection Relief for BWR Internal Components with Hydrogen Injection (BWRVIP-62)," was forwarded to the Principal Investigator on August 10, 1999.

OTHER APPLICABLE INFORMATION

License Fee Recovery

The work to be performed under this project is non-licensee fee recoverable.

Assumptions and Understandings

None.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

December 3, 1999

Carl Terry, BWRVIP Chairman
Niagara Mohawk Power Company
Post Office Box 63
Lycoming, NY 13093

SUBJECT: FINAL SAFETY EVALUATION OF PROPRIETARY REPORT TR 105873 "BWR VESSEL AND INTERNALS PROJECT, EVALUATION OF CRACK GROWTH IN BWR STAINLESS STEEL INTERNALS (BWRVIP-14)" (TAC NO. M94975)

Dear Mr. Terry:

The NRC staff has completed its review of the proposed revisions to the Electric Power Research Institute (EPRI) proprietary report TR-105873, "BWR Vessel and Internals Project, Evaluation of Crack Growth in BWR Stainless Steel Internals (BWRVIP-14)," dated March 1996. This report was submitted by letter dated July 28, 1997, as supplemented by letter dated July 28, 1997, for NRC staff review and approval.

On June 8, 1998, the NRC staff issued its initial safety evaluation (SE) of the BWRVIP-14 report, which found the BWRVIP-14 evaluation of crack growth to be acceptable for use except where the staff's conclusions differed from the BWRVIP's. The BWRVIP was requested to resolve the open issues raised in the staff's initial SE. By letter dated November 24, 1998, you provided a response which proposed to resolve the issues.

The NRC staff, with assistance from the Argonne National Laboratory (ANL), has reviewed your responses to the staff's initial SE and finds, in the enclosed technical evaluation report (TER) prepared by ANL, with one exception, that your response to the open issues is acceptable. The exception is your conclusion that the use of the generic stress intensity factors, provided in Figure 4 of the response, avoids the need to document weld repairs when using a K based approach for estimating crack growth rates (CGR).

Specifically, the generic stress intensity factor distributions proposed in your response will generally be conservative for cylinder-to-cylinder horizontal welds and cylinder-to-edge-ring welds. However, because local weld repairs inherently introduce three dimensional effects, this is an artificial constraint and may lead to nonconservative estimates of the local stresses. In most cases, such local effects would have little impact on the overall structural integrity of the components, but, as the extent of the cracking increases, the uncertainties introduced by such local effects could become more significant and cannot be ignored. If the extent of weld repairs cannot be documented, the bounding CGR approach should be used for safety evaluations.

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FOIA 2000-0186

We believe that, by using an appropriately reduced value for the CGR from the 5×10^{-5} in/hr value found in NUREG-0313, Rev. 2, it would be possible for licensees to get credit for improved water chemistry and other measures to mitigate cracking, e.g., hydrogen water chemistries (HWC) and/or noble metal additions. The revised CGR of 2.2×10^{-5} in/hr corresponds to water chemistries with a conductivity of $\leq$ 0.15 µS/cm and an electro-chemical potential (ECP) of +200 mV. The BWRVIP-14 correlation indicates that this bounding CGR could be reduced for HWC with ECP $\leq$ -230 mV. The staff finds acceptable a reduction in the CGR from 2.2×10^{-5} in/hr to 1.1×10^{-5} in/hr for plants with HWC. The crack growth rates stated are only applicable to components with fluences $< 5 \times 10^{20}$ n/cm^2 (E > 1MeV) since the CGR database is presently based only on unirradiated materials.

The above is discussed in greater detail in the enclosed TER, and was discussed with members of the BWRVIP during a telephone call on October 18, 1999. Therefore, based on the information you have provided, the staff has concluded that licensee usage of the BWRVIP-14 guidelines as revised is acceptable.

The staff requests that you incorporate the staff's recommendation regarding the above issue on generic stress intensity factors, as well as your response to other issues raised in the staff's initial SE, into a revised, final BWRVIP-14 report. Please inform the staff within 90 days of the date of this letter as to your proposed actions and schedule for such a revision.

Please contact C. E. (Gene) Carpenter, Jr., of my staff at (301) 415-2169 if you have any further questions regarding this subject.

Sincerely,

Jack R. Strosnider, Director
Division of Engineering
Office of Nuclear Reactor Regulation

Enclosure: TER

cc: See next page

cc:

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Technical Evaluation Report on Project No. 704—*BWRVIP Response to NRC Safety Evaluation of BWRVIP–14*

August 5, 1999

Prepared by
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S. Majumdar

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Technical Monitor
Donald Naujock
USNRC Office of Nuclear Reactor Regulation

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Introduction

This technical evaluation report reviews Project 704, *BWR Vessels and Internals Project, Response to NRC Safety Evaluation of BWRVIP-14*. The overall conclusions of the ANL Assessment are:

1. We agree with the BWRVIP conclusion that the crack growth rate of 2.2×10^{-5} in/hr is sufficiently conservative that it should bound crack growth in cylinder-to-cylinder horizontal welds, shroud vertical welds, and cylinder-to-edge-ring welds. These would include the H1-H7 welds for BWR 3/4 models (Fig. 1-2 of BWRVIP-14) and the H3-H7 welds for BWR 6 models (Fig. 1-3 of BWRVIP-14).

2. We do not agree with the BWRVIP conclusion that the use of the generic stress intensity factor factors provided in Fig. 4 of the response avoids the need to document weld repairs when using a K based approach for estimating CGRs. The generic stress intensity factor distributions proposed in the BWRVIP response will generally be conservative for cylinder-to-cylinder horizontal welds and cylinder-to-edge-ring welds. However, it restricts the local stress distributions to be self-equilibrating at every azimuth. Because local weld repairs inherently introduce three dimensional effects, this is an artificial constraint and may lead to nonconservative estimates of the local stresses. In most cases such local effects would have little impact on the overall structural integrity of the shroud, but as the extent of the cracking increases the uncertainties introduced by such local effects could become more significant and cannot be ignored. If the extent of weld repairs cannot be documented, the bounding CGR approach should be used for safety evaluations.

   We believe that by using an appropriately reduced value for the bounding CGR, it still would be possible for licensees to get credit for improved water chemistry and other measures to mitigate cracking such as noble metal additions. The bounding crack growth rate of 2.2×10^{-5} in/hr corresponds to water chemistries with a conductivity of ≤ 0.15 μS/cm and an ECP of 200 mV. The BWRVIP-14 correlation indicates that this bounding CGR could be reduced by a factor of 4.5 for hydrogen water chemistries (HWC) with ECP ≤ -230 mV. The staff has already concluded in its SER that the estimates of the benefits of improved water chemistry provided by the BWRVIP-14 correlation are conservative, and hence we believe that it can be used to determine appropriate factors of reduction in the current bounding CGR.

3. As the BWRVIP has noted in its response to an earlier RIA on BWRVIP-14 and in the response to the staff SER, the stress intensity factor solutions in BWRVIP-14 using the Bamford and Buchelet weight functions, which were determined for $R/t = 10$, may give significantly nonconservative values for K and should not be used. The BWRVIP has presented improved stress intensity factor solutions based on the work of Cheng and Finnie. These solutions take into account the R/t ratio and have been benchmarked with the solutions by Labbens and Bamford and Buchelet for $R/t = 10$ and the limiting case of the flat plate ($R/t = \infty$).

Discussion

The BWRVIP response to the NRC Safety Evaluation of BWRVIP-14 discusses a number of issues. These are addressed on a point by point basis.

BWRVIP Response to Issues Identified in the NRC Safety Evaluation Report

1. Item 1, pg. 2: The BWRVIP argues that the effect of weld repairs, fit-up stresses can be bounded for core shroud horizontal and vertical welds and that the bounding crack growth rate of 2.2×10^{-5} in/hr should be applicable for these welds without plant specific NRC review of weld repairs and residual stress distributions. (Item 1 of the BWRVIP response essentially also includes their Item 3 which rephrases the argument to consider vertical welds.)

   Assessment:

   We agree with the BWRVIP conclusion that the crack growth rate of 2.2×10^{-5} in/hr is sufficiently conservative that it should bound crack growth in cylinder-to-cylinder horizontal welds, shroud vertical welds, and cylinder-to-edge-ring welds.

   This conclusion is supported by detailed analyses of the residual stresses and crack growth for these geometries which show that typically the crack growth rates should be much less than 2.2×10^{-5} in/hr. Because a CGR of 2.2×10^{-5} in/hr corresponds to growth of a throughwall crack in about 7 years, field experience suggests that even in situations where high local residual stresses are present so that crack arrest does not occur, this is still a conservative estimate of the average throughwall CGR.

   BWRVIP-14 presents residual stress calculations and measurements on a number of core shroud cylinder-to-cylinder welds and cylinder-to-edge-ring welds. It also defines a generic residual stress distribution for such welds. The calculated and measured distributions are compared in Figs. 4-4 through 4-11. It can be difficult to interpret whether one throughwall residual stress distribution is conservative with respect to another. It is much more meaningful to compare the corresponding distributions of K, because it really controls the crack growth. The throughwall variation of the stress intensity factors which correspond to the various throughwall residual stress distributions given in BWRVIP-14 are shown in Fig. 1. The K solution corresponding to the "generic" BWRVIP residual stress distribution is bounding in the sense that it predicts crack arrest at a larger depth than all but one of the distributions, the values in the positive portion of the curve are greater than those of the other distributions at most locations, and it is less compressive for deep cracks so that any prediction of crack arrest in the presence of additional stresses is conservative.

2. Item 2, pg. 2: The BWRVIP notes that although the NRC SER indicates that an electrochemical potential (ECP) of 200 mV (SHE) for stainless steel was typical of BWRs operating under moderate hydrogen water chemistry, this ECP is actually typical of normal water chemistry operations.

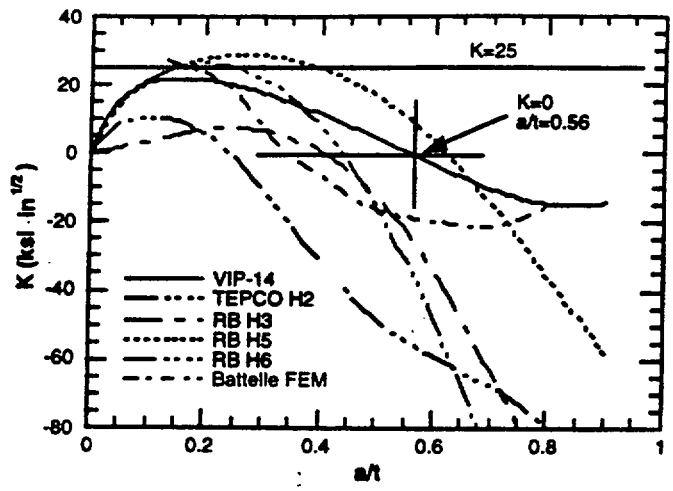


Figure 1
Throughwall variation of the stress intensity factor K for various cylinder-to-cylinder and cylinder to edge-ring welds based on the corresponding residual stress distributions reported in BWRVIP-14 and in independent calculations by Battelle.

Assessment:

The BWRVIP observation is correct. An ECP of 200 mV would be typical (perhaps somewhat conservative) for the core shroud for normal water chemistry conditions.

3. Item 3 pg.2: The BWRVIP argues that the residual stress distributions for vertical double V groove welds are similar to those for double V groove horizontal welds for which more measurements and calculations are available. Hence the CGR of 2.2 x 10^{-5} in/hr is also bounding for these welds.

Assessment:

We agree with BWRVIP argument and believe, as noted in our assessment of Point 1, that the crack growth rate of 2.2 x 10^{-5} in/hr is sufficiently conservative that it should bound crack growth shroud vertical welds as well as in cylinder-to-cyclinder horizontal welds and cylinder-to-edge-ring welds.

4. Item 4 pg.2: The BWRVIP in its response defines a new K distribution (Fig. 4 of the BWRVIP response) that it argues can be used on a generic basis for core shroud horizontal welds to determine K distributions that can be used with K dependent CGR models such as the BWRVIP-14 correlation or PLEDGE.

Assessment:

Despite the arguments of the BWRVIP, we believe that if the extent of weld repairs cannot be documented, then for safety evaluations the bounding CGR approach should be used rather than a K dependent CGR approach. For its own planning purposes the licensee may choose to use the conservative, but not necessarily bounding, K distributions proposed by the BWRVIP.

It would still be possible for the licensee to take credit for improved water chemistry and other measures to mitigate cracking by reducing the bounding CGR appropriately. The bounding CGR, 2.2 x10^{-5} in/h, corresponds to water chemistries with a conductivity

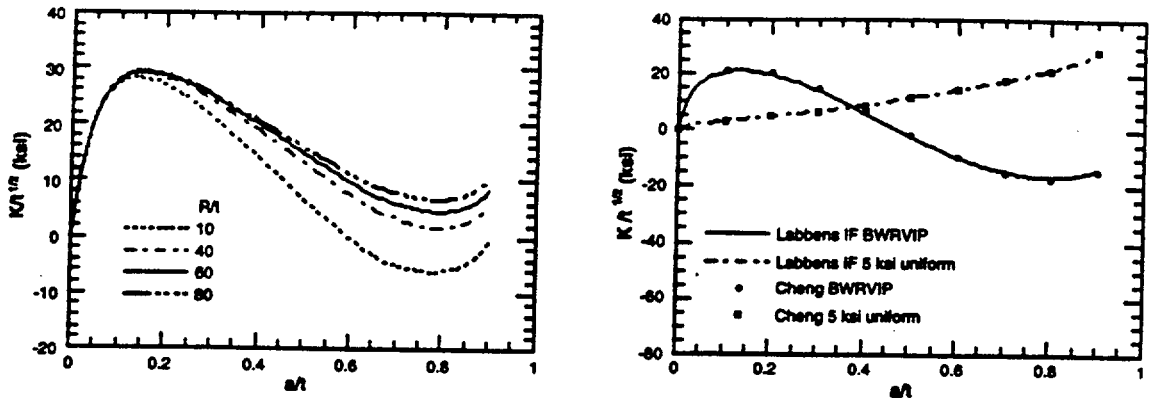
of ≤ 0.15 μS/cm and an ECP of 200 mV. The BWRVIP–14 correlation indicates that this bounding CGR could be reduced by a factor of 4.5 for hydrogen water chemistries (HWC) with ECP ≤ –230 mV.

The new K distribution is based on a stress distribution (Figs. 1 and 2 of 704) that consists of the generic residual stress distribution given in BWRVIP–14, plus a conservative estimate of the membrane stresses acting on the shroud, and plus an estimate of the local stresses due to weld repair. The overall stress state is still dominated by the welding residual stresses. As noted in the assessment of Item 1, we agree that the generic welding residual stress distribution given in BWRVIP–14 is conservative for cylinder–to–cylinder horizontal welds, shroud vertical welds, and cylinder–to–edge–ring welds. The estimate of the membrane stresses is also clearly conservative. However, the basis for the selection of the surface residual stress distribution used to account for weld repairs and other local phenomena is unclear. There is a reference in the BWRVIP response to FEA of the effects of weld repairs performed as part of the BWRVIP–14 study (pg. 6), but the results of those analyses do not appear to be included in the BWRVIP–14 report.

The local residual stress distribution proposed by the BWRVIP is forced to be self–equibrating through the thickness, i.e., the distribution is self–aquilibrating at every azimuth. However, local weld repairs lead to three dimensional nonaxisymmetric stress states, and in such case the axial stresses need not be self equilibrating at every azimuth and it is certainly possible that at some azimuths high tensile stresses persist to greater depths than assumed in the BWRVIP estimate.

The BWRVIP response does recognize that the proposed distribution need not be bounding at every location. They argue that the local regions where the stresses could be higher are limited in extent and hence do not affect the overall structural integrity. If the cracking is limited in extent and the crack size much smaller than the critical size for failure, the BWRVIP argument is valid, but in such cases one could probably just use the bounding CGR approach. If the crack size is close to critical, then the sizes and nature of the local regions in which the stresses are more severe than assumed by the BWRVIP could be more important. Local perturbations in stress can have dramatic effects on CGRs. From Figure 5 in the BWRVIP response, it can be seen that an increase of 1 ksi in the average throughwall stress could reduce the time required for the crack to grow throughwall from 30 years to 5 years.

There is substantial experience in the calculation and measurement of residual stresses for "good" welds. There is much less information available on the nature and distribution of the local stresses due to repairs. In the ANL TLR on BWRVIP–14, the approach taken was to assume that experience permits a good estimate of a bounding CGR, in particular 2.2×10^{-5} in/h, which corresponds to 0.2 in/y. The throughwall average stress which had to be added to the generic welding residual stress to obtain this average CGR was ≈10 ksi. As the BWRVIP response notes such a stress state could not exist over the entire circumference, but it was intended to be used to estimate CGR only in local weld repair regions. It probably would have been simpler to just state that the bounding CGR should be used in local weld repair regions and a "best estimate" CGR used elsewhere.



(a) (b)

Figure 2 *(a) Distributions of $K/t^{1/2}$ for the proposed generic stress distribution in the BWRVIP response for R/t = 10, 40, 60, 80. (b) Comparison of influence function solutions for K for a uniform stress and the BWRVIP-14 residual stress distribution with the Cheng and Finnie solutions for these distributions.*

As the BWRVIP has noted in its response to an earlier RIA on BWRVIP-14 and in the response to the staff SER, the stress intensity factor solutions in BWRVIP-14 using the Bamford and Buchelet weight functions, which were determined for R/t = 10 may give significantly nonconservative values for K and should not be used. The BWRVIP has presented improved stress intensity factor solutions based on the work of Cheng and Finnie. These solutions take into account the R/t ratio and have been benchmarked with the solutions by Labbens and Bamford and Buchelet for R/t = 10 and the limiting case of the flat plate (R/t = ∞) for a variety of stress distributions. In Fig 2a are shown the K solutions for R/t = 10, 40, 60, 80 for the proposed generic weld residual stress in the BWRVIP response to the SER. The solutions for R/t = 10 are nonconservative. They predict arrest for the new generic stress distribution, whereas the solutions for R/t ≥ 40 predict throughwall crack growth.

These Cheng and Finnie solutions take into account the R/t ratio and have been benchmarked with the solutions by Labbens and Bamford and Buchelet for R/t = 10 and the limiting case of the flat plate (R/t = ∞). Comparisons with the Labbens influence function solutions for a uniform stress and the BWRVIP-14 residual stress with the corresponding Cheng and Finnie solutions for R/t=10 are shown in Fig. 2b.

5. Item 5 pg. 2: The BWRVIP concurs that application of BWRVIP-14 above a fluence level of 5×10^{20} n/cm^2 requires plant specific analysis.

BWRVIP Response to NRC Evaluation and Observations

1. Item 1, pg. 4: The BWRVIP observes that in most cases the value of 2.2×10^{-5} in/h will be very conservative for core shrouds. They state that "In view of the conservatism associated with this number, the BWRVIP believes that where significant improvement in the water chemistry and ECP are maintained by the plant, this CGR rate can be substantially reduced by use of the empirical model in BWRVIP-14."

Assessment

We agree that the CGR of 2.2×10^{-5} in/h is expected to be conservative for most locations. Indeed, away from weld repairs, crack growth in the core shroud would be expected to arrest about half way through the wall.

It is not clear to us exactly what is being argued in the last sentence of the paragraph. For a plant with normal water chemistry, but low conductivity and for which the regions of local weld repair can be appropriately characterized, that the empirical model in BWRVIP-14 can be applied and will in general predict lower crack growth rates. We do not agree as discussed previously under element 4 that the generic residual stress/K solutions provided in the BWRVIP response can be assumed to be bounding without additional knowledge of the weld repair history. If local weld repairs can't be characterized, we believe that credit can still be given for improved water chemistries such as hydrogen water chemistry by using an appropriately reduced value of the bounding CGR. For example, for a plant with HWC so that the conductivity is ≤ 0.15 µS/cm and ECP ≤ -230 mV the empirical model in BWRVIP-14 predicts that the CGR is a factor of 4.5 than a plant with an ECP = 200 mV. For such a plant an appropriate bounding CGR might be:

$$2.2 \times 10^{-5} / 4.5 = 4.9 \times 10^{-6} \text{ in/h.}$$

The reduction factor for the lower ECP predicted by the BWRVIP-14 model is quite conservative compared with the reductions predicted by other models such as PLEDGE and observed in most laboratory and in-reactor tests.

2. Item 2, pg. 4: The bounding CGR of 2.2×10^{-5} in/h is appropriate for BWRs under normal water chemistry in which case the in-core ECP would be expected to be on the order of 200 mV. Hydrogen water chemistry conditions are not required for use of this bounding CGR.

   Assessment

   We agree.

3. Item 3, pg. 4: The BWRVIP cites data from NDE measurements at a Swiss BWR at a fluences up to 1×10^{21} n/cm^2 indicating that little or no crack growth was occurring. They argue that these data suggest that the effects of irradiation on IGSCC crack growth are limited and may be bounded by thermally sensitized data.

   Assessment

   We do not believe any conclusion can be drawn from the cited data on the effects of irradiation on IGSCC crack growth.

   The slow crack growth in the Swiss core shroud could well be due to a favorable residual stress distribution. It is almost impossible to deduce anything about the basic crack growth response of a material from such measurements because the effects of other variable like the residual stresses cannot be isolated. It is probably true that chromium depletion levels due to irradiation are bounded by those in heavily thermally sensitized

materials. Irradiation, however, also has profound effects on mechanical properties which could strongly affect crack tip strain rates, and without additional data it is premature to conclude that the CGRs on thermally sensitized materials bound those for irradiated materials. The limited data on IASCC CGRs currently available seem to indicate that for equivalent environmental and loading conditions the CGRs for irradiated materials may be greater than those in materials thermally sensitized by typical welding practices.

4. Item 4, pg. 5: BWRVIP argues again that the generic welding residual stress in BWRVIP-14 is applicable to both cylinder-to-cylinder horizontal welds and cylinder-to-ring welds.

   Assessment

   We agree.

   As noted in our assessment of Item 1, pg. 2, BWRVIP-14 presents residual stress calculations and measurements on a number of core shroud cylinder-to-cylinder welds and cylinder-to-edge-ring welds. It also defines a generic residual stress distribution for such welds. The K solution corresponding to the "generic" BWRVIP residual stress distribution is appropriately bounding, and hence we agree that the generic residual stress distribution is applicable to both cylinder-to-cylinder horizontal welds and cylinder-to-ring welds.

5. Item 5, pg. 5: The BWRVIP seems to believe the use of a bounding CGR is appropriated only for plants with normal water chemistry and that plants using hydrogen water chemistry will use K dependent CGRs based on the BWRVIP-14 correlation.

   Assessment

   The BWRVIP seems to make the distinction between whether a plant uses the bounding approach or a K dependent CGR approach based on the water chemistry. We believe that the decision of whether to use the bounding approach or a K dependent approach depends on knowledge of residual stresses not water chemistry. The use of a K dependent CGR requires that residual stresses be characterized. We do not believe that this can be done adequately without knowledge of the weld repairs.

   As noted previously a bounding CGR approach can be used for both plants on normal water chemistry and those on hydrogen water chemistry. The value of the bounding CGR can be adjusted appropriately. The bounding CGR would be much lower in plants operating on HWC. The staff has already concluded in its SER that the estimates of the benefits of improved water chemistry provided by the BWRVIP-14 correlation are conservative, and hence we believe that it can be used to determine appropriate factors of reduction in the current bounding CGR.

6. Item 6, pg. 5: The BWRVIP proposes a generic stress distribution that includes the welding residual stresses, a conservative estimate of the membrane stresses due to loads on the shroud, and a surface stress distribution intended to represent the effect of local weld repairs.

Assessment

We do not agree with the BWRVIP conclusion that the residual stress distribution given in Fig. 1 of the response and the corresponding K solutions given in Fig. 4 are sufficiently conservative that they can be used to estimate CGRs with no additional knowledge of weld repairs.

The intent of the BWRVIP proposal is to avoid having to document weld repairs. As discussed in the assessment of Item 4 pg.2 of the BWRVIP response, the proposed distribution appears reasonable "on the average." However, it restricts the local stress distributions to be self-equilibrating at every azimuth. Because local weld repairs inherently introduce three dimensional effects, this is an artificial constraint and may lead to nonconservative estimates of the local stresses. In most cases such local effects will not be important for the overall structural integrity of the shroud. In any given case, however, it is difficult to ensure integrity without knowing more about the extent and nature of any weld repairs. Without knowledge of the repairs, we believe it is prudent to use a bounding CGR approach for safety analyses.

95-456

BWRVIP

BWR Vessel &
Internals Project _____ *Issue Management and Resolution*

October 5, 1995

Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: C. E. Carpenter

Subject: "Safety Assessment of BWR Reactor Internals (BWRVIP-06)," EPRI TR-105707, September 1995.

Enclosed are ten (10) copies of the document "Safety Assessment of BWR Reactor Internals (BWRVIP-06)," EPRI TR-105707, October 1995". This document is being submitted as a means of exchanging information with the NRC for the purpose of supporting generic regulatory efforts related to assessing the safety consequences of potential cracking of BWR reactor internals.

The enclosed document evaluates postulated failures of vessel internal components and establishes short-term and long-term actions which are appropriate to assure continued safe operation.

The report has been endorsed by the BWRVIP, however, this endorsement should not be interpreted as a commitment by all members to a specific course of action. Each member must formally endorse the BWRVIP position for it to become that member's position.

Please note that the enclosed document contains proprietary information of both EPRI and GENE. Therefore, letters requesting the report be withheld from public disclosure and affidavits describing the basis for withholding this information are provided as Attachment 1 for the EPRI information and Attachment 2 for the GENE information

Representatives of the BWRVIP would be pleased to meet with the NRC staff to discuss any comments or issues related to the enclosed document. If you have

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Reply To: J. T. Beckham, Jr., BWRVIP Chairman, Southern Nuclear Operating Co., 42 Inverness Center Parkway, Birmingham, AL 35242 o (205) 877-7279 o Fax: (205) 802-0393

any questions on the enclosed document or the general subject it addresses, please call Robin Dyle of Southern Nuclear at (205) 877-7121.

Sincerely,

J. T. Beckham, Jr.
Southern Nuclear Operating Company
Chairman, BWR Vessel & Internals Project

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Attachments

95-454

BWRVIP

BWR Vessel &
Internals Project _____ *Issue Management and Resolution*

September 28, 1995

Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: C. E. Carpenter

Subject: "BWR Vessel and Internals Project, BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations (BWRVIP-05)," EPRI TR-105697, September 1995.

Enclosed are ten (10) copies of the document "BWR Vessel and Internals Project, BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations (BWRVIP-05)," EPRI TR-105697, September 1995. This document is being submitted as a means of exchanging information with the NRC for the purpose of supporting generic regulatory improvements related to BWR reactor pressure vessel shell weld inspection requirements.

The enclosed document evaluates the current inspection requirements for reactor pressure vessel shell welds in BWRs, formulates recommendations for alternative inspection requirements and provides a technically justified basis for these recommendations.

The subject of the enclosed document was reviewed and discussed with various members of the NRC staff at a meeting on July 19, 1995. At that meeting a question arose regarding the possible presence of flaws in a Duane Arnold low-alloy steel vessel nozzle. Following the meeting, this question was investigated and it was determined that flaws found during safe end replacement at Duane Arnold were located in the safe-end and not the low-alloy steel nozzle material. This clarification has also been provided informally to NRC staff members.

The enclosed report has been endorsed by the BWRVIP, however, this endorsement should not be interpreted as a commitment by all members to a specific course of action. Each member must formally endorse the BWRVIP position for it to become that member's position.

Please note that the enclosed document contains proprietary information. Therefore, a letter requesting the report be withheld from public disclosure and an

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Reply To: J. T. Beckham, Jr., BWRVIP Chairman, Southern Nuclear Operating Co., 42 Inverness Center Parkway, Birmingham, AL 35242 • (205) 877-7279 • Fax: (205) 802-0393

F12
2

affidavit describing the basis for withholding this information is provided as Attachment 1.

Representatives of the BWRVIP would be pleased to meet with the NRC staff to discuss any comments or issues related to the enclosed document. If you have any questions on the enclosed document or the general subject it addresses, please call Robin Dyle of Southern Nuclear at (205) 877-7121.

Sincerely,

J. T. Beckham, Jr.
Southern Nuclear Operating Company
Chairman, BWR Vessel & Internals Project

May 20, 1996

Mr. J. T. Beckham, Jr., Chairman
Southern Nuclear Operating Company
BWR Vessel & Internals Project
42 Inverness Center Parkway
Birmingham, Alabama 35242

SUBJECT: BWR OWNERS' GROUP, THE SECOND REQUEST FOR ADDITIONAL INFORMATION FOR TOPICAL REPORT EPRI TR-105697, "BWR VESSEL AND INTERNALS PROJECT, BWR REACTOR PRESSURE VESSEL SHELL WELD INSPECTION RECOMMENDATIONS (BWRVIP-05)"

Dear Mr. Beckham:

By letter dated September 28, 1995, the Boiling Water Reactor Owners' Group (BWROG) submitted Topical Report EPRI TR-105697, "BWR Vessel and Internals Project, BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations (BWRVIP-05)," for NRC review. The report was intended to demonstrate the adequacy of alternative inspection requirements which would significantly reduce the extent of volumetric inservice inspections currently required on BWR reactor pressure vessel shell welds. The staff's position is that the augmented inspection requirements of 10 CFR 50.55a (g)(6)(ii)(A)(3) are still appropriate and that all BWR licensees should perform inspections per this regulation. This position is based on the fact that most earlier BWR vessels did not receive baseline inspections; few BWR vessels have received beltline inservice weld examinations; and previous reliefs granted because of inaccessibility are no longer warranted based on improved inspection technology. Nonetheless, the staff is continuing review of this report in that it may provide additional technical support for granting limited relief under 50.55a when the coverage requirement of "more than 90% of the examination volume of each weld" cannot be fully met due to physical interferences.

The Materials and Chemical Engineering Branch, with assistance from the Electrical, Materials and Mechanical Engineering Branch of the Office of Nuclear Regulatory Research, issued the initial request for additional information (RAI) on April 2, 1996, and is now transmitting the second RAI (Enclosure 1). The RAI referenced Enclosure 2, which is the backfit analysis used to support the NRC final rule that incorporated the augmented reactor vessel examination into 10 CFR 50.55a. In order to complete our assessment on schedule, we request a response to the enclosed RAI within 30 days of receipt. If you have any questions, please contact Keith Wichman of my staff at (301) 415-2757.

Sincerely,
/s/
Brian W. Sheron, Director
Division of Engineering
Office of Nuclear Reactor Regulation

Enclosures: As stated

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96-758

BWRVIP

BWR Vessel &
Internals Project _____ *Issue Management and Resolution*

December 27, 1996

Document Control Desk
U. S. Nuclear Regulatory Commission
11555 Rockville Pike
Rockville, MD 20852

Attention: C. E. Carpenter

Subject: "BWR Vessel and Internals Project, Top Guide Inspection and Flaw Evaluation Guideline (BWRVIP-26)," EPRI Report TR-107285, December, 1996.

Enclosed are ten (10) copies of the document "BWR Vessel and Internals Project, Top Guide Inspection and Flaw Evaluation Guideline (BWRVIP-26)," EPRI Report TR-107285, December, 1996. This document is being submitted as a means of exchanging information with the NRC for the purpose of supporting generic regulatory improvements related to inspection and flaw evaluation of intergranular stress corrosion cracking (IGSCC) in BWR top guide.

The enclosed document contains generic guidelines that describe locations on the top guide for which inspection is needed, the extent of inspection and reinspection for each location, and flaw evaluation procedures (where applicable) to determine allowable flaw sizes. It is the intent that, for BWRVIP member utilities, the enclosed guidelines can be followed in place of prior General Electric Services Information Letters.

The enclosed document has been endorsed by the BWRVIP, however, this endorsement should not be interpreted as a commitment by all members to a specific course of action. Each member must formally endorse the BWRVIP position for it to become that member's position.

Please note that the enclosed document contains proprietary information. Therefore, a letter requesting the report be withheld from public disclosure and an affidavit describing the basis for withholding this information is provided as Attachment 1.

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Reply To: Carl Terry, BWRVIP Vice Chairman, Niagara Mohawk Power Company, P. O. Box 63, Lycoming, NY 13093 • Phone: (315) 349-7263 • Fax: (315) 349-4753

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3

Representatives of the BWRVIP would be pleased to meet with the NRC staff to discuss any comments or issues related to the enclosed document. If you have any questions on the enclosed document or the general subject it addresses, please call Robin Dyle of Southern Nuclear at (205) 877-7121.

Sincerely,

Warren Bilanin for

Carl Terry
Niagara Mohawk Power Company
Vice Chairman, BWR Vessel & Internals Project

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BWRVIP

BWR Vessel & Internals Project — *Issue Management and Resolution*

December 27, 1996

Document Control Desk
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11555 Rockville Pike
Rockville, MD 20852

Attention: C. E. Carpenter

Subject: "BWR Vessel and Internals Project, Top Guide Inspection and Flaw Evaluation Guideline (BWRVIP-26)," EPRI Report TR-107285, December, 1996.

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Please note that the enclosed document contains proprietary information. Therefore, a letter requesting the report be withheld from public disclosure and an affidavit describing the basis for withholding this information is provided as Attachment 1.

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Reply To: Carl Terry, BWRVIP Vice Chairman, Niagara Mohawk Power Company, P. O. Box 63, Lycoming, NY 13093 • Phone: (315) 349-7263 • Fax: (315) 349-4753

Representatives of the BWRVIP would be pleased to meet with the NRC staff to discuss any comments or issues related to the enclosed document. If you have any questions on the enclosed document or the general subject it addresses, please call Robin Dyle of Southern Nuclear at (205) 877-7121.

Sincerely,

Warren Bilanin for

Carl Terry
Niagara Mohawk Power Company
Vice Chairman, BWR Vessel & Internals Project

96-753

BWRVIP

BWR Vessel &
Internals Project _____ *Issue Management and Resolution*

MEMORANDUM

December 23, 1996

Document Control Desk
U. S. Nuclear Regulatory Commission
11555 Rockville Pike
Rockville, MD 20852

Attention: C. E. Carpenter

Subject: "BWR Vessel and Internals Project, Assessment of BWR Jet Pump Riser Elbow to Thermal Sleeve Weld Cracking (BWRVIP-28)," EPRI Report TR-107667, December 1996.

Enclosed are ten (10) copies of the document "BWR Vessel and Internals Project, Assessment of BWR Jet Pump Riser Elbow to Thermal Sleeve Weld Cracking (BWRVIP-28)," EPRI Report TR-107667, December 1996. Also enclosed are ten (10) copies of the non-proprietary version of the report. This document is being submitted as a means of exchanging information with the NRC for the purpose of supporting generic regulatory efforts related to assessing the implications of potential cracking of BWR jet pump riser elbow to thermal sleeve welds.

The enclosed report describes an assessment of the implications of observed cracking in BWR jet pump riser elbow to thermal sleeve welds and identifies operating recommendations for owners of GE BWRs with jet pumps.

The enclosed report is being submitted on behalf of the BWRVIP utilities, however, this submittal should not be interpreted as a commitment by all BWRVIP members to a specific course of action. Each member must formally endorse the BWRVIP position for it to become that member's position.

Please note that the enclosed proprietary document contains proprietary information of both EPRI and GE Nuclear Energy. Therefore, letters requesting the report be withheld from public disclosure and affidavits describing the basis for withholding this information are provided as Attachment 1 for the EPRI information and Attachment 2 for the GE Nuclear Energy information.

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Representatives of the BWRVIP will be meeting with the NRC staff on January 16, 1997 at which time the subject of the enclosed document will be discussed.

If you have any questions on this subject please call Robin Dyle of Southern Nuclear at (205) 992-7121.

Sincerely,

Robin Dyle for Carl Terry

Carl Terry
Niagara Mohawk Power Company
Vice Chairman, BWR Vessel and Internals Project

EPRI
Electric Power
Research Institute

Keywords:
Assessment
Boiling water reactor
Jet pump
Riser
Stress corrosion cracking
Vessel and internals

EPRI TR-107667
December 1996

BWR Vessel and Internals Project

Assessment of BWR Jet Pump Riser Elbow to Thermal Sleeve Weld Cracking (BWRVIP-28)

Prepared by
GE Nuclear Energy

NON-PROPRIETARY INFORMATION

NOTICE: This report contains the non-proprietary information that is included in the proprietary version of this report. The proprietary version of this report contains proprietary information that is the intellectual property of BWRVIP utility members and EPRI. Accordingly, the proprietary report is available only under license from EPRI and may not be reproduced or disclosed, wholly or in part, by any Licensee to any other person or organization.

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BWR Vessel and Internals Project

Assessment of BWR Jet Pump Riser Elbow to Thermal Sleeve Weld Cracking (BWRVIP-28)

TR-107667
Research Project B301

Final Report, December,1996

Prepared by:

GE NUCLEAR ENERGY

Principal Investigator

K. C. Horstman

Prepared for

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DISCLAIMER OF WARRANTIES AND LIMITATION OF LIABILITIES

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Organization(s) that prepared this report:

GE NUCLEAR ENERGY

ORDERING INFORMATION

Requests for copies of this report should be directed to the BWRVIP Program Manager, 3412 Hillview Ave., Palo Alto, Ca. 94304, (415) 855-2340.

TABLE OF CONTENTS

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1. INTRODUCTION

During a recent refueling and maintenance outage of a BWR plant cracks were detected in two jet pump riser pipe elbows located inside of the reactor pressure vessel. The cracks are on the elbow side of the field fabricated circumferential weld joining the riser elbow to the recirculation inlet thermal sleeve. The root cause has not yet been established as the evaluation is still in progress; however, the indications are characteristic of intergranular stress corrosion cracking (IGSCC) in that they show evidence of branching. This represents the first time that indications have been observed in jet pump riser pipe circumferential welds.

The purpose of this report is to document the safety assessment of the implications of the observed cracking and to develop operating recommendations for owners of GE BWRs with jet pumps. The specific analysis contained in this report was performed for the plant in which the cracking was found, referred to as the reference plant; however, the generic applicability of the safety assessment to other BWRs is discussed as well.

2. BACKGROUND

2.1. Configuration

Jet pumps are used as part of the reactor coolant recirculation system in BWR/3-6 plants. They are installed in the annulus between the core shroud and the reactor pressure vessel (RPV), and each vessel contains between 12 and 24 jet pumps. Most BWRs have 20 jet pumps. There are several BWR jet pump designs; however, the basic jet pump configuration is the same for all plants. The jet pump geometry is shown in Figure 1.

Each jet pump assembly consists of a riser assembly, two inlet-mixer assemblies, two diffuser assemblies, and a riser brace. The riser assembly elbow is a 10 inch, 90 degree short radius elbow, fabricated from Type 304 stainless steel. It is welded to the recirculation inlet nozzle thermal sleeve. The riser assembly is supported near the top by the riser brace, which is welded to the riser pipe and to pads on the RPV wall (some plants have two riser braces for each riser). Lateral support is also provided by the jet pump restrainer brackets which attach the adjacent jet pump inlet mixers to the riser. The entrance end of each inlet-mixer assembly is clamped to the top of the riser transition piece by the beam-bolt assembly. The exit end of the inlet-mixer forms a slip joint with the entrance end of the diffuser. This interface between the inlet mixers and the diffusers provides some additional lateral support to the riser. The top of the slip joint is located near the bottom of the fuel; the exact elevation of the slip joint is plant specific. All components of the riser pipes at the reference plant were manufactured by GE and assembled at the site during original plant construction using GE approved procedures.

The jet pump assemblies are non-ASME Code components. They are classified as Safety Related, as the structural integrity of the jet pump assembly is important for assuring core refloodability following a postulated recirculation line break Loss of Coolant Accident (LOCA).

2.2. Inspection Considerations

Through the service life of jet pump assemblies, degradation of jet pump components has occurred which warranted industry notification and recommendations by GE Nuclear Energy (GENE). Although there are no specific requirements to perform inservice inspection of the jet pump riser piping circumferential welds, there are several GENE documents (References 2 through 12) which recommend inspections to verify the integrity of various jet pump assembly components. Jet pump assembly integrity is monitored frequently in accordance with the BWR Owner's inservice inspection programs.

The inspection efforts at plants have continually addressed many internal components. These include the jet pump rams head, the inlet mixers, the jet pump set screws, jet pump riser brace, and the jet pump sensing lines. The in-vessel visual inspection (IVVI) efforts

also include inspection of other areas of the jet pump diffusers and can at times include limited areas of the jet pump riser piping. These inspections vary with different utilities and with each outage. However, the visual inspection of internal components in the annulus region at different elevations provides assurance that significant cracking in the riser piping is unlikely.

Requirements and recommendations for inspection of jet pump assemblies are documented in several industry publications which are listed in the reference section. The examinations recommended in these reference documents are summarized below:

- Periodic volumetric examination of jet pump beams to verify structural integrity of the beam bolt and radius areas.
- Periodic visual examination of jet pump assemblies to verify overall condition and structural integrity.
- Periodic visual examination of jet pump sensing lines and support brackets for integrity of bracket welds.
- Periodic visual examination of the throat area of the jet pump mixers to detect the presence of crud deposits which could inhibit flow.
- Periodic visual examination of jet pump riser brace arm welds (both the reactor vessel and riser pipe connections).
- Periodic visual examination of jet pump gap adjusting screws and the adjusting screw tack welds.

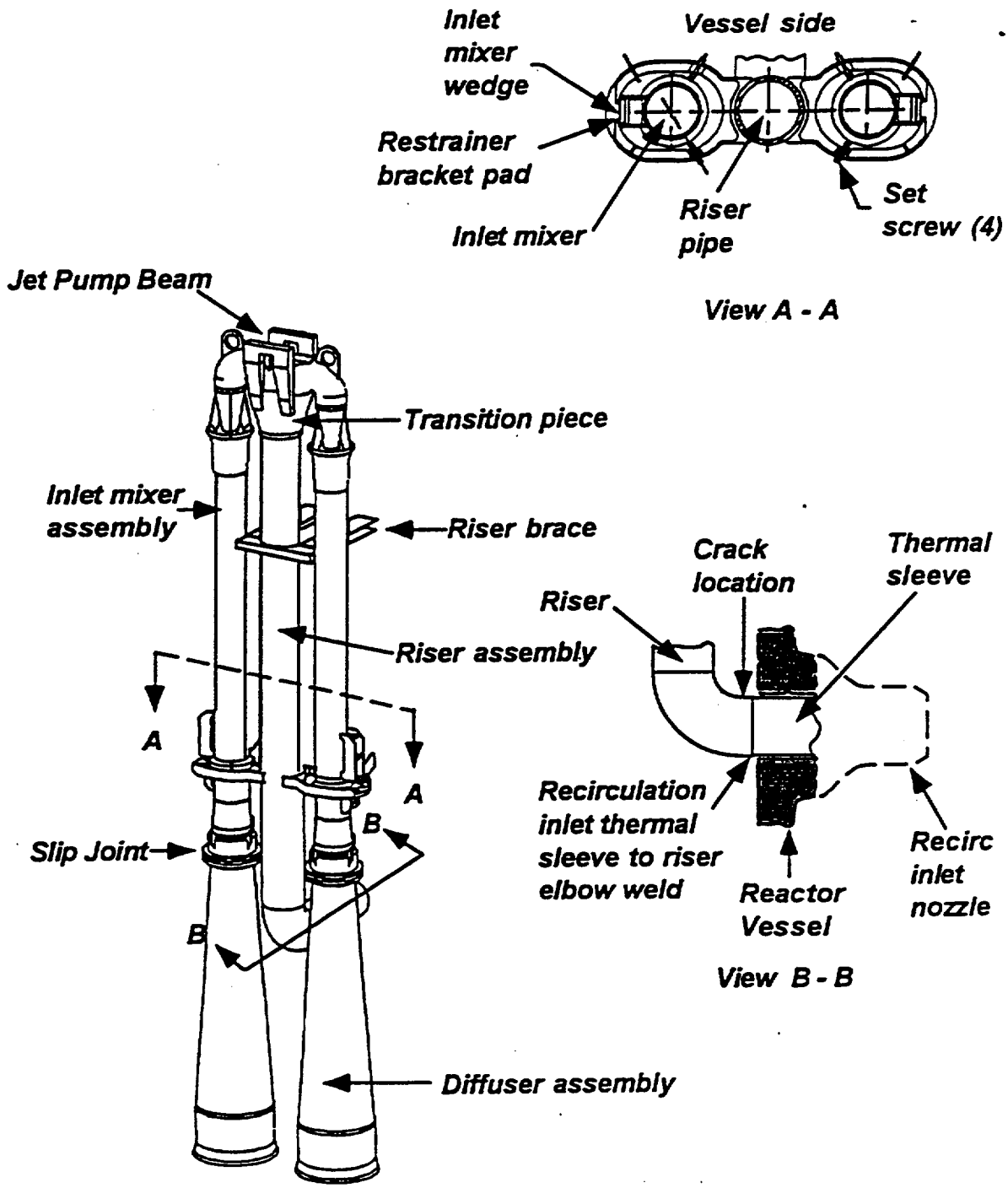

Figure 1. Jet Pump Configuration

PAGES 5 - 19 ARE PROPRIETARY

8. RECOMMENDATIONS

Specific recommendations are as follows:

- The jet pump M ratio should be added to the jet pump surveillance program. Deviation of more than 10% in the jet pump M ratio from its characteristic value can be an indication of jet pump riser degradation.
- If it is determined that performance of a jet pump is significantly degraded, the reactor should be brought to a safe shutdown condition.
- Prior IVVI videotapes from the jet pump region should be reviewed to establish, to the degree possible, the current condition of the riser welds.
- Consideration should be given to inspection of the riser welds at the next normal refueling outage, factoring in plant age, operating history, water chemistry, etc.

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3

ABOUT BWRVIP

The BWR Vessel and Internals Project (BWRVIP) is an association of utilities owning and operating boiling water reactors. The project is focused exclusively on reactor vessel and vessel internals issues in operating plants. Objectives of the BWRVIP are to lead the BWR industry toward generic resolution of vessel and internals integrity and operability issues; to identify or develop generic, cost-effective strategies from which each operating plant will select the most appropriate alternative; to serve as the focal point for the regulatory interface with the industry on BWR vessel and internals integrity and operability issues; and to share information among members. EPRI manages the technical program on behalf of the utility members of the BWRVIP.

ABOUT EPRI

Electricity is increasingly recognized as a key to societal progress throughout the world, driving economic prosperity and improving the quality of life. The Electric Power Research Institute delivers the science and technology to make the generation, delivery, and use of electricity affordable, efficient, and environmentally sound.

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BWR Vessel &
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MEMORANDUM

December 20, 1996

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11555 Rockville Pike
Rockville, MD 20852

Attention: C. E. Carpenter

Subject: BWRVIP Response to NRC Request for Additional Information on BWRVIP-06

Reference: Letter from C. E. Carpenter (NRC) to J. T. Beckham, Jr. (BWRVIP Chairman), "Request for Additional Information - Review of BWR Vessel and Internals Project Proprietary Report, BWR Vessel and Internals Project, Safety Assessment of BWR Reactor Internals (BWRVIP-06)," dated May 21, 1996.

Enclosed are 10 copies of the BWRVIP response to the NRC Request for Additional Information transmitted by the NRC letter referenced above. As indicated in that NRC letter, since the enclosed information concerns a report that the NRC staff has found to be proprietary in nature, the enclosed information is also proprietary and should be withheld from public disclosure.

If you have any questions on this subject please contact the undersigned at (205) 992-7121.

Sincerely,

Robin Dyle
Technical Chairman
BWRVIP Assessment Committee
Southern Nuclear Operating Company

RD-10-1

G 4

Information in this record was deleted
in accordance with the Freedom of Information
Act, exemptions 4
FOIA- 2000-0186

c: Carl Terry, BWRVIP Vice Chairman, Niagara Mohawk Power Co.

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BWRVIP

BWR Vessel &
Internals Project _____ *Issue Management and Resolution*

November 27, 1996

Document Control Desk
U. S. Nuclear Regulatory Commission
11555 Rockville Pike
Rockville, MD 20852

Attention: C. E. Carpenter

Subject: "BWR Vessel and Internals Project, Roll/Expansion Repair of Control Rod Drive and In-Core Instrument Penetrations in BWR Vessels (BWRVIP-17)," EPRI Report TR 106712, November, 1996.

Enclosed are ten (10) copies of the document "BWR Vessel and Internals Project, Roll/Expansion Repair of Control Rod Drive and In-Core Instrument Penetrations in BWR Vessels (BWRVIP-17)," EPRI Report TR 106712, November,1996. Also enclosed are ten (10) copies of the non-proprietary version of the report. This document is being submitted as a means of exchanging information with the NRC for the purpose of supporting generic regulatory improvements related to repair of the subject reactor components.

The enclosed document contains a description of the Roll/Expansion process for repair of leaking control rod drive and in-core instrument penetrations, includes a discussion of the technical basis and requirements for a successful repair, and identifies plant-specific requirements for repair implementation. It is intended that, when approved by the NRC, the report could be used by BWRVIP member utilities as a guideline for designing and implementing the repair.

The enclosed document has been endorsed by the BWRVIP, however, this endorsement should not be interpreted as a commitment by all members to a specific course of action. Each member must formally endorse the BWRVIP position for it to become that member's position.

Please note that the enclosed document contains proprietary information. Therefore, a letter requesting the report be withheld from public disclosure and an affidavit describing the basis for withholding this information is provided as Attachment 1.

C/A: PDR LTR ENCL
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RD-10-1

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F/7

Reply To: J. T. Beckham, Jr., BWRVIP Chairman, Southern Nuclear Operating Co., 42 Inverness

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PDR TOPRP EXIEPRI
C PDR

Information in this record was deleted in accordance with the Freedom of Information Act, exemptions 4
FOIA- 2000-0896

Representatives of the BWRVIP would be pleased to meet with the NRC staff to discuss any comments or issues related to the enclosed document. If you have any questions on the enclosed document or the general subject it addresses, please call Bruce McLeod, BWRVIP Repair Committee Technical Chairman, of Southern Nuclear, at (205) 992-7446.

Sincerely,

J. T. Beckham, Jr.
Southern Nuclear Operating Company
Chairman, BWR Vessel & Internals Project

EPRI
Electric Power
Research Institute *Powering Progress through Innovative Solutions*

November 27, 1996

Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Subject: Request for Withholding of Proprietary Document; 10CFR2.790(a)(4) TR-106712, "BWR Vessel and Internals Project, Roll/Expansion Repair of Control Rod Drive and In-Core Instrument Penetrations in BWR Vessels (BWRVIP-17)," EPRI Report, November, 1996

Gentlemen:

This a request under 10CFR2.790(a)(4) that the NRC withhold from public disclosure the proprietary document identified above (the "Report"). Ten copies of the Report and the affidavit in support of this request are enclosed.

EPRI desires to disclose the Report to the NRC as a means of exchanging information with the NRC for the purpose of supporting generic regulatory improvements related to repair of the subject reactor components. EPRI would welcome any discussions between EPRI and the NRC related to the Report that the NRC desires to conduct.

The Report is for the NRC's internal use and may be used only for the purpose for which it is disclosed by EPRI. The Report should not be otherwise used or disclosed to any person outside the NRC without prior written permission from EPRI.

If you have any questions about the legal aspects of this request for withholding, please do not hesitate to contact me at (415) 855-2845. Questions on the contents of the Report should be directed to Warren Bilanin of EPRI at (415) 855-2340.

Sincerely,

Art Kenny

Arthur Kenny
Intellectual Property Attorney
Intellectual Property Department

Enclosures

c: W. Bilanin
K. Wolfe

9612020132 961127
PDR TOPRP EXIEPRI
C PDR

Headquarters: 3412 Hillview Avenue, Post Office Box 10412, Palo Alto, CA 94303, USA • (415) 855-2000 • www.epri.com
Washington Office: 2000 L Street, NW, Suite 805, Washington, DC 20036, USA • (202) 872-9222 • Fax: (202) 293-2697

RE: TR-106712, "BWR Vessel and Internals Project Roll/Expansion Repair of Control Rod Drive and In-Core Instrument Penetrations in BWR Vessels (BWRVIP-17)," EPRI Report, November, 1996

I. ARTHUR KENNY, being duly sworn, depose and state as follows:

I am an attorney at the Electric Power Research Institute ("EPRI") and I have been specifically delegated responsibility for reviewing the report listed above that is sought under this affidavit to be withheld (the "Report") and authorized to apply for its withholding on behalf of EPRI. This affidavit is submitted to the Nuclear Regulatory Commission ("NRC") pursuant to 10 CFR 2.790(a)(4) based on the fact that the Report consists of trade secrets of EPRI and that the NRC will receive the Report from EPRI under privilege and in confidence.

The basis for which the Report should be withheld from the public is set forth below:

(i) The Report has been held in confidence by EPRI, its owner. All those accepting the Report must agree to preserve the confidentiality of the Report.

(ii) The Report is of a type customarily held in confidence by EPRI and there is a rational basis therefor. The Report is trade secrets and is held in confidence by EPRI because to disclose it would prevent EPRI from licensing the Report at fees which would allow EPRI to recover its investment. If consultants and other businesses providing services in the nuclear power industry were able to publicly obtain the Report, they would be able to use it commercially for profit and avoid spending the large amount of money that EPRI was required to spend to prepare the Report. The rational basis that EPRI has for classifying the Report as trade secrets is the Uniform Trade Secrets Act which California adopted in 1984 and which has been adopted by over twenty states. The Uniform Trade Secrets Act defines a "trade secret" as follows:

> "Trade secret" means information, including a formula, pattern, compilation, program, device, method, technique, or process, that:
>
> (1) Derives independent economic value, actual or potential, from not being generally known to the public or to other persons who can obtain economic value from its disclosure or use; and

(2) Is the subject of efforts that are reasonable under the circumstances to maintain its secrecy.

(iii) The Report will be transmitted to the NRC in confidence.

(iv) The Report is not available in public sources. EPRI developed the Report only after making a determination that the Report was not available from public sources. It required a large expenditure of dollars for EPRI to develop the Report. In addition, EPRI was required to use a large amount of time of EPRI employees. The money spent, plus the value of EPRI's staff time in preparing the Report, show that the Report is highly valuable to EPRI. Finally, the Report was developed only after a long period of effort of at least several months.

(v) A public disclosure of the Report would cause substantial harm to EPRI's competitive position and the ability of EPRI to license the Report both domestically and internationally. The Report can be properly acquired or duplicated by others only with an equivalent investment of time and effort.

I have read the foregoing and the matters stated therein are true and correct to the best of my knowledge, information and belief. I make this affidavit under penalty of perjury under the laws of the United States of America and under the laws of the State of California.

Executed at 3412 Hillview Avenue, Palo Alto, being the premises and place of business of the Electric Power Research Institute:

November 27, 1996

Art Kenny

Arthur Kenny

TAMSEN HELEN GAGNON
COMM. # 1016243
Notary Public — California
SANTA CLARA COUNTY
My Comm. Expires FEB 5, 1998

Subscribed and sworn before me this day: November 27, 1996

Tamsen Helen Gagnon, Notary Public

EPRI
Electric Power
Research Institute

Keywords:
Boiling water reactor
Control rod drive
In-core housing
Repair
Stress corrosion cracking
Vessel and internals

EPRI TR-106712
November 1996

BWR Vessel and Internals Project

Roll/Expansion Repair of Control Rod Drive and In-Core Instrument Penetrations in BWR vessels (BWRVIP-17)

Prepared by
BWRVIP Repair Committee
and
MPR Associates

NON-PROPRIETARY VERSION

This report contains the non-proprietary information that is included in the proprietary version of this report. The proprietary version of this report contains proprietary information that is the intellectual property of BWRVIP utility members and EPRI. Accordingly, the proprietary report is available only under license from EPRI and may not be reproduced or disclosed, wholly or in part, by any Licensee to any other person or organization.

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PDR TOPRP EXIEPRI
C PDR

REPORT SUMMARY

BWRVIP Vessel and Internals Project

Roll/Expansion Repair of Control Rod Drive and In-Core Instrument Penetrations in BWR Vessels (BWRVIP-17)

The Boiling Water Reactor Vessel and Internals Project (BWRVIP), formed in June, 1994, is an association of utilities focused exclusively on BWR vessel and internals issues. This BWRVIP report describes a roll/expansion method for repairing leaks which have been known to occur at control rod drive (CRD) and instrument penetrations in BWR vessels. The report includes a description of the roll/expansion repair process, the technical basis and requirements for performing a successful repair and identification of plant-specific actions required for repair implementation.

INTEREST CATEGORIES

Piping, reactor, vessel & internals
Licensing and safety assessment

KEYWORDS

Boiling water reactor
Control rod drive
In-core housing
Repair
Stress corrosion cracking
Vessel and internals

BACKGROUND CRD and in-core instrument penetrations are located in the bottom head of BWR vessels. A number of operating plants have experienced leakage at these penetrations due to stress corrosion cracking near fabrication welds. Utilities require a repair method which eliminates or reduces this leakage to acceptable limits. One method which has been used in the past is to expand the CRD or in-core housing against the vessel wall by use of a rolling technique. Use of this method has required extensive analysis and testing on the part of the utility in order to obtain NRC acceptance of the approach.

OBJECTIVE To obtain regulatory approval of the use of the roll/expansion method for repair of leaking CRD and in-core instrument penetrations and to define plant-specific design/implementation criteria.

APPROACH The project team reviewed relevant utility experience with roll/expansion repair of penetrations. Based on this information and the results of a detailed technical qualification performed by one domestic utility, the committee developed a list of technical issues that needed to be addressed in applying the process. For each of these issues, either a generic resolution was developed or plant-specific actions that must be performed were identified.

RESULTS Evaluation of each technical issue leads to the conclusion that use of the roll/expansion technique is a viable approach for repair of leaking penetrations. Many of the technical issues are resolved generically in the report such that a utility would not be required to provide additional justification. A second class of issues relates to specific features of the CRD or in-core housing such as material type or physical dimensions. The report shows that, if a plant's configuration is sufficiently similar to the plant for which detailed technical qualification was performed, the results of that prior analysis are sufficient justification to resolve these issues. Finally, a third class of issue exists which will typically need to be addressed on a plant-specific basis. The number of these plant-specific issues, however, has been minimized.

The report includes a detailed, standardized process that a BWR owner should follow to support the use of the roll-repair method in his plant.

EPRI PERSPECTIVE The roll/expansion repair method provides an effective means of leakage control which requires relatively short implementation time, low tooling and implementation costs and a significant radiation dose saving over other methods. NRC acceptance of the approach described in the report will significantly reduce the utility effort required to obtain regulatory approval of plant-specific repairs.

PROJECT
WOB501
EPRI Project Manager: Warren Bilanin
Nuclear Power Group
Contractor: MPR Associates

For ordering information about this report, call the EPRI Program Manager at (415) 855-2340.

For membership information, call (415) 855-2514.

BWR Vessel and Internals Project

Roll/Expansion Repair of Control Rod Drive and In-Core Instrument Penetrations in BWR Vessels (BWRVIP-17)

TR-106712
Research Project B501

Final Report, November, 1996

Prepared by:

MPR Associates
320 King Street
Alexandria, VA 22314

BWRVIP Repair Committee

Prepared for

BOILING WATER REACTOR VESSEL & INTERNALS PROJECT
and
ELECTRIC POWER RESEARCH INSTITUTE
3412 Hillview Ave.
Palo Alto, California 94304

DISCLAIMER OF WARRANTIES AND LIMITATION OF LIABILITIES

This report was prepared by the organization(s) named below as an account of work sponsored or cosponsored by the BWR Vessel and Internals Project (BWRVIP) and the Electric Power Research Institute, Inc. (EPRI). Neither BWRVIP, EPRI, any member of EPRI, any cosponsor, the organization(s) named below, nor any person acting on behalf of any of them:

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(b) assumes any responsibility for any damages or other liability whatsoever (including any consequential damages, even if BWRVIP, EPRI or any EPRI representative has been advised of the possibility of such damages) resulting from your selection or use of this report or any information, apparatus, method, process or similar item disclosed in this report.

Organization(s) that prepared this report:

MPR ASSOCIATES, INC.

BWRVIP REPAIR COMMITTEE

ORDERING INFORMATION

Requests for copies of this report should be directed to the BWRVIP Program Manager, 3412 Hillview Ave., Palo Alto, Ca. 94304, (415) 855-2240.

ABSTRACT

This report describes a roll/expansion technique for repairing leaking CRD and in-core housing penetrations. The report includes a discussion of the technical basis for the method, detailed procedures for its implementation, and utility experience with the roll/expanion technique.

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EXECUTIVE SUMMARY

In some domestic and foreign BWR plants, control rod drive (CRD) and in-core instrument penetrations have leaked due to cracking of the CRD penetration stub tube or the in-core housing to vessel weld. In these plants, a roll/expansion repair was used to mitigate or eliminate leakage from the penetration. The advantages of this repair include its effectiveness in mitigating leakage as well as significant time, dose and cost savings compared to other repair methods such as weld repair or replacement of the penetration components.

This report provides the technical basis for acceptance of the roll/expansion repair process as a suitable permanent repair option. It includes:

- A description of the roll/expansion repair process
- Technical basis and requirements for a successful repair
- Identification of the plant-specific requirements for repair implementation

It is intended that this report will be submitted to industry regulators for the purpose of obtaining approval of:

- Roll/expansion as a repair option
- Generic qualification that has been performed
- Plant specific design/implementation criteria

96-417

BWRVIP

BWR Vessel &
Internals Project _____ *Issue Management and Resolution*

August 12, 1996

Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Attention: C. E. Carpenter

Subject: Non Proprietary version of BWR Vessel and Internals Project, BWR Core Spray Internals Inspection and Flaw Evaluation Guidelines (BWRVIP-18), EPRI Report TR-106740, July 1996

Enclosed are ten (10) copies of the non-proprietary version of the above referenced document. You are already in possession of the proprietary version of this document which was mailed on July 26, 1996. This document is being submitted as means of exchanging information with the NRC for the purpose of supporting generic regulatory improvements related to the BWR reactor pressure vessel and internals examinations.

The enclosed report has been endorsed by the BWRVIP, however, this endorsement should not be interpreted as commitment by all members to a specific course of action. Each member must formally endorse the BWRVIP position for it to become the member's position.

Representatives of the BWRVIP would be pleased to meet with the NRC staff to discuss any comments or issues related to the enclosed documents. If you have any questions on the enclosed documents, please call Madelaine Campbell, of EPRI at 415-855-2879.

Sincerely,

J.T. Beckman, Jr.
Southern Nuclear Operating Company
Chairman, BWR Vessel & Internals Project

DF12 1/1 Add: Timm - Paper Misc Corr

Enclosure

Information in this record was deleted in accordance with the Freedom of Information Act, exemptions 4
FOIA- 2000-0186

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Electric Power
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Keywords:
Boiling water reactor
Core spray internals
Flaw evaluation
Inspection strategy
Stress corrosion cracking
Vessel and internals

EPRI TR-106740
July 1996

BWR Vessel and Internals Project

BWR Core Spray Internals Inspection and Flaw Evaluation Guidelines (BWRVIP-18)

Prepared by
BWR Vessel and Internals Project Assessment Committee
Core Spray Inspection and Evaluation Focus Group

GE Nuclear Energy

NON-PROPRIETARY VERSION

This report contains the non-proprietary information that is included in the proprietary version of this report. The proprietary version of this report contains proprietary information that is the intellectual property of BWRVIP utility members and EPRI. Accordingly, the proprietary report is available only under license from EPRI and may not be reproduced or disclosed, wholly or in part, by any Licensee to any other person or organization.

REPORT SUMMARY

BWR Vessel and Internals Project

BWR Core Spray Internals Inspection and Flaw Evaluation Guidelines (BWRVIP-18)

The Boiling Water Reactor Vessel and Internals Project (BWRVIP), formed in June 1994, is an association of utilities focused exclusively on BWR vessel and internals issues. This report, eighteenth in a continuing series, contains generic guidelines that describe locations on the core spray piping and spargers for which inspection is needed, categories of plants for which inspection need would differ, extent of inspection and reinspection for each location, and flaw evaluation procedures to determine allowable flaw sizes for each location or type of location.

INTEREST CATEGORIES

Piping, reactor, vessel & internals
Licensing and safety assessment

KEYWORDS

Boiling water reactor
Core spray internals
Flaw evaluation
Inspection strategy
Stress corrosion cracking
Vessel and internals

BACKGROUND Events in 1993 and 1994 confirmed that intergranular stress corrosion cracking (IGSCC) is a significant issue for BWR internals. US BWR executives formed the BWRVIP in June, 1994, to address integrity issues arising from service-related degradation of these key components, beginning with core shroud cracking.

Recently, the BWRVIP prepared a safety assessment of BWR internals as a follow-on to the activities completed on shroud cracking. Based on the consequences of core spray cracking, it was determined that inspection of the core spray piping and spargers is necessary to assure long term integrity, and thus the ability to achieve safe shutdown for worst case scenarios. As a result, the BWRVIP made development of the core spray inspection and evaluation guidelines a high priority for 1996.

OBJECTIVE To provide generic inspection and flaw evaluation (I&E) guidelines for BWR core spray internals .

APPROACH The Assessment Committee of the BWRVIP formed a Focus Group to develop core spray internals I&E guidelines. The group, comprised of utility and industry experts, reviewed available information including IGSCC experience to develop generic I&E guidelines. This information was used to identify the core spray internals locations susceptible to IGSCC and to develop approaches for inspection, reinspection and flaw evaluation.

RESULTS The I&E guidelines provide information on weld locations in core spray piping, spargers and brackets, a discussion of susceptibility considerations, a "baseline" approach for the first inspection each plant will do to new BWRVIP requirements for the core spray components and an approach for subsequent reinspections.

The guidelines also contain a flaw evaluation methodology that provides guidance on performing flaw evaluations and determining loading and stresses. Loading combinations are recommended for plants that do not already have such information. Methodology is provided to use stress information calculated from finite element analyses of the core spray system under these loading combinations and to perform limit load flaw evaluation at each weld. Example evaluations are provided for piping and sparger locations.

EPRI PERSPECTIVE It is the intent that, for BWRVIP members, these guidelines should be followed in the place of prior GE Services Information Letters (SILs) and, when approved by the regulator, in the place of the requirements of NRC IE Bulletin 80-13. The BWRVIP Assessment Committee plans to monitor the results of all core spray inspections closely so that new information obtained from these inspections can be factored into subsequent revisions of this document.

PROJECT
WOB301
EPRI Project Manager: Robert G. Carter
Nuclear Power Group
Contractor: GE Nuclear Energy

For ordering information about this report, call the EPRI Program Manager at (415) 855-2340.

For membership information, call (415) 855-2514.

96-388

BWRVIP

BWR Vessel &
Internals Project _____ *Issue Management and Resolution*

July 26, 1996

Document Control Desk
U. S. Nuclear Regulatory Commission
11555 Rockville Pike
Rockville, MD 20852

Attention: C. E. Carpenter

Subject: "BWR Vessel and Internals Project, BWR Core Spray Internals Inspection and Flaw Evaluation Guidelines (BWRVIP-18)," EPRI Report TR-106740, July 1996.

Enclosed are ten (10) copies of the document "BWR Vessel and Internals Project, BWR Core Spray Internals Inspection and Flaw Evaluation Guidelines (BWRVIP-18)," EPRI Report TR-106740, July 1996. This document is being submitted as a means of exchanging information with the NRC for the purpose of supporting generic regulatory improvements related to inspection and flaw evaluation of intergranular stress corrosion cracking (IGSCC) in BWR core spray internals.

The enclosed document contains generic guidelines that describe locations on the core spray internals for which inspection is needed, categories of plants for which inspection need would differ, the extent of inspection and reinspection for each location, and flaw evaluation procedures to determine allowable flaw sizes. It is the intent that, for BWRVIP member utilities, the enclosed guidelines can be followed in place of prior General Electric Services Information Letters (SILs) and, when approved by the NRC, in place of the requirements of IE Bulletin 80-13.

The enclosed document has been endorsed by the BWRVIP, however, this endorsement should not be interpreted as a commitment by all members to a specific course of action. Each member must formally endorse the BWRVIP position for it to become that member's position.

6004 10

Please note that the enclosed document contains proprietary information. Therefore, a letter requesting the report be withheld from public disclosure and an affidavit describing the basis for withholding this information is provided as Attachment 1.

NDAC NF

9607300115 960726
PDR TOPRP EXIEPRI
C PDR
Information in this record was deleted
in accordance with the Freedom of Information
Act, exemptions 4
FOIA- 2000-0186

Reply to: U.T. Bechtel Jr., BWRVIP Chairman, Southern Nuclear Operating Co., 42 Inverness Center Parkway, Birmingham, AL 35242 • (205) 877-7279 • Fax: (205) 802-0393

F19 9

Representatives of the BWRVIP would be pleased to meet with the NRC staff to discuss any comments or issues related to the enclosed document. If you have any questions on the enclosed document or the general subject it addresses, please call Robin Dyle of Southern Nuclear at (205) 877-7121.

Sincerely,

J. T. Beckham, Jr.
Southern Nuclear Operating Company
Chairman, BWR Vessel & Internals Project

Attachment 1

EPRI
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July 24, 1996

Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Subject: Request for Withholding of Proprietary Document; 10CFR2.790(a)(4) TR-106740, "BWR Vessel and Internals Project, BWR Core Spray Internals Inspection and Flaw Evaluation Guidelines (BWRVIP-18)," EPRI Report, July 1996

Gentlemen:

This a request under 10CFR2.790(a)(4) that the NRC withhold from public disclosure the proprietary document identified above (the "Report"). Ten copies of the Report and the affidavit in support of this request are enclosed.

EPRI desires to disclose the Report to the NRC as a means of exchanging information with the NRC for the purpose of supporting generic regulatory improvements related to inspection and flaw evaluation of intergranular stress corrosion cracking (IGSCC) in BWR core spray internals. EPRI would welcome any discussions between EPRI and the NRC related to the Report that the NRC desires to conduct.

The Report is for the NRC's internal use and may be used only for the purpose for which it is disclosed by EPRI. The Report should not be otherwise used or disclosed to any person outside the NRC without prior written permission from EPRI.

9607300128 960726
PDR TOPRP EXIEPRI
C PDR

Headquarters: 3412 Hillview Avenue, Post Office Box 10412, Palo Alto, CA 94303, USA • (415)855-2000 • Telex: 82977 EPRI UF • Fax: (415) 855-1026
Washington Office: 2000 L Street, NW, Suite 805, Washington, DC 20036, USA • (202) 872-9222 • Fax: (202) 296-6040

Headquarters: 3412 Hillview Avenue, Post Office Box 10412, Palo Alto, CA 94303, USA • (415) 855-2000
Washington Office: 2000 L Street, NW, Suite 805, Washington, DC 20036, USA • (202) 872-9222 • Fax: (202) 293-2697

If you have any questions about the legal aspects of this request for withholding, please do not hesitate to contact me at (415) 855-2845. Questions on the contents of the Report should be directed to Warren Bilanin of EPRI at (415) 855-2340.

Sincerely,

Arthur Kenny
Intellectual Property Attorney
Intellectual Property Department

cc: W. Bilanin

Enclosures

AFFIDAVIT

I, ARTHUR KENNY, being duly sworn, depose and state as follows:

I am an attorney at the Electric Power Research Institute ("EPRI") and I have been specifically delegated responsibility for reviewing the report listed above that is sought under this affidavit to be withheld (the "Report") and authorized to apply for its withholding on behalf of EPRI. This affidavit is submitted to the Nuclear Regulatory Commission ("NRC") pursuant to 10 CFR 2.790(a)(4) based on the fact that the Report consists of trade secrets of EPRI and that the NRC will receive the Report from EPRI under privilege and in confidence.

The basis for which the Report should be withheld from the public is set forth below:

(i) The Report has been held in confidence by EPRI, its owner. All those accepting the Report must agree to preserve the confidentiality of the Report.

(ii) The Report is of a type customarily held in confidence by EPRI and there is a rational basis therefor. The Report is trade secrets and is held in confidence by EPRI because to disclose it would prevent EPRI from licensing the Report at fees which would allow EPRI to recover its investment. If consultants and other businesses providing services in the nuclear power industry were able to publicly obtain the Report, they would be able to use it commercially for profit and avoid spending the large amount of money that EPRI was required to spend to prepare the Report. The rational basis that EPRI has for classifying the Report as trade secrets is the Uniform Trade Secrets Act which California adopted in 1984 and which has been adopted by over twenty states. The Uniform Trade Secrets Act defines a "trade secret" as follows:

> "Trade secret" means information, including a formula, pattern, compilation, program, device, method, technique, or process, that:
>
> (1) Derives independent economic value, actual or potential, from not being generally known to the public or to other persons who can obtain economic value from its disclosure or use; and
>
> (2) Is the subject of efforts that are reasonable under the circumstances to maintain its secrecy.

(iii) The Report will be transmitted to the NRC in confidence.

(iv) The Report is not available in public sources. EPRI developed the Report only after making a determination that the Report was not available from public sources. It required a large expenditure of dollars for EPRI to develop the Report. In addition, EPRI was required to use a large amount of time of EPRI employees. The money spent, plus the value of EPRI's staff time in preparing the Report, show that the Report is highly valuable to EPRI. Finally, the Report was developed only after a long period of effort of at least several months.

(v) A public disclosure of the Report would cause substantial harm to EPRI's competitive position and the ability of EPRI to license the Report both domestically and internationally. The Report can be properly acquired or duplicated by others only with an equivalent investment of time and effort.

I have read the foregoing and the matters stated therein are true and correct to the best of my knowledge, information and belief. I make this affidavit under penalty of perjury under the laws of the United States of America and under the laws of the State of California.

Executed at 3412 Hillview Avenue, Palo Alto, being the premises and place of business of the Electric Power Research Institute:

July 24, 1996

Arthur Kenny

Subscribed and sworn before me this day: July 24, 1996

Tamsen Helen Gagnon Notary Public

TAMSEN HELEN GAGNON
COMM. # 1016243
Notary Public — California
SANTA CLARA COUNTY
My Comm. Expires FEB 5, 1998

96-329

BWRVIP

BWR Vessel &
Internals Project _____ *Issue Management and Resolution*

MEMORANDUM

June 24, 1996

Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Attention: Brian W. Sheron

Subject: BWRVIP Response to NRC Requests for Additional Information on BWRVIP-05

References:
1.) Letter from Brian W. Sheron (NRC) to J. T. Beckham, Jr. (BWRVIP Chairman), "BWR Owners' Group, Request for Additional Information for Topical Report EPRI TR-105697, BWR Vessel and Internals Project, BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations (BWRVIP-05)," dated April 2, 1996.
2.) Letter from Brian W. Sheron (NRC) to J. T. Beckham, Jr., (BWRVIP Chairman), "BWR Owners' Group, The Second Request for Additional Information for Topical Report EPRI TR-105697, BWR Vessel and Internals Project, BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations (BWRVIP-05)," dated May 20, 1996.
3.) Letter from J. T. Beckham (BWRVIP Chairman) to NRC Document Control Desk, "BWR Vessel and Internals Project, BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations (BWRVIP-05), EPRI TR-105697, September 1995," dated September 28, 1995.

Enclosed is one copy of the BWRVIP response to the NRC Requests for Additional Information (RAIs) identified in References 1 and 2 above. The BWRVIP previously committed to provide this response to you on this date.

As you are aware, a meeting has been scheduled for July 11, 1996 at the NRC offices in Rockville, MD to discuss the BWRVIP report that is the subject of the RAIs referenced above. We look forward to a productive and successful interchange on July 11.

Please note that the enclosed material concerns a BWRVIP report that contains proprietary information. Thus, the enclosed material also contains proprietary information. Therefore, the request for withholding the BWRVIP report from

Information in this record was deleted in accordance with the Freedom of Information Act, exemptions 4
FOIA- 2000-0186
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public disclosure transmitted to the NRC by Reference 3 identified above also applies to the enclosed information.

If you have any questions on this subject please contact the undersigned at (205) 877-7121.

Sincerely,

Warren Bilanin for

Robin Dyle
Technical Chairman
BWRVIP Assessment Committee
Southern Nuclear Operating Company

c: Keith Wichman, NRC
C. E. Carpenter, NRC
Carl Terry, Executive Chairman, BWRVIP Assessment Committee, NMPC

POWERING PROGRESS THROUGH
SCIENCE AND TECHNOLOGY

EPRI

BWRVIP BWR Vessel & Internals Project 99-503

December 6, 1999

Document Control Desk
Nuclear Regulatory Commission
11555 Rockville Pike
Rockville, MD 20852-2738

Attention: C.E. Carpenter

Subject: BWRVIP Response to NRC Request for Additional Information on BWRVIP-55 (Project 704)

Reference 1: Letter from C.E. Carpenter to Carl Terry dated 8/13/99, "Proprietary Request for Additional Information - Review of EPRI Topical Reports TR-10872, TR-108719, and TR-108721 (TAC NOs. MA33673 and MA4464)

Enclosed are 10 copies of the BWRVIP response to the NRC Request for Additional Information (RAI) on the BWRVIP report "BWR Vessel and Internals Project, Lower Plenum Repair Design Criteria (BWRVIP-55)." The RAI was transmitted by the NRC to the BWRVIP in the Reference 1 letter. The enclosed response to the RAI repeats each of the individual issues in the NRC RAI, followed by the BWRVIP response to that issue.

As indicated in the referenced letter, the NRC staff has found the subject report to be proprietary in nature. Therefore, the enclosed information is also proprietary and should be withheld from public disclosure.

If you have any questions regarding this subject, please contact Bruce McLeod of Southern Nuclear (BWRVIP Repair Committee Technical Chairman) at (205) 992-7446.

Sincerely,

Tom J. Mulford for
Vaughn Wagoner
Chairman, BWRVIP Integration Committee

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in accordance with the Freedom of Information
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CORPORATE HEADQUARTERS
3412 Hillview Avenue | Palo Alto CA 94304-1395 USA | 650.855.2000 | Customer Service 800.313.3774 | www.epri.com

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96-191

BWRVIP

BWR Vessel &
Internals Project _____ *Issue Management and Resolution*

April 16, 1996

Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Attention: C. E. Carpenter

Subject: "BWR Vessel and Internals Project, Reactor Pressure Vessel and Internals Examination Guidelines (BWRVIP-03), Section 6A, Standards for Visual Inspection of Core Spray Piping, Spargers, and Associated Components," EPRI Report TR-105696, February 1996.

Reference: "BWR Vessel and Internals Project, Reactor Pressure Vessel and Internals Examination Guidelines (BWRVIP-03), EPRI Report TR-105696, October 1995," letter from J. T. Beckham, Jr. (Southern Nuclear Operating Company) to Document Control Desk (NRC) dated November 10,1996.

Enclosed are ten (10) copies of the document "BWR Vessel and Internals Project, Reactor Pressure Vessel and Internals Examination Guidelines (BWRVIP-03), Section 6A, Standards for Visual Inspection of Core Spray Piping, Spargers, and Associated Components," EPRI Report TR-105696, February 1996. This document is being submitted as a means of exchanging information with the NRC for the purpose of supporting generic regulatory improvements related to BWR reactor pressure vessel and internals examinations.

The enclosed document provides guidance regarding visual examinations of BWR core spray piping, spargers and associated components inside the reactor vessel. Please note that the enclosed document should be inserted in the three-ring binder transmitted to you by the letter identified in the reference above. The enclosed document should be inserted behind the tab labeled "Core Spray Piping."

The enclosed document has been endorsed by the BWRVIP, however, this endorsement should not be interpreted as a commitment by all members to a specific course of action. Each member must formally endorse the BWRVIP position for it to become that member's position.

Please note that the enclosed document contains proprietary information. Therefore, a letter requesting the report be withheld from public disclosure and

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in accordance with the Freedom of Information
Act, exemptions 4
FOIA- 2000-0186

Reply To: J. T. Beckham, Jr., BWRVIP Chairman, Southern Nuclear Operating Co., 42 Inverness Center Parkway, Birmingham, AL 35242 • (205) 877-7279 • Fax: (205) 802-0393

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an affidavit describing the basis for withholding this information is provided as Attachment 1.

Representatives of the BWRVIP would be pleased to meet with the NRC staff to discuss any comments or issues related to the enclosed document. If you have any questions on the enclosed document or the general subject it addresses, please call Steve Leonard of Niagara Mohawk Power Company at (315) 349-4039.

Sincerely,

J. T. Beckham, Jr.
Southern Nuclear Operating Company
Chairman, BWR Vessel & Internals Project

Attachment 1

EPRI
Electric Power
Research Institute *Powering Progress through Innovative Solutions*

April 16, 1996

Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Subject: Request for Withholding of Proprietary Document; 10CFR2.790(a)(4) TR-105696, "BWR Vessel and Internals Project, Reactor Pressure Vessel and Internals Examination Guidelines (BWRVIP-03), Section 6A, Standards for Visual Inspection of Core Spray Piping, Spargers, and Associated Components," EPRI Report, February 1996

Gentlemen:

This a request under 10CFR2.790(a)(4) that the NRC withhold from public disclosure the proprietary document identified above (the "Report"). Ten copies of the Report and the affidavit in support of this request are enclosed.

EPRI desires to disclose the Report to the NRC as a means of exchanging information with the NRC for the purpose of supporting generic regulatory improvements relating to BWR reactor pressure vessel and internals examinations. EPRI would welcome any discussions between EPRI and the NRC related to the Report that the NRC desires to conduct.

The Report is for the NRC's internal use and may be used only for the purpose for which it is disclosed by EPRI. The Report should not be otherwise used or disclosed to any person outside the NRC without prior written permission from EPRI.

If you have any questions about the legal aspects of this request for withholding, please do not hesitate to contact me at (415) 855-2845. Questions on the contents of the Report should be directed to Warren Bilanin of EPRI at (415) 855-2340.

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PDR TOPRP EXIEPRI
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Sincerely,

Arthur Kenny
Intellectual Property Attorney
Intellectual Property Department

Enclosure

Headquarters: 3412 Hillview Avenue, Post Office Box 10412, Palo Alto, CA 94303, USA • (415) 855-2000
Washington Office: 2000 L Street, NW, Suite 805, Washington, DC 20036, USA • (202) 872-9222 • Fax: (202) 293-2697

RE: TR-105696, "BWR Vessel and Internals Project, Reactor Pressure Vessel and Internals Examination Guidelines (BWRVIP-03), Section 6A, Standards for Visual Inspection of Core Spray Piping, Spargers, and Associated Components," EPRI Report, February 1996

I. ARTHUR KENNY, being duly sworn, depose and state as follows:

I am an attorney at the Electric Power Research Institute ("EPRI") and I have been specifically delegated responsibility for reviewing the report listed above that is sought under this affidavit to be withheld (the "Report") and authorized to apply for its withholding on behalf of EPRI. This affidavit is submitted to the Nuclear Regulatory Commission ("NRC") pursuant to 10 CFR 2.790(a)(4) based on the fact that the Report consists of trade secrets of EPRI and that the NRC will receive the Report from EPRI under privilege and in confidence.

The basis for which the Report should be withheld from the public is set forth below:

(i) The Report has been held in confidence by EPRI, its owner. All those accepting the Report must agree to preserve the confidentiality of the Report.

(ii) The Report is of a type customarily held in confidence by EPRI and there is a rational basis therefor. The Report is trade secrets and is held in confidence by EPRI because to disclose it would prevent EPRI from licensing the Report at fees which would allow EPRI to recover its investment. If consultants and other businesses providing services in the nuclear power industry were able to publicly obtain the Report, they would be able to use it commercially for profit and avoid spending the large amount of money that EPRI was required to spend to prepare the Report. The rational basis that EPRI has for classifying the Report as trade secrets is the Uniform Trade Secrets Act which California adopted in 1984 and which has been adopted by over twenty states. The Uniform Trade Secrets Act defines a "trade secret" as follows:

> "Trade secret" means information, including a formula, pattern, compilation, program, device, method, technique, or process, that:
>
> (1) Derives independent economic value, actual or potential, from not being generally known to the public or to other

persons who can obtain economic value from its disclosure or use; and

(2) Is the subject of efforts that are reasonable under the circumstances to maintain its secrecy.

(iii) The Report will be transmitted to the NRC in confidence.

(iv) The Report is not available in public sources. EPRI developed the Report only after making a determination that the Report was not available from public sources. It required a large expenditure of dollars for EPRI to develop the Report. In addition, EPRI was required to use a large amount of time of EPRI employees. The money spent, plus the value of EPRI's staff time in preparing the Report, show that the Report is highly valuable to EPRI. Finally, the Report was developed only after a long period of effort of at least several months.

(v) A public disclosure of the Report would cause substantial harm to EPRI's competitive position and the ability of EPRI to license the Report both domestically and internationally. The Report can be properly acquired or duplicated by others only with an equivalent investment of time and effort.

I have read the foregoing and the matters stated therein are true and correct to the best of my knowledge, information and belief. I make this affidavit under penalty of perjury under the laws of the United States of America and under the laws of the State of California.

Executed at 3412 Hillview Avenue, Palo Alto, being the premises and place of business of the Electric Power Research Institute:

April 16, 1996

Arthur Kenny

Subscribed and sworn before me this day: April 16, 1996

Tamsen Helen Gagnon, Notary Public

TAMSEN HELEN GAGNON
COMM. # 1016243
Notary Public — California
SANTA CLARA COUNTY
My Comm. Expires FEB 5, 1998

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

April 2, 1996

Mr. J. T. Beckham, Jr., Chairman
Southern Nuclear Operating Company
BWR Vessel & Internals Project
42 Inverness Center Parkway
Birmingham, Alabama 35242

Dear Mr. ~~Beckham~~ Tom

SUBJECT: BWR OWNERS' GROUP, REQUEST FOR ADDITIONAL INFORMATION FOR TOPICAL REPORT EPRI TR-105697, "BWR VESSEL AND INTERNALS PROJECT, BWR REACTOR PRESSURE VESSEL SHELL WELD INSPECTION RECOMMENDATIONS (BWRVIP-05)"

By letter dated September 28, 1995, the Boiling Water Reactor Owners' Group (BWROG) submitted Topical Report EPRI TR-105697, "BWR Vessel and Internals Project, BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations (BWRVIP-05)," for NRC review. The report was intended to demonstrate the adequacy of alternative inspection requirements which would significantly reduce the extent of volumetric inservice inspections currently required on BWR reactor pressure vessel shell welds.

The Materials and Chemical Engineering Branch has completed a preliminary assessment of this report and is transmitting a request for additional information (RAI). Due to the complexity and multidisciplinary subject matter of this report, we anticipate another RAI. In order to complete our assessment on schedule, we request a response to the enclosed RAI within 45 days of receipt. If you have any questions, please contact Keith Wichman of my staff at (301) 415-2757.

Sincerely,

Brian W. Sheron

Brian W. Sheron, Director
Division of Engineering
Office of Nuclear Reactor Regulation

Enclosure: As stated

cc w/enclosure: Robin Dyle, Southern Nuclear Operating Company

Information in this record was deleted in accordance with the Freedom of Information Act, exemptions 4
FOIA- 2000-0186
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96-113

BWRVIP

BWR Vessel &
Internals Project _____ *Issue Management and Resolution*

March 18, 1996

Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Attention: C. E. Carpenter

Subject: "BWR Vessel and Internals Project, Evaluation of Crack Growth in BWR Stainless Steel RPV Internals (BWRVIP-14)," EPRI Report TR-105873, March 1996.

Enclosed are ten (10) copies of the document "BWR Vessel and Internals Project, Evaluation of Crack Growth in BWR Stainless Steel RPV Internals (BWRVIP-14)," EPRI Report TR-105873, March 1996. This document is being submitted as a means of exchanging information with the NRC for the purpose of supporting generic regulatory improvements relating to crack growth in stainless steel BWR vessel internal components.

The enclosed document has been endorsed by the BWRVIP, however, this endorsement should not be interpreted as a commitment by all members to a specific course of action. Each member must formally endorse the BWRVIP position for it to become that member's position.

Please note that the enclosed document contains proprietary information. Therefore, a letter requesting the report be withheld from public disclosure and an affidavit describing the basis for withholding this information is provided as Attachment 1.

Representatives of the BWRVIP would be pleased to meet with the NRC staff to discuss any comments or issues related to the enclosed document. If you have any questions on the enclosed document or the general subject it addresses, please call Robin Dyle of Southern Nuclear at (205) 877-7121.

Sincerely,

J. T. Beckham, Jr.
Southern Nuclear Operating Company
Chairman, BWR Vessel & Internals Project

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Information in this record was deleted in accordance with the Freedom of Information Act, exemptions 4
F.O.I.A. 2000-0186

Reply To: J. T. Beckham, Jr., BWRVIP Chairman, Southern Nuclear Operating Co., 42 Inverness Center Parkway, Birmingham, AL 35242 • (205) 877-7279 • Fax: (205) 802-0393

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PDR

EPRI
Electric Power
Research Institute — *Leadership in Electrification through Global Collaboration*

March 14, 1996

Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Subject: Request for Withholding of Proprietary Document; 10CFR2.790(a)(4) TR-105873, "BWR Vessel and Internals Project, Evaluation of Crack Growth in BWR Stainless Steel RPV Internals (BWRVIP-14)," EPRI Report, March 1996

Gentlemen:

This a request under 10CFR2.790(a)(4) that the NRC withhold from public disclosure the proprietary document identified above (the "Report"). Ten copies of the Report and the affidavit in support of this request are enclosed.

EPRI desires to disclose the Report to the NRC to as a means of exchanging information with the NRC for the purpose of supporting generic regulatory improvements relating to crack growth in stainless steel BWR vessel internal components. EPRI would welcome any discussions between EPRI and the NRC related to the Report that the NRC desires to conduct.

The Report is for the NRC's internal use and may be used only for the purpose for which it is disclosed by EPRI. The Report should not be otherwise used or disclosed to any person outside the NRC without prior written permission from EPRI.

If you have any questions about the legal aspects of this request for withholding, please do not hesitate to contact me at (415) 855-8957. Questions on the contents of the Report should be directed to Warren Bilanin of EPRI at (415) 855-2340.

Sincerely,

Mark S. Edwards
Intellectual Property Attorney
Intellectual Property Department

Enclosures

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PDR TOPRP EXIEPRI
C PDR

Headquarters: 3412 Hillview Avenue, Post Office Box 10412, Palo Alto, CA 94303, USA • (415) 855-2000
Washington Office: 2000 L Street, NW, Suite 805, Washington, DC 20036, USA • (202) 872-9222 • Fax: (202) 293-2697

RE: TR-105873, "BWR Vessel and Internals Project, Evaluation of Crack Growth in BWR Stainless Steel RPV Internals (BWRVIP-14)," EPRI Report, March 1996

I, MARK S. EDWARDS, being duly sworn, depose and state as follows:

I am an attorney at the Electric Power Research Institute ("EPRI") and I have been specifically delegated responsibility for reviewing the report listed above that is sought under this affidavit to be withheld (the "Report") and authorized to apply for its withholding on behalf of EPRI. This affidavit is submitted to the Nuclear Regulatory Commission ("NRC") pursuant to 10 CFR 2.790(a)(4) based on the fact that the Report consists of trade secrets of EPRI and that the NRC will receive the Report from EPRI under privilege and in confidence.

The basis for which the Report should be withheld from the public is set forth below:

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(ii) The Report is of a type customarily held in confidence by EPRI and there is a rational basis therefor. The Report is trade secrets and is held in confidence by EPRI because to disclose it would prevent EPRI from licensing the Report at fees which would allow EPRI to recover its investment. If consultants and other businesses providing services in the nuclear power industry were able to publicly obtain the Report, they would be able to use it commercially for profit and avoid spending the large amount of money that EPRI was required to spend to prepare the Report. The rational basis that EPRI has for classifying the Report as trade secrets is the Uniform Trade Secrets Act which California adopted in 1984 and which has been adopted by over twenty states. The Uniform Trade Secrets Act defines a "trade secret" as follows:

> "Trade secret" means information, including a formula, pattern, compilation, program, device, method, technique, or process, that:
>
> (1) Derives independent economic value, actual or potential, from not being generally known to the public or to other persons who can obtain economic value from its disclosure or use; and
>
> (2) Is the subject of efforts that are reasonable under the circumstances to maintain its secrecy.

(iii) The Report will be transmitted to the NRC in confidence.

(iv) The Report is not available in public sources. EPRI developed the Report only after making a determination that the Report was not available from public sources. It required a large expenditure of dollars for EPRI to develop the Report. In addition, EPRI was required to use a large amount of time of EPRI employees. The money spent, plus the value of EPRI's staff time in preparing the Report, show that the Report is highly valuable to EPRI. Finally, the Report was developed only after a long period of effort of at least several months.

(v) A public disclosure of the Report would cause substantial harm to EPRI's competitive position and the ability of EPRI to license the Report both domestically and internationally. The Report can be properly acquired or duplicated by others only with an equivalent investment of time and effort.

I have read the foregoing and the matters stated therein are true and correct to the best of my knowledge, information and belief. I make this affidavit under penalty of perjury under the laws of the United States of America and under the laws of the State of California.

Executed at 3412 Hillview Avenue, Palo Alto, being the premises and place of business of the Electric Power Research Institute:

March 14, 1996

Mark S. Edwards, Esq.

Subscribed and sworn before me this day:

CALIFORNIA ALL-PURPOSE ACKNOWLEDGMENT

State of CALIFORNIA

County of SANTA CLARA

On MARCH 15, 1996 (Date) before me, PATRICIA S. BALLARD (Name and Title of Officer (e.g., "Jane Doe, Notary Public")),

personally appeared MARK S. EDWARDS (Name(s) of Signer(s)),

☒ personally known to me – OR – ☐ proved to me on the basis of satisfactory evidence to be the person(s) whose name(s) is/are subscribed to the within instrument and acknowledged to me that he/she/they executed the same in his/her/their authorized capacity(ies), and that by his/her/their signature(s) on the instrument the person(s), or the entity upon behalf of which the person(s) acted, executed the instrument.

NNA1 PATRICIA S. BALLARD COMM. #1062301 Notary Public — California SANTA CLARA COUNTY My Comm. Expires JUN 18,1999 NNA1

WITNESS my hand and official seal.

Patricia S. Ballard
Signature of Notary Public

OPTIONAL

Though the information below is not required by law, it may prove valuable to persons relying on the document and could prevent fraudulent removal and reattachment of this form to another document.

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Document Date: _____ Number of Pages: _____

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Capacity(ies) Claimed by Signer(s)

Signer's Name: _____

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☐ Corporate Officer
Title(s): _____
☐ Partner — ☐ Limited ☐ General
☐ Attorney-in-Fact
☐ Trustee
☐ Guardian or Conservator
☐ Other: _____

Signer Is Representing:

RIGHT THUMBPRINT OF SIGNER
Top of thumb here

Signer's Name: _____

☐ Individual
☐ Corporate Officer
Title(s): _____
☐ Partner — ☐ Limited ☐ General
☐ Attorney-in-Fact
☐ Trustee
☐ Guardian or Conservator
☐ Other: _____

Signer Is Representing:

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EPRI

BWRVIP BWR Vessel & Internals Project 99-502

December 6, 1999

Document Control Desk
Nuclear Regulatory Commission
11555 Rockville Pike
Rockville, MD 20852-2738

Attention: C.E. Carpenter

Subject: BWRVIP Response to NRC Request for Additional Information on BWRVIP-53 (Project 704)

Reference 1: Letter from C.E. Carpenter to Carl Terry dated 4/7/99, "Proprietary Request for Additional Information - Review of "BWR Vessel And Internals Project, Standby Liquid Control Line Repair Design Criteria (Bwrvip-53)," TAC NO. MA2328

Enclosed are 10 copies of the BWRVIP response to the NRC Request for Additional Information (RAI) on the BWRVIP report "BWR Vessel and Internals Project, Standby Liquid Control Line Repair Design Criteria (BWRVIP-53)." The RAI was transmitted by the NRC to the BWRVIP in the Reference 1 letter. The enclosed response to the RAI repeats each of the individual issues in the NRC RAI, followed by the BWRVIP response to that issue.

As indicated in the referenced letter, the NRC staff has found the subject report to be proprietary in nature. Therefore, the enclosed information is also proprietary and should be withheld from public disclosure.

If you have any questions regarding this subject, please contact Bruce McLeod of Southern Nuclear (BWRVIP Repair Committee Technical Chairman) at (205) 992-7446.

Sincerely,

Tom J. Mulford for
Vaughn Wagoner
Chairman, BWRVIP Integration Committee

DF02 Add Timir Misra - Paper copy

Information in this record was deleted
in accordance with the Freedom of Information
Act, exemptions 4
FOIA- 2000-0186

CORPORATE HEADQUARTERS
3412 Hillview Avenue | Palo Alto CA 94304-1395 USA | 650.855.2000 | Customer Service 800.313.3774 | www.epri.com

ML003704632

97-943

BWRVIP

BWR Vessel &
Internals Project _____ *Issue Management and Resolution*

December 18, 1997

Document Control Desk
U. S. Nuclear Regulatory Commission
11555 Rockville Pike
Rockville, MD 20852-2738

Attention: C. E. Carpenter

Subject: BWRVIP Response to NRC Request for Additional Information on BWRVIP-05

Reference: Letter from Brian W. Sheron (NRC) to Carl Terry (BWRVIP Chairman), "Transmittal of NRC Staff's Independent Assessment of the Boiling Water Reactor Vessel and Internals Project BWRVIP-05 Report and Proprietary Request for Additional Information," dated August 14, 1997.

Enclosed are 10 copies of the BWRVIP response to the NRC Request for Additional Information (RAI) on the BWRVIP report "BWR Vessel and Internals Project, BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations (BWRVIP-05)" transmitted by the NRC letter referenced above.

The enclosed response repeats each of the individual items in the NRC RAI verbatim, followed by the BWRVIP response to that item. The enclosed information concerns a report that the NRC staff has found to be proprietary in nature. Therefore, the enclosed information is also proprietary and should be withheld from public disclosure.

If you have any questions on this subject, please contact Dana Covill of GPU Nuclear (BWRVIP Assessment Committee Technical Chairman) at (973) 316-7525.

Sincerely,

Carl Terry
Niagara Mohawk Power Company
Chairman, BWR Vessel and Internals Project

c: Dana Covill, GPU Nuclear

Information in this record was deleted in accordance with the Freedom of Information Act, exemptions 4
FOIA- 2000-0186

9712240085 / PDR

F114

Reply To: Carl Terry, BWRVIP Chairman, Niagara Mohawk Power Company, P. O. Box 63, Lycoming, NY 13093 • Phone: (315) 349-7263 • Fax: (315) 349-4753

6

December 15, 1997

Carl Terry, BWRVIP Vice-Chairman
Niagara Mohawk Power Company
Post Office Box 63
Lycoming, NY 13093

SUBJECT: PROPRIETARY REQUEST FOR ADDITIONAL INFORMATION - REVIEW OF "BWR VESSEL AND INTERNALS PROJECT, INTERNAL CORE SPRAY PIPING AND SPARGER REPLACEMENT DESIGN CRITERIA (BWRVIP-16)" (TAC NO. M98266)

Dear Mr. Terry:

By your application dated March 18, 1997, you submitted for NRC staff review the Electric Power Research Institute (EPRI) proprietary report, "BWR Vessel and Internals Project, Internal Core Spray Piping and Sparger Replacement Design Criteria (BWRVIP-16)," EPRI Report TR-106708, March 1997. The BWRVIP-16 report was submitted as a means of exchanging information with the staff for the purpose of supporting generic regulatory efforts related to repair of the subject reactor components.

The NRC staff has completed its preliminary review of the BWRVIP-16 report. As indicated in the attached request for additional information (RAI), the NRC staff has determined that additional information is needed. We request that the BWRVIP respond to the RAI as soon as possible in order for the NRC staff to complete its review in a timely manner. Since the attached concerns a report that the NRC staff has found to be proprietary in nature, the requested information will also be considered proprietary.

If you have any questions, please contact me at (301) 415-2169.

Sincerely,

[original signed by:]
C. E. Carpenter, Jr., Lead Project Manager
Materials and Chemical Engineering Branch
Division of Engineering
Office of Nuclear Reactor Regulation

Enclosure: As stated

cc: See next page

Enclosure: As stated
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

December 15, 1997

Carl Terry, BWRVIP Vice-Chairman
Niagara Mohawk Power Company
Post Office Box 63
Lycoming, NY 13093

SUBJECT: PROPRIETARY REQUEST FOR ADDITIONAL INFORMATION - REVIEW OF "BWR VESSEL AND INTERNALS PROJECT, INTERNAL CORE SPRAY PIPING AND SPARGER REPLACEMENT DESIGN CRITERIA (BWRVIP-16)" (TAC NO. M98266)

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C. E. Carpenter, Jr., Lead Project Manager
Materials and Chemical Engineering Branch
Division of Engineering
Office of Nuclear Reactor Regulation

Enclosure: As stated

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cc:

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Allentown, PA 18101

Joe Hagan, Executive Chairman
BWRVIP Inspection Task
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Paul Bemis, Executive Chairman
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Lewis Sumner, Executive Chairman
BWRVIP Mitigation Task
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Bill Campbell, BWRVIP Vice Chairman
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Robert Carter, EPRI BWRVIP
Assessment Manager
Greg Selby, EPRI BWRVIP
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Carl Larsen, Technical Chairman
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Vaughn Wagoner, Technical Chairman
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John Wilson, Technical Chairman
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Hancocks Bridge, NJ 08038

Bruce McLeod, Technical Chairman
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Southern Nuclear Operating Co.
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Warren Bilanin, EPRI BWRVIP
Integration Manager
Joe Gilman, EPRI BWRVIP
Mitigation Manager
Ken Wolfe, EPRI BWRVIP
Repair Manager
Electric Power Research Institute
P. O. Box 10412
3412 Hillview Ave.
Palo Alto, CA 94303

December 14, 1997

Carl Terry, BWRVIP Vice-Chairman
Niagara Mohawk Power Company
Post Office Box 63
Lycoming, NY 13093

SUBJECT: PROPRIETARY REQUEST FOR ADDITIONAL INFORMATION - REVIEW OF "BWR VESSEL AND INTERNALS PROJECT, TECHNICAL BASIS FOR PART CIRCUMFERENCE WELD OVERLAY REPAIR OF VESSEL INTERNAL CORE SPRAY PIPING (BWRVIP-34)" (TAC NO. M98880)

Dear Mr. Terry:

By your application dated May 22, 1997, you submitted for NRC staff review the Electric Power Research Institute (EPRI) proprietary report, "BWR Vessel and Internals Project, Technical Basis for Part Circumference Weld Overlay Repair of Vessel Internal Core Spray Piping (BWRVIP-34)," EPRI Report TR-108198, May 1997. The BWRVIP-34 report was submitted as a means of exchanging information with the staff for the purpose of supporting generic regulatory efforts related to repair of the subject reactor components.

The NRC staff has completed its preliminary review of the BWRVIP-34 report. As indicated in the attached request for additional information (RAI), the NRC staff has determined that additional information is needed. We request that the BWRVIP respond to the RAI as soon as possible in order for the NRC staff to complete its review in a timely manner. Since the attached concerns a report that the NRC staff has found to be proprietary in nature, the requested information will also be considered proprietary.

If you have any questions, please contact me at (301) 415-2169.

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[original signed by:]
C. E. Carpenter, Jr., Lead Project Manager
Materials and Chemical Engineering Branch
Division of Engineering
Office of Nuclear Reactor Regulation

Enclosure: As stated

cc: See next page

Enclosure: As stated

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

December 14, 1997

Carl Terry, BWRVIP Vice-Chairman
Niagara Mohawk Power Company
Post Office Box 63
Lycoming, NY 13093

SUBJECT: PROPRIETARY REQUEST FOR ADDITIONAL INFORMATION - REVIEW OF "BWR VESSEL AND INTERNALS PROJECT, TECHNICAL BASIS FOR PART CIRCUMFERENCE WELD OVERLAY REPAIR OF VESSEL INTERNAL CORE SPRAY PIPING (BWRVIP-34)" (TAC NO. M98880)

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If you have any questions, please contact me at (301) 415-2169.

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C. E. Carpenter, Jr., Lead Project Manager
Materials and Chemical Engineering Branch
Division of Engineering
Office of Nuclear Reactor Regulation

Enclosure: As stated

cc: See next page

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Palo Alto, CA 94303

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BWRVIP

BWR Vessel &
Internals Project _____ *Issue Management and Resolution*

December 11, 1997

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U. S. Nuclear Regulatory Commission
11555 Rockville Pike
Rockville, MD 20852

Attn: C.E. Carpenter

Subject: Transmittal of "BWR Vessel and Internals Project: BWR LPCI Coupling Inspection and Flaw Evaluation Guidelines (BWRVIP-42)," EPRI Report TR-108726, December, 1997.

Enclosed are ten (10) copies of the document "BWR Vessel and Internals Project: BWR LPCI Coupling Inspection and Flaw Evaluation Guidelines (BWRVIP-42)" EPRI Report TR-108726, December, 1997. Also enclosed are ten (10) copies of the non-proprietary version of the report. This document is being submitted as a means of exchanging information with the NRC for the purpose of supporting generic regulatory improvements related to the inspection of jet pumps.

The enclosed report is being submitted on behalf of the BWRVIP utilities. However, this submittal should not be interpreted as a commitment by all BWRVIP members to a specific course of action. Each member must formally endorse the BWRVIP position for it to become that member's position.

Please note that the enclosed proprietary document contains proprietary information. A letter requesting the report be withheld from public disclosure and affidavits describing the basis for withholding this information are provided as Attachment 1.

If you have any questions on this subject please call Dana Covill of GPU Nuclear, BWRVIP Assessment Committee Chairman, at (973) 316-7525.

Sincerely,

Carl Terry
Niagara Mohawk Power Company
Chairman, BWR Vessel and Internals Project

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FOIA- 2000 - 0186

c: Warren Bilanin

9712170167 / PDR

F/19

Reply To: Carl Terry, BWRVIP Chairman, Niagara Mohawk Power Company, P. O. Box 63, Lycoming, NY 13093 • Phone: (315) 349-7263 • Fax: (315) 349-4753

RE: TR-108726 "BWRVIP Vessel and Internals Project: LPCI Coupling Inspection and Flaw Evaluation Guidelines (BWRVIP-42)", December 1997

I. MARK D. FOX, being duly sworn, depose and state as follows:

I am an attorney at the Electric Power Research Institute ("EPRI") and I have been specifically delegated responsibility for reviewing the report listed above that is sought under this affidavit to be withheld (the "Report") and authorized to apply for its withholding on behalf of EPRI. This affidavit is submitted to the Nuclear Regulatory Commission ("NRC") pursuant to 10 CFR 2.790(a)(4) based on the fact that the Report consists of trade secrets of EPRI and that the NRC will receive the Report from EPRI under privilege and in confidence.

The basis for which the Report should be withheld from the public is set forth below:

(i) The Report has been held in confidence by EPRI, its owner. All those accepting the Report must agree to preserve the confidentiality of the Report.

(ii) The Report is of a type customarily held in confidence by EPRI and there is a rational basis therefor. The Report is trade secrets and is held in confidence by EPRI because to disclose it would prevent EPRI from licensing the Report at fees which would allow EPRI to recover its investment. If consultants and other businesses providing services in the nuclear power industry were able to publicly obtain the Report, they would be able to use it commercially for profit and avoid spending the large amount of money that EPRI was required to spend to prepare the Report. The rational basis that EPRI has for classifying the Report as trade secrets is the Uniform Trade Secrets Act which California adopted in 1984 and which has been adopted by over twenty states. The Uniform Trade Secrets Act defines a "trade secret" as follows:

> "Trade secret" means information, including a formula, pattern, compilation, program, device, method, technique, or process, that:
>
> (1) Derives independent economic value, actual or potential, from not being generally known to the public or to other

Headquarters: 3412 Hillview Avenue, Post Office Box 10412, Palo Alto, CA 94303, USA • (415)855-2000• Telex: 82977 EPRI UF • Fax: (415) 855-1026
Washington Office: 2000 L Street, NW, Suite 805, Washington, DC 20036, USA • (202) 872-9222 • Fax: (202) 296-6040

persons who can obtain economic value from its disclosure or use; and

(2) Is the subject of efforts that are reasonable under the circumstances to maintain its secrecy.

(iii) The Report will be transmitted to the NRC in confidence.

(iv) The Report is not available in public sources. EPRI developed the Report only after making a determination that the Report was not available from public sources. It required a large expenditure of dollars for EPRI to develop the Report. In addition, EPRI was required to use a large amount of time of EPRI employees. The money spent, plus the value of EPRI's staff time in preparing the Report, show that the Report is highly valuable to EPRI. Finally, the Report was developed only after a long period of effort of at least several months.

(v) A public disclosure of the Report would cause substantial harm to EPRI's competitive position and the ability of EPRI to license the Report both domestically and internationally. The Report can be properly acquired or duplicated by others only with an equivalent investment of time and effort.

I have read the foregoing and the matters stated therein are true and correct to the best of my knowledge, information and belief. I make this affidavit under penalty of perjury under the laws of the United States of America and under the laws of the State of California.

Executed at 3412 Hillview Avenue, Palo Alto, being the premises and place of business of the Electric Power Research Institute:

December 11, 1997

Mark D. Fox

Subscribed and sworn before me this day: December 11, 1997

Tamsen Helen Gagnon, Notary Public

Tamsen Helen Gagnon, Notary Public

TAMSEN HELEN GAGNON
COMM. # 1016743
Notary Public — California
SANTA CLARA CO
My Comm. Expires F

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

December 9, 1997

MEMORANDUM TO: Edmund J. Sullivan, Acting Chief
Materials and Chemical Engineering Branch
Division of Engineering

FROM: C. E. Carpenter, Jr., Lead Project Manager
Materials and Chemical Engineering Branch
Division of Engineering

SUBJECT: MEETING SUMMARY FOR NOVEMBER 5 AND 6, 1997, MEETINGS WITH BOILING WATER REACTORS VESSEL AND INTERNALS PROJECT TECHNICAL CHAIRS, REGARDING BWRVIP-07 SAFETY EVALUATION REPORT AND TO DISCUSS ISSUES RELATED TO BWR LICENSEE VESSEL SURVEILLANCE PROGRAMS, RELATIVE TO MONITORING RADIATION EMBRITTLEMENT AT BWR FACILITIES.

On Wednesday, November 5, 1997, several members of the U.S. Nuclear Regulatory Commission (NRC) staff participated in a meeting with members of the Boiling Water Reactors Vessel & Internals Project (BWRVIP) at the NRC's Rockville, MD, Headquarters to discuss the staff's safety evaluation report (SER) of the BWRVIP's proprietary report "BWR Vessel and Internals Project, Guidelines for Reinspection of BWR Core Shrouds (BWRVIP-07)." Attachment 1 is the November 5, 1997, meeting participants. Attachment 2 is the BWRVIP's November 5, 1997, presentation viewgraphs (proprietary, to be withheld from public release).

On Thursday, November 6, 1997, the staff met with members of the BWRVIP in a public meeting to discuss issues related to BWR licensee vessel surveillance programs, relative to monitoring radiation embrittlement at BWR facilities. Attachment 3 is the November 6, 1997, meeting participants. Attachment 4 is the BWRVIP's November 6, 1997, presentation viewgraphs.

November 5, 1997, BWRVIP-07 SER Meeting

On Wednesday, November 5, 1997, several members of the staff participated in a meeting with members of the BWRVIP to discuss the staff's SER of the BWRVIP's proprietary "Guidelines for Reinspection of BWR Core Shrouds (BWRVIP-07)" report. The SER was issued September 15, 1997. The BWRVIP-07 guidelines allow for plant-specific analyses to be performed for a given weld or for the entire shroud. These plant-specific analyses are not addressed in the scope of this report, and NRC approval must be obtained on a case-by-case basis. The staff review of flaw analyses performed according to the more conservative "BWR Core Shroud Inspection and Flaw Evaluation Guidelines," (GENE-523-113-0894, Revs. 0 and 1) will continue to be evaluated in accordance with the previous SEs. Flaw evaluations conducted according to the more comprehensive but less conservative guidelines of BWRVIP-07 will be evaluated by the staff and will consider the recommendations contained within this SE. The staff reviewed BWRVIP-07 to determine whether its guidance would ensure the structural integrity of the BWR core shroud.

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In general, the staff found that the guidance presented in BWRVIP-07 ensures the structural integrity of the core shroud. The scope of the guidelines is comprehensive, covering circumferential, vertical, and ring segment welds, as well as repair components for both Category B and C plants. Industry experience shows that the inspection methodologies — enhanced visual and ultrasonic methods — are appropriate for detecting cracking. In addition, the guidance is consistent with that in earlier staff-approved reports on the inspection and evaluation of core shrouds. The guidelines of BWRVIP-07 incorporate many of the recommendations in the referenced safety evaluations and also allow for revision on the basis of new data and lessons learned. The staff found much of the proposed guidance acceptable. However, the staff also found the guidance to be needlessly complicated and difficult to implement and audit. Therefor, the staff's SER contained several recommended modification to the BWRVIP-07 report, and then listed several concerns with the overall report. In summary, the BWRVIP-07 report will require several significant modifications prior to full staff acceptance.

The BWRVIP and NRC staff met in an attempt to answer and close several of these issues; however, the staff believes that addressing the open items *in toto* are necessary in order to adequately address the possibility for a new or different accident, or the possible loss in safety margin. As such, the BWRVIP agreed to address the staff's concerns raised in the BWRVIP-07 SER, and to provide a more complete description of several items that were unclear to the staff, and to address the possibility of vertical cracking in the BWR shroud's axial welds.

The BWRVIP staff and the NRC staff agreed to continue to meet to discuss emerging technical issues, and the status of the technical reviews and of NRC and BWRVIP resource allocation. Further meetings will be scheduled as needed.

November 6, 1997, BWR Vessel Surveillance Programs

On Thursday, November 6, 1997, the staff met with members of the BWRVIP in a public meeting to discuss issues related to BWR licensee RPV surveillance programs, monitoring radiation embrittlement. The topics covered included the background of the issue, its historical perspective, recent BWRVIP activities in this area, and an open discussion.

The staff's principal concern in this area was that some BWR surveillance programs may be missing baseline data to measure radiation embrittlement. The staff noted that the missing data reduces the staff's ability to assess radiation embrittlement in BWR environments. Current pressure-temperature (P-T) limit curves are, in the absence of direct RPV surveillance data, determined based on generic models from RG 1.99, Rev. 2, which are based in large part on PWR data. While the staff has concluded that the use of these models for BWRs at this time is appropriate, sufficient data from a number of BWRs is necessary to continue to support this conclusion.

The BWRVIP participants stated that many BWRs were fabricated and constructed before 10 CFR Appendices G and H were issued. Consequently, baseline data for some of these plants does not exist. While the industry has been attempting to discover additional archival material, the search has not been especially successful. However, the BWRVIP is

GENE VTRA

exploring an integrated BWR program which would make use of 56 capsules at 38 plants which have yet to be tested. The BWRVIP stated that the major advantage to this approach is that samples could be removed at times to permit coverage of a wide fluence range and minimize duplication. The staff cautioned that the integration approach is limited to situations where the environment is equivalent and there is no reduction in the amount of tests.

The staff commented as to the advisability of several alternatives for those plants that don't have baseline data, including (a) contacting vendors and looking for the same material that might have been used at another plant (e.g., "sister" plant method), (2) reconstitution and annealing of broken Charpy specimens, and (3) an integrated surveillance program.

The BWRVIP questioned the staff as to the acceptability of reconstitution and annealing; would annealing samples out of surveillance capsules be acceptable as substitute for baseline data? The staff stated that relating the annealed state to the baseline state would have to be defended technically.

The BWRVIP agreed to expedite its expanded response to GL 92-01, Rev. 1, Supp. 1, which is due by the end of 1997. Further, the BWRVIP agreed to submit a letter to the staff during the first quarter of CY-1998 stating their intentions as to the resolution of this issue. this letter will also discuss the regulatory impact of their intentions (i.e., exemptions, etc.).

The BWRVIP staff and the NRC staff agreed to continue to meet to discuss emerging technical issues, and the status of the technical reviews and of NRC and BWRVIP resource allocation. Further meetings will be scheduled as needed.

Attachments:
1. November 5, 1997, Meeting Participants
2. November 5, 1997, BWRVIP Viewgraphs
3. November 6, 1997, Meeting Participants
4. November 6, 1997, BWRVIP Viewgraphs

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The BWRVIP questioned the staff as to the acceptability of reconstitution and annealing; would annealing samples out of surveillance capsules be acceptable as substitute for baseline data? The staff stated that relating the annealed state to the baseline state would have to be defended technically.

The BWRVIP agreed to expedite its expanded response to GL 92-01, Rev. 1, Supp. 1, which is due by the end of 1997. Further, the BWRVIP agreed to submit a letter to the staff during the first quarter of CY-1998 stating their intentions as to the resolution of this issue. this letter will also discuss the regulatory impact of their intentions (i.e., exemptions, etc.).

The BWRVIP staff and the NRC staff agreed to continue to meet to discuss emerging technical issues, and the status of the technical reviews and of NRC and BWRVIP resource allocation. Further meetings will be scheduled as needed.

cc: See next page

Attachments:
1. November 5, 1997, Meeting Participants
2. November 5, 1997, BWRVIP Viewgraphs
3. November 6, 1997, Meeting Participants
4. November 6, 1997, BWRVIP Viewgraphs

Distribution:

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BWRVIP

BWR Vessel &
Internals Project _____ *Issue Management and Resolution*

December 8, 1997

Document Control Desk
Nuclear Regulatory Commission
11555 Rockville Pike
Rockville, MD 20852-2738

Attention: C.E. Carpenter

Subject: BWRVIP Response to NRC Request for Additional Information on BWRVIP-28

Reference: Letter from C.E. Carpenter, Jr., (NRC) to Carl Terry (BWRVIP Chairman), "Proprietary Request for Additional Information - Review of BWR Vessel and Internals Project Reports, "BWR Vessel and Internals Project, Assessment of BWR Jet Pump Riser Elbow to Thermal Sleeve Weld Cracking (BWRVIP-28)" (TAC NO. M97395)"

Enclosed are 10 copies of the BWRVIP response to the NRC Request for Additional Information (RAI) on the BWRVIP report "BWR Vessel and Internals Project, Assessment of BWR Jet Pump Riser Elbow to Thermal Sleeve Weld Cracking (BWRVIP-28)" transmitted by the referenced NRC letter.

The enclosed response repeats each of the individual items in the NRC RAI verbatim, followed by the BWRVIP response to that item. As indicated in the referenced letter, the enclosed information concerns a report that the NRC staff has found to be proprietary in nature. Therefore, the enclosed information is also proprietary and should be withheld from public disclosure.

If you have any questions on this subject, please contact Dana Covill of GPU Nuclear (BWRVIP Assessment Committee Technical Chairman) at (201) 316-7525.

Sincerely,

Warren Bilanin for

Vaughn Wagoner
Technical Chairman
BWRVIP Integration Committee
Carolina Power and Light

c: Dana Covill

Information in this record was deleted in accordance with the Freedom of Information Act, exemptions 4
FOIA- 2000-0186

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97-909

BWRVIP

BWR Vessel &
Internals Project _____ *Issue Management and Resolution*

November 26, 1997

Document Control Desk
U. S. Nuclear Regulatory Commission
11555 Rockville Pike
Rockville, MD 20852-2738

Attention: C. E. Carpenter

Subject: Response to NRC Staff Safety Evaluation of the BWR Vessel and Internals Project BWRVIP-07 Report

References:
1. Letter from Gus C. Lainas (NRC) to Carl Terry (BWRVIP Chairman), "Transmittal of NRC Staff's Safety Evaluation of the BWR Vessel and Internals Project BWRVIP-07 Report," dated September 15, 1997.
2. Letter from J. T. Beckham, Jr. (BWRVIP Chairman) to NRC Document Control Desk, "BWR Vessel and Internals Project, Guidelines for Reinspection of BWR Core Shrouds (BWRVIP-07), EPRI Report TR-105747, February 1996," dated February 29, 1996.

Enclosed is the BWRVIP response to issues identified in the NRC staff Safety Evaluation (SE) of the BWRVIP-07 report. The NRC staff issues were transmitted to the BWRVIP by the Reference 1 letter identified above.

Representatives of the BWRVIP met with the NRC staff on November 5, 1997 to discuss the issues identified in the NRC SE. The enclosed BWRVIP response to the issues in the NRC SE are consistent with the discussions held on November 5. Note that the BWRVIP has proposed changes to Category B plant shroud reinspection guidelines. These proposed changes as described in the enclosure have not yet been formally approved by the BWRVIP. However, we expect such approval to be forthcoming shortly and we will advise you when that approval has been finalized.

Please note that the enclosed information contains proprietary information. Therefore, the request for withholding the BWRVIP-07 report from public disclosure transmitted to the NRC with the Reference 2 letter identified above also applies to the enclosed information.

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If you have any questions on the enclosed information or the general subject it addresses, please call Dana Covill of GPU Nuclear at (201) 316-7525 or Bob Carter of EPRI at (704) 547-6019.

Sincerely,

Larry Stewart

for Vaughn Wagoner
Technical Chairman
BWRVIP Integration Committee
Carolina Power & Light Company

97-909

BWRVIP

BWR Vessel &
Internals Project _____ *Issue Management and Resolution*

November 26, 1997

Document Control Desk
U. S. Nuclear Regulatory Commission
11555 Rockville Pike
Rockville, MD 20852-2738

Attention: C. E. Carpenter

Subject: Response to NRC Staff Safety Evaluation of the BWR Vessel and Internals Project BWRVIP-07 Report

References:
1. Letter from Gus C. Lainas (NRC) to Carl Terry (BWRVIP Chairman), "Transmittal of NRC Staff's Safety Evaluation of the BWR Vessel and Internals Project BWRVIP-07 Report," dated September 15, 1997.
2. Letter from J. T. Beckham, Jr. (BWRVIP Chairman) to NRC Document Control Desk, "BWR Vessel and Internals Project, Guidelines for Reinspection of BWR Core Shrouds (BWRVIP-07), EPRI Report TR-105747, February 1996," dated February 29, 1996.

Enclosed is the BWRVIP response to issues identified in the NRC staff Safety Evaluation (SE) of the BWRVIP-07 report. The NRC staff issues were transmitted to the BWRVIP by the Reference 1 letter identified above.

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Please note that the enclosed information contains proprietary information. Therefore, the request for withholding the BWRVIP-07 report from public disclosure transmitted to the NRC with the Reference 2 letter identified above also applies to the enclosed information.

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If you have any questions on the enclosed information or the general subject it addresses, please call Dana Covill of GPU Nuclear at (201) 316-7525 or Bob Carter of EPRI at (704) 547-6019.

Sincerely,

Larry Steinert

for Vaughn Wagoner
Technical Chairman
BWRVIP Integration Committee
Carolina Power & Light Company

97-865

BWRVIP

BWR Vessel &
Internals Project _____ *Issue Management and Resolution*

October 27, 1997

Document Control Desk
U.S. Nuclear Regulatory Commission
11555 Rockville Pike
Rockville, MD 20852

Attention: C.E. Carpenter

Subject: Proprietary version of "BWR Vessel and Internals Project, Weldability of Irradiated LWR Structural Components (BWRVIP-45)," EPRI Report TR-108707, September, 1997

Enclosed are two copies of the subject report. The report is being provided as you requested for your information.

Please note that the report contain proprietary information. A letter requesting the report be withheld from public disclosure and affidavits describing the basis for withholding this information are enclosed.

If you have any questions on this subject, please call Bruce McLeod, BWRVIP Repair Committee Technical Chairman, Southern Nuclear, at 205/992-7446.

Sincerely,

Carl Terry
Niagra Mohawk Power Company
Chairman, BWR Vessel and Internals Project

Information in this record was deleted
in accordance with the Freedom of Information
Act, exemptions 4
FOIA- 2000-0186

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4

EPRI
Electric Power
Research Institute *Powering Progress through Innovative Solutions*

October 27, 1997

Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Subject: Request for Withholding of Proprietary Document; 10CFR2.790(a)(4)"BWR Vessel and Internals Project, Weldability of Irradiated LWR Structural Components (BWRVIP-45)", EPRI Report TR-108707, September, 1997

Gentlemen:

This a request under 10CFR2.790(a)(4) that the NRC withhold from public disclosure the proprietary document identified above (the "Report"). Two copies of the Report and the affidavit in support of this request are enclosed.

EPRI desires to disclose the Report to the NRC for your information. EPRI would welcome any discussions between EPRI and the NRC related to the Report that the NRC desires to conduct.

The Report is for the NRC's internal use and may be used only for the purpose for which it is disclosed by EPRI. The Report should not be otherwise used or disclosed to any person outside the NRC without prior written permission from EPRI.

If you have any questions about the legal aspects of this request for withholding, please do not hesitate to contact me at (415) 855-2845. Questions on the contents of the Report should be directed to Warren Bilanin of EPRI at (650) 855-2340.

Sincerely,

Arthur Kenny
Intellectual Property Attorney
Intellectual Property Department

Enclosures

9710300140/ ODR

Headquarters: 3412 Hillview Avenue, Post Office Box 10412, Palo Alto, CA 94303, USA • (415)855-2000• Telex: 82977 EPRI UF • Fax: (415) 855-1026
Washington Office: 2000 L Street, NW, Suite 805, Washington, DC 20036, USA • (202) 872-9222 • Fax: (202) 296-6040

Headquarters: 3412 Hillview Avenue, Post Office Box 10412, Palo Alto, CA 94303, USA • (415) 855-2000 • www.epri.com
Washington Office: 2000 L Street, NW, Suite 805, Washington, DC 20036, USA • (202) 872-9222 • Fax: (202) 293-2697

EPRI
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October 27, 1997

Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Subject: Request for Withholding of Proprietary Document; 10CFR2.790(a)(4)"BWR Vessel and Internals Project, Weldability of Irradiated LWR Structural Components (BWRVIP-45)", EPRI Report TR-108707, September, 1997

Gentlemen:

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EPRI desires to disclose the Report to the NRC for your information. EPRI would welcome any discussions between EPRI and the NRC related to the Report that the NRC desires to conduct.

The Report is for the NRC's internal use and may be used only for the purpose for which it is disclosed by EPRI. The Report should not be otherwise used or disclosed to any person outside the NRC without prior written permission from EPRI.

If you have any questions about the legal aspects of this request for withholding, please do not hesitate to contact me at (415) 855-2845. Questions on the contents of the Report should be directed to Warren Bilanin of EPRI at (650) 855-2340.

Sincerely,

Arthur Kenny
Intellectual Property Attorney
Intellectual Property Department

Enclosures

97,0300140/ ODR

Headquarters: 3412 Hillview Avenue, Post Office Box 10412, Palo Alto, CA 94303, USA • (415)855-2000• Telex: 82977 EPRI UF • Fax: (415) 855-1026
Washington Office: 2000 L Street, NW, Suite 805, Washington, DC 20036, USA • (202) 872-9222 • Fax: (202) 296-6040

Headquarters: 3412 Hillview Avenue, Post Office Box 10412, Palo Alto, CA 94303, USA • (415) 855-2000 • www.epri.com

RE: EPRI Report "BWR Vessel and Internals Project, Weldability of Irradiated LWR Structural Components (BWRVIP-45)", EPRI Report TR-108707, September, 1997

I. ARTHUR KENNY, being duly sworn, depose and state as follows:

I am an attorney at the Electric Power Research Institute ("EPRI") and I have been specifically delegated responsibility for reviewing the report listed above that is sought under this affidavit to be withheld (the "Report") and authorized to apply for its withholding on behalf of EPRI. This affidavit is submitted to the Nuclear Regulatory Commission ("NRC") pursuant to 10 CFR 2.790(a)(4) based on the fact that the Report consists of trade secrets of EPRI and that the NRC will receive the Report from EPRI under privilege and in confidence.

The basis for which the Report should be withheld from the public is set forth below:

(i) The Report has been held in confidence by EPRI, its owner. All those accepting the Report must agree to preserve the confidentiality of the Report.

(ii) The Report is of a type customarily held in confidence by EPRI and there is a rational basis therefor. The Report is trade secrets and is held in confidence by EPRI because to disclose it would prevent EPRI from licensing the Report at fees which would allow EPRI to recover its investment. If consultants and other businesses providing services in the nuclear power industry were able to publicly obtain the Report, they would be able to use it commercially for profit and avoid spending the large amount of money that EPRI was required to spend to prepare the Report. The rational basis that EPRI has for classifying the Report as trade secrets is the Uniform Trade Secrets Act which California adopted in 1984 and which has been adopted by over twenty states. The Uniform Trade Secrets Act defines a "trade secret" as follows:

> "Trade secret" means information, including a formula, pattern, compilation, program, device, method, technique, or process, that:
>
> (1) Derives independent economic value, actual or potential, from not being generally known to the public or to other persons who can obtain economic value from its disclosure or use; and

(2) Is the subject of efforts that are reasonable under the circumstances to maintain its secrecy.

(iii) The Report will be transmitted to the NRC in confidence.

(iv) The Report is not available in public sources. EPRI developed the Report only after making a determination that the Report was not available from public sources. It required a large expenditure of dollars for EPRI to develop the Report. In addition, EPRI was required to use a large amount of time of EPRI employees. The money spent, plus the value of EPRI's staff time in preparing the Report, show that the Report is highly valuable to EPRI. Finally, the Report was developed only after a long period of effort of at least several months.

(v) A public disclosure of the Report would cause substantial harm to EPRI's competitive position and the ability of EPRI to license the Report both domestically and internationally. The Report can be properly acquired or duplicated by others only with an equivalent investment of time and effort.

I have read the foregoing and the matters stated therein are true and correct to the best of my knowledge, information and belief. I make this affidavit under penalty of perjury under the laws of the United States of America and under the laws of the State of California.

Executed at 3412 Hillview Avenue, Palo Alto, being the premises and place of business of the Electric Power Research Institute:

October 27, 1997

Arthur Kenny, Esq.

Subscribed and sworn before me this day: October 27, 1997

Tamsen Helen Gagnon, Notary Public

Tamsen Helen Gagnon, NOTARY PUBLIC

NNA1 TAMSEN HELEN GAGNON
COMM. # 1016243
Notary Public — California
SANTA CLARA COUNTY
My Comm. Expires FEB 5, 1998 NNA1

97-866

BWRVIP

BWR Vessel &
Internals Project _____ *Issue Management and Resolution*

October 27, 1997

Document Control Desk
U.S. Nuclear Regulatory Commission
11555 Rockville Pike
Rockville, MD 20852

Attention: C.E. Carpenter

Subject: Proprietary version of "BWR Vessel and Internals Project, Underwater Weld Repair of Nickel Alloy Reactor Vessel Internals (BWRVIP-44)," EPRI Report TR-108708, September, 1997

Enclosed are two copies of the subject report. The report is being provided as you requested for your information.

Please note that the report contain proprietary information. A letter requesting the report be withheld from public disclosure and affidavits describing the basis for withholding this information are enclosed.

If you have any questions on this subject, please call Bruce McLeod, BWRVIP Repair Committee Technical Chairman, Southern Nuclear, at 205/992-7446.

Sincerely,

Carl Terry
Niagra Mohawk Power Company
Chairman, BWR Vessel and Internals Project

Information in this record was deleted
in accordance with the Freedom of Information
Act, exemptions _____
FOIA- 2000-0186
9711030055/PDR

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EPRI
Electric Power
Research Institute *Powering Progress through Innovative Solutions*

October 27, 1997

Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Subject: Request for Withholding of Proprietary Document; 10CFR2.790(a)(4) "BWR Vessel and Internals Project, Underwater Weld Repair of Nickel Alloy Reactor Vessel Internals (BWRVIP-44)", EPRI Report TR-108708, September, 1997

Gentlemen:

This a request under 10CFR2.790(a)(4) that the NRC withhold from public disclosure the proprietary document identified above (the "Report"). Two copies of the Report and the affidavit in support of this request are enclosed.

EPRI desires to disclose the Report to the NRC for your information. EPRI would welcome any discussions between EPRI and the NRC related to the Report that the NRC desires to conduct.

The Report is for the NRC's internal use and may be used only for the purpose for which it is disclosed by EPRI. The Report should not be otherwise used or disclosed to any person outside the NRC without prior written permission from EPRI.

If you have any questions about the legal aspects of this request for withholding, please do not hesitate to contact me at (415) 855-2845. Questions on the contents of the Report should be directed to Warren Bilanin of EPRI at (415) 855-2340.

Sincerely,

Arthur Kenny
Intellectual Property Attorney
Intellectual Property Department

Enclosures

9711030015/971

Headquarters: 3412 Hillview Avenue, Post Office Box 10412, Palo Alto, CA 94303, USA • (415)855-2000• Telex: 82977 EPRI UF • Fax: (415) 855-1026
Washington Office: 2000 L Street, NW, Suite 805, Washington, DC 20036, USA • (202) 872-9222 • Fax: (202) 296-6040

Headquarters: 3412 Hillview Avenue, Post Office Box 10412, Palo Alto, CA 94303, USA • (415) 855-2000 • www.epri.com

RE: EPRI Report "BWR Vessel and Internals Project, Underwater Weld Repair of Nickel Alloy Reactor Vessel Internals (BWRVIP-44)", EPRI Report TR-108708, September, 1997

I. ARTHUR KENNY, being duly sworn, depose and state as follows:

I am an attorney at the Electric Power Research Institute ("EPRI") and I have been specifically delegated responsibility for reviewing the report listed above that is sought under this affidavit to be withheld (the "Report") and authorized to apply for its withholding on behalf of EPRI. This affidavit is submitted to the Nuclear Regulatory Commission ("NRC") pursuant to 10 CFR 2.790(a)(4) based on the fact that the Report consists of trade secrets of EPRI and that the NRC will receive the Report from EPRI under privilege and in confidence.

The basis for which the Report should be withheld from the public is set forth below:

(i) The Report has been held in confidence by EPRI, its owner. All those accepting the Report must agree to preserve the confidentiality of the Report.

(ii) The Report is of a type customarily held in confidence by EPRI and there is a rational basis therefor. The Report is trade secrets and is held in confidence by EPRI because to disclose it would prevent EPRI from licensing the Report at fees which would allow EPRI to recover its investment. If consultants and other businesses providing services in the nuclear power industry were able to publicly obtain the Report, they would be able to use it commercially for profit and avoid spending the large amount of money that EPRI was required to spend to prepare the Report. The rational basis that EPRI has for classifying the Report as trade secrets is the Uniform Trade Secrets Act which California adopted in 1984 and which has been adopted by over twenty states. The Uniform Trade Secrets Act defines a "trade secret" as follows:

> "Trade secret" means information, including a formula, pattern, compilation, program, device, method, technique, or process, that:
>
> (1) Derives independent economic value, actual or potential, from not being generally known to the public or to other persons who can obtain economic value from its disclosure or use; and

(2) Is the subject of efforts that are reasonable under the circumstances to maintain its secrecy.

(iii) The Report will be transmitted to the NRC in confidence.

(iv) The Report is not available in public sources. EPRI developed the Report only after making a determination that the Report was not available from public sources. It required a large expenditure of dollars for EPRI to develop the Report. In addition, EPRI was required to use a large amount of time of EPRI employees. The money spent, plus the value of EPRI's staff time in preparing the Report, show that the Report is highly valuable to EPRI. Finally, the Report was developed only after a long period of effort of at least several months.

(v) A public disclosure of the Report would cause substantial harm to EPRI's competitive position and the ability of EPRI to license the Report both domestically and internationally. The Report can be properly acquired or duplicated by others only with an equivalent investment of time and effort.

I have read the foregoing and the matters stated therein are true and correct to the best of my knowledge, information and belief. I make this affidavit under penalty of perjury under the laws of the United States of America and under the laws of the State of California.

Executed at 3412 Hillview Avenue, Palo Alto, being the premises and place of business of the Electric Power Research Institute:

October 27, 1997

Arthur Kenny, Esq.

Subscribed and sworn before me this day: October 27, 1997

Tamsen Helen Gagnon, NOTARY PUBLIC

TAMSEN HELEN GAGNON
COMM. # 1016243
Notary Public — California
SANTA CLARA COUNTY
My Comm. Expires FEB 5, 1998

97-832

BWRVIP

BWR Vessel &
Internals Project _____ *Issue Management and Resolution*

October 15, 1997

Document Control Desk
U. S. Nuclear Regulatory Commission
11555 Rockville Pike
Rockville, MD 20852

Attention: C.E. Carpenter

Subject: "BWR Vessel and Internals Project ,BWR Jet Pump Assembly Inspection and Flaw Evaluation Guidelines (BWRVIP-41)," EPRI Report TR-108728, October, 1997.

Enclosed are ten (10) copies of the document "BWR Vessel and Internals Project, BWR Jet Pump Assembly Inspection and Flaw Evaluation Guidelines (BWRVIP-41)," EPRI Report TR-108728, October, 1997. Also enclosed are ten (10) copies of the non-proprietary version of the report. This document is being submitted as a means of exchanging information with the NRC for the purpose of supporting generic regulatory improvements related to the inspection of jet pumps.

The enclosed report is being submitted on behalf of the BWRVIP utilities. However, this submittal should not be interpreted as a commitment by all BWRVIP members to a specific course of action. Each member must formally endorse the BWRVIP position for it to become that member's position.

Please note that the enclosed proprietary document contains proprietary information. A letter requesting the report be withheld from public disclosure and affidavits describing the basis for withholding this information are provided as Attachment 1.

If you have any questions on this subject please call Dana Covill of GPU Nuclear, BWRVIP Assessment Committee Chairman, at (201) 316-7525.

Sincerely,

Carl Terry
Niagara Mohawk Power Company
Vice Chairman, BWR Vessel and Internals Project

Enclosure

Information in this record was deleted
in accordance with the Freedom of Information
Act, exemptions 4
FOIA-2000-0186 9910170021

Reply To: Carl Terry, BWRVIP Chairman, Niagara Mohawk Power Company, P. O. Box 63, Lycoming, NY 13093 • Phone: (315) 349-7263 • Fax: (315) 349-4753

F126

EPRI
Electric Power
Research Institute *Leadership in Science and Technology*

8/29/97

Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555-2738

Subject: TR-108728 report, "BWR Vessel and Internals Project: BWR Jet Pump Assembly Inspection and Flaw Evaluation Guidelines (BWRVIP-41) EPRI Report TR-108728, ~~September,~~ October 1997.

Gentlemen:

This a request under 10CFR2.790(a)(4) that the NRC withhold from public disclosure the proprietary document identified above (the "Report"). Ten copies of the Report and the affidavit in support of this request are enclosed.

EPRI desires to disclose the Report to the NRC as a means of exchanging information with the NRC for the purpose of supporting generic regulatory improvements related to repair of the subject reactor components. EPRI would welcome any discussions between EPRI and the NRC related to the Report that the NRC desires to conduct.

The Report is for the NRC's internal use and may be used only for the purpose for which it is disclosed by EPRI. The Report should not be otherwise used or disclosed to any person outside the NRC without prior written permission from EPRI.

Headquarters: 3412 Hillview Avenue, Post Office Box 10412, Palo Alto, CA 94303, USA • (415)855-2000 • Telex: 82977 EPRI UF • Fax: (415) 855-1026
Washington Office: 2000 L Street, NW, Suite 805, Washington, DC 20036, USA • (202) 872-9222 • Fax: (202) 296-6040

If you have any questions about the legal aspects of this request for withholding, please do not hesitate to contact me at (415) 855-8957. Questions on the contents of the Report should be directed to Warren Bilanin of EPRI at (415) 855-2340.

Sincerely,

Mark D. Fox
Intellectual Property Attorney
Intellectual Property Department

Enclosures

RE: TR-108728 report, "BWR Vessel and Internals Project: BWR Jet Pump Assembly Inspection and Flaw Evaluation Guidelines (BWRVIP-41) EPRI Report TR-108728, ~~September~~ October MDF, 1997.

I. MARK D. FOX, being duly sworn, depose and state as follows:

I am an attorney at the Electric Power Research Institute ("EPRI") and I have been specifically delegated responsibility for reviewing the report listed above that is sought under this affidavit to be withheld (the "Report") and authorized to apply for its withholding on behalf of EPRI. This affidavit is submitted to the Nuclear Regulatory Commission ("NRC") pursuant to 10 CFR 2.790(a)(4) based on the fact that the Report consists of trade secrets of EPRI and that the NRC will receive the Report from EPRI under privilege and in confidence.

The basis for which the Report should be withheld from the public is set forth below:

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> "Trade secret" means information, including a formula, pattern, compilation, program, device, method, technique, or process, that:
>
> (1) Derives independent economic value, actual or potential, from not being generally known to the public or to other persons who can obtain economic value from its disclosure or use; and

(2) Is the subject of efforts that are reasonable under the circumstances to maintain its secrecy.

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(v) A public disclosure of the Report would cause substantial harm to EPRI's competitive position and the ability of EPRI to license the Report both domestically and internationally. The Report can be properly acquired or duplicated by others only with an equivalent investment of time and effort.

I have read the foregoing and the matters stated therein are true and correct to the best of my knowledge, information and belief. I make this affidavit under penalty of perjury under the laws of the United States of America and under the laws of the State of California.

Executed at 3412 Hillview Avenue, Palo Alto, being the premises and place of business of the Electric Power Research Institute:

8/29/97

Mark D. Fox

Mark D. Fox, Esq.

Subscribed and sworn before me this day:

August 29, 1997

Tamsen Helen Gagnon, Notary Public

TAMSEN HELEN GAGNON
COMM. # 1016243
Notary Public — California
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97-819

BWRVIP

BWR Vessel &
Internals Project _____ *Issue Management and Resolution*

MEMORANDUM

October 8, 1997

Document Control Desk
Nuclear Regulatory Commission
11555 Rockville Pike
Rockville, MD 20852-2738

Attention: C. E. Carpenter

Subject: BWRVIP Response to NRC Request for Additional Information on BWRVIP-18

Reference: Letter from C. E. Carpenter, Jr., (NRC) to Carl Terry (BWRVIP Vice-Chairman), "Proprietary Request for Additional Information - Review of BWR Vessel and Internals Project Proprietary Reports, BWR Core Spray Internals Inspection and Flaw Evaluation Guidelines (BWRVIP-18), and Core Spray Piping and Sparger Repair Design Criteria (BWRVIP-19)," dated January 22, 1997.

Enclosed are 10 copies of the BWRVIP response to the NRC Request for Additional Information (RAI) on the BWRVIP report "BWR Vessel and Internals Project, BWR Core Spray Internals Inspection and Flaw Evaluation Guidelines (BWRVIP-18)" transmitted by the NRC letter referenced above. The BWRVIP response to the NRC RAI on BWRVIP-19, also transmitted by the NRC letter referenced above, was previously submitted to the NRC on February 24, 1997.

The enclosed response repeats each of the individual items in the NRC RAI verbatim, followed by the BWRVIP response to that item.

As indicated in the referenced NRC letter, the enclosed information concerns a report that the NRC staff has found to be proprietary in nature. Therefore, the enclosed information is also proprietary and should be withheld from public disclosure.

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in accordance with the Freedom of Information
Act, exemptions 4
FOIA- 2000-0186
9710090293/PDR

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If you have any questions on this subject please contact Dana Covill of GPU Nuclear (BWRVIP Assessment Committee Technical Chairman) at (201) 316-7525.

Sincerely,

Vaughn Wagoner

Vaughn Wagoner
Technical Chairman
BWRVIP Integration Committee
Carolina Power & Light Company

c: Dana Covill, GPU Nuclear, BWRVIP Assessment Committee Technical Chairman

97-740

BWRVIP

BWR Vessel &
Internals Project _____ *Issue Management and Resolution*

September 15, 1997

Document Control Desk
U. S. Nuclear Regulatory Commission
11555 Rockville Pike
Rockville, MD 20852

Attention: C.E. Carpenter

Subject: "BWR Vessel and Internals Project, BWR Shroud Support Inspection and Flaw Evaluation Guidelines (BWRVIP-38)," EPRI Report TR-108823, September 1997

Enclosed are ten (10) copies of the proprietary document "BWR Vessel and Internals Project, BWR Shroud Support Inspection and Flaw Evaluation Guidelines (BWRVIP-38)," EPRI Report TR-108823, September 1997. Also enclosed are ten (10) copies of the non-proprietary version of the report. This document is being submitted as a means of exchanging information with the NRC for the purpose of supporting generic regulatory improvements related to the inspection and flaw evaluation of BWR shroud supports.

The enclosed report is being submitted on behalf of the BWRVIP utilities. However, this submittal should not be interpreted as a commitment by all BWRVIP members to a specific course of action. Each member must formally endorse the BWRVIP position for it to become that member's position.

Please note that the enclosed document contains proprietary information. Therefore, a letter requesting the report be withheld from public disclosure and an affidavit describing the basis for withholding this information are provided as Attachment 1.

If you have any questions on this subject please call Vaughn Wagoner of Carolina Power & Light, BWRVIP Integration Committee Technical Chairman, at (919) 546-7959 or Warren Bilanin of EPRI, Manager of BWRVIP, at (650) 855-2340.

Sincerely,

Information in this record was deleted in accordance with the Freedom of Information Act, exemptions 4
FOIA 2000-0186

Carl Terry
Niagara Mohawk Power Company
Chairman, BWR Vessel and Internals Project

9709170225 970915
PDR TOPRP EMV*****
C PDR

Reply To: Carl Terry, BWRVIP Chairman, Niagara Mohawk Power Company, P. O. Box 63, Lycoming, NY 13093 • Phone: (315) 349-7263 • Fax: (315) 349-4753

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EPRI
Electric Power
Research Institute *Leadership in Science and Technology*

August 7, 1997

Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555-2738

Subject: TR-108823 "BWRVIP Vessel and Internals Project: BWR Shroud Support Inspection and Flaw Evaluation Guidelines (BWRVIP-38)", September, 1997

Gentlemen:

This a request under 10CFR2.790(a)(4) that the NRC withhold from public disclosure the proprietary document identified above (the "Report"). Two copies of the Report and the affidavit in support of this request are enclosed.

EPRI desires to disclose the Report to the NRC as a means of exchanging information with the NRC for the purpose of supporting generic regulatory improvements related to repair of the subject reactor components. EPRI would welcome any discussions between EPRI and the NRC related to the Report that the NRC desires to conduct.

The Report is for the NRC's internal use and may be used only for the purpose for which it is disclosed by EPRI. The Report should not be otherwise used or disclosed to any person outside the NRC without prior written permission from EPRI.

Headquarters: 3412 Hillview Avenue, Post Office Box 10412, Palo Alto, CA 94303, USA • (415)855-2000 • Telex: 82977 EPRI UF • Fax: (415) 855-1026
Washington Office: 2000 L Street, NW, Suite 805, Washington, DC 20036, USA • (202) 872-9222 • Fax: (202) 296-6040

9709170227 970915
PDR TOPRP EMV*****
C PDR

If you have any questions about the legal aspects of this request for withholding, please do not hesitate to contact me at (415) 855-2845. Questions on the contents of the Report should be directed to Warren Bilanin of EPRI at (415) 855-2340.

Sincerely,

Mark D. Fox
Intellectual Property Attorney
Intellectual Property Department

Enclosures

RE: TR-108823 "BWRVIP Vessel and Internals Project: BWR Shroud Support Inspection and Flaw Evaluation Guidelines (BWRVIP-38)", September, 1997

I. MARK D. FOX, being duly sworn, depose and state as follows:

I am an attorney at the Electric Power Research Institute ("EPRI") and I have been specifically delegated responsibility for reviewing the report listed above that is sought under this affidavit to be withheld (the "Report") and authorized to apply for its withholding on behalf of EPRI. This affidavit is submitted to the Nuclear Regulatory Commission ("NRC") pursuant to 10 CFR 2.790(a)(4) based on the fact that the Report consists of trade secrets of EPRI and that the NRC will receive the Report from EPRI under privilege and in confidence.

The basis for which the Report should be withheld from the public is set forth below:

(i) The Report has been held in confidence by EPRI, its owner. All those accepting the Report must agree to preserve the confidentiality of the Report.

(ii) The Report is of a type customarily held in confidence by EPRI and there is a rational basis therefor. The Report is trade secrets and is held in confidence by EPRI because to disclose it would prevent EPRI from licensing the Report at fees which would allow EPRI to recover its investment. If consultants and other businesses providing services in the nuclear power industry were able to publicly obtain the Report, they would be able to use it commercially for profit and avoid spending the large amount of money that EPRI was required to spend to prepare the Report. The rational basis that EPRI has for classifying the Report as trade secrets is the Uniform Trade Secrets Act which California adopted in 1984 and which has been adopted by over twenty states. The Uniform Trade Secrets Act defines a "trade secret" as follows:

> "Trade secret" means information, including a formula, pattern, compilation, program, device, method, technique, or process, that:

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(1) Derives independent economic value, actual or potential, from not being generally known to the public or to other persons who can obtain economic value from its disclosure or use; and

(2) Is the subject of efforts that are reasonable under the circumstances to maintain its secrecy.

(iii) The Report will be transmitted to the NRC in confidence.

(iv) The Report is not available in public sources. EPRI developed the Report only after making a determination that the Report was not available from public sources. It required a large expenditure of dollars for EPRI to develop the Report. In addition, EPRI was required to use a large amount of time of EPRI employees. The money spent, plus the value of EPRI's staff time in preparing the Report, show that the Report is highly valuable to EPRI. Finally, the Report was developed only after a long period of effort of at least several months.

(v) A public disclosure of the Report would cause substantial harm to EPRI's competitive position and the ability of EPRI to license the Report both domestically and internationally. The Report can be properly acquired or duplicated by others only with an equivalent investment of time and effort.

I have read the foregoing and the matters stated therein are true and correct to the best of my knowledge, information and belief. I make this affidavit under penalty of perjury under the laws of the United States of America and under the laws of the State of California.

Executed at 3412 Hillview Avenue, Palo Alto, being the premises and place of business of the Electric Power Research Institute:

August 7, 1997

Mark D. Fox

Mark D. Fox

Subscribed and sworn before me this day: August 7, 1997

Tamsen Helen Gagnon, Notary Public

Tamsen Helen Gagnon, Notary Public

NNA1 TAMSEN HELEN GAGNON
COMM. # 1016243
Notary Public — California
SANTA CLARA COUNTY
My Comm. Expires FEB 5, 1998 NNA1

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

September 15, 1997

Carl Terry, BWRVIP Chairman
Niagara Mohawk Power Company
Post Office Box 63
Lycoming, NY 13093

SUBJECT: TRANSMITTAL OF NRC STAFF'S SAFETY EVALUATION OF THE BWR VESSEL AND INTERNALS PROJECT BWRVIP-07 REPORT (TAC NO. M94959)

Dear Mr. Terry,

By letter dated February 2, 1996, as supplemented by letters dated July 17, May 21, and October 21, 1996, and January 8, 1997, you submitted the Electric Power Research Institute (EPRI) Report TR-105747, "BWR Vessel and Internals Project, Guidelines for Reinspection of BWR Core Shrouds (BWRVIP-07)," for NRC staff review and approval. The BWRVIP-07 report provides a uniform industry approach to reinspection of the BWR core shroud that would ensure that the structural and functional integrity would be maintained for either a repaired or unrepaired BWR core shroud.

The NRC staff has reviewed the BWRVIP-07 report and finds, in the enclosed Safety Evaluation Report (SER), that the scope of the guidelines is comprehensive, covering circumferential, vertical, and ring segment welds, as well as repair components for both Category B and C plants. Industry experience has found that the inspection methodologies – enhanced visual and ultrasonic methods – are appropriate for finding cracking. In addition, the guidance is consistent with that in earlier staff-approved reports on the inspection and evaluation of core shrouds. The guidelines of BWRVIP-07 incorporate many of the recommendations in previous safety evaluations and also allow for revision on the basis of new data and lessons learned. The staff, therefore, finds much of the proposed guidance acceptable. However, the staff also finds the guidance complicated and difficult to implement and audit and some of the provisions not sufficiently conservative. The SE discusses a number of issues that should be resolved by the BWRVIP and presents acceptable alternatives.

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Act, exemptions 4
FOIA- 2000-0196

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F129

Please contact C. E. (Gene) Carpenter, Jr., of my staff at (301) 415-2169 if you have any further questions regarding this subject.

Sincerely,

Jack Strosnider
for Gus C. Lainas, Acting Director
Division of Engineering
Office of Nuclear Reactor Regulation

Enclosures: as stated
cc: See next page

Please contact C. E. (Gene) Carpenter, Jr., of my staff at (301) 415-2169 if you have any further questions regarding this subject.

Sincerely,

Gus C. Lainas, Acting Director
Division of Engineering
Office of Nuclear Reactor Regulation

Enclosures: as stated
cc: See next page

SJCollins/FJMiraglia
RPZimmerman
TTMartin
GMHolahan/SFNewberry
LCShao/MEMayfield
TECollins/KAKavanagh
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* See Previous Concurrence

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Please contact C. E. (Gene) Carpenter, Jr., of my staff at (301) 415-2169 if you have any further questions regarding this subject.

Sincerely,

Brian W. Sheron, Director
Division of Engineering
Office of Nuclear Reactor Regulation

Enclosures: as stated
cc: See next page

SJCollins/FJMiraglia
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U.S. NUCLEAR REGULATORY COMMISSION

OFFICE OF NUCLEAR REACTOR REGULATION

DIVISION OF ENGINEERING

SAFETY EVALUATION OF EPRI TOPICAL REPORT TR-105747

"BWR VESSEL AND INTERNALS PROJECT, GUIDELINES FOR

REINSPECTION OF BWR CORE SHROUDS (BWRVIP-07)"

1.0 INTRODUCTION

1.1 Background

By letter dated February 2, 1996, the Boiling Water Reactor Vessel and Internals Project (BWRVIP) submitted the Electric Power Research Institute (EPRI) Topical Report TR-105747, "BWR Vessel and Internals Project, Guidelines for Reinspection of BWR Core Shrouds" (BWRVIP-07, Ref. 1), for staff review and approval. The BWRVIP clarified the content of BWRVIP-07 with letters dated July 17, 1996 (Ref. 2), and May 21, 1996 (Ref. 3), and supplemented the content of the guidelines with additional information on October 21, 1996 (Ref. 4). The BWRVIP further modified the guidelines in BWRVIP-07 by letter dated January 8, 1997 (Ref. 5).

The objective of the BWRVIP in submitting BWRVIP-07 was to provide a uniform industry approach to reinspection that would ensure the structural and functional integrity of the BWR core shroud whether repaired or unrepaired. This inspection criteria for unrepaired shrouds specify the time interval to the first required reinspection following the baseline inspection, as well as the welds to be inspected, the extent of the inspection, and intervals for subsequent reinspections. For repaired shrouds, the inspection criteria addresses reinspections of both the core shroud and the repair components. The guidelines are meant to be a "living" document and are considered subject to revision as the industry gathers more data after completing baseline inspections.

The General Electric Company (GE) addressed the flaw acceptance criteria for circumferential welds in unrepaired shrouds in "BWR Core Shroud Inspection and Flaw Evaluation Guidelines," GENE-523-113-0894, September 2, 1994 (Ref. 6) and "BWR Core Shroud Inspection and Flaw Evaluation Guidelines," GENE-523-113-0894, Rev. 1, March 1995 (Ref. 7). The staff's safety evaluations (SEs) for these documents were issued on December 28, 1994, and June 16, 1995, respectively (Refs. 8 and 9).

The BWRVIP-07 guidelines allow for plant-specific analyses to be performed for a given weld or for the entire shroud. These plant-specific analyses are not addressed in the scope of this report, and NRC approval must be obtained on a case-by-case basis. The staff review of flaw analyses performed according to the more conservative "BWR Core Shroud Inspection and Flaw Evaluation Guidelines," (GENE-523-113-0894, Revs. 0 and 1) will continue to be evaluated in accordance with the previous SEs. Flaw evaluations conducted according to the

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more comprehensive but less conservative guidelines of BWRVIP-07 will be evaluated by the staff and will consider the recommendations contained within this SE.

1.2 Purpose

The staff reviewed BWRVIP-07 to determine whether its guidance would ensure the structural integrity of the BWR core shroud.

1.3 Organization of this Report

Because BWRVIP-07 is proprietary, this SE was written so as not to repeat information contained in the report. The SE gives a brief summary of the general contents of the report (Section 2.0) and the detailed evaluation in Section 3.0. The SE does not discuss in any detail the provisions of the guidelines because they are proprietary, nor does it discuss the parts of the guidelines it finds acceptable. Recommendations of issues for the BWRVIP to resolve and conclusions are summarized in Section 4.0. The staff has structured its evaluation according to the organization of the BWRVIP-07 report.

2.0 SUMMARY OF TOPICAL REPORT

The BWRVIP-07 report addresses the following topics in the following order:

- Reinspection of BWRs with Unrepaired Core Shrouds

  — Category B plants
  — Category C plants

- Reinspection of BWRs with Repaired Core Shrouds

  — General requirements
  — Repair component inspections
  — Repair Anchorage Inspections
  — Horizontal and vertical core shroud weld inspections

The BWRVIP-07 report contains appendices on:

- Plant Categories for Core Shroud Inspections

- Generic Fracture Mechanics Analyses

- A Statistical Method for Estimating the Cracking in Inaccessible Regions of Core Shroud Welds

In the staff's evaluation, several terms in BWRVIP-07 are referenced in discussion. For clarification, these terms are defined or described below:

Category B plant: plants operated more than eight hot operating years with core shrouds made with Type 304L stainless steel (low carbon content) and average conductivity less than 0.3 μS/cm during the first five cycles of operation.

Category C plant: plants operated more than six hot operating years with core shroud made with Type 304 stainless steel (high carbon content) or plants operated more than eight hot operating years with core shrouds made of Type 304L stainless steel (low carbon content) and average conductivity greater than 0.3 μS/cm during the first five cycles of operation.

L_{min} : the minimum required length of uncracked weld, including two cycles of crack growth, to ensure the structural integrity of the weld.

L'_{min} : the minimum required length of uncracked weld, including an allowance of crack growth up to a maximum of 6 years, to ensure the structural integrity of the weld. L'_{min} is the sum of the minimum required length of sound weld (assuming inaccessible regions are 100 percent cracked) and the crack growth over the desired inspection interval (crack growth rate of 5×10^{-5} in/hr).

Option A : inspection strategy for category B plants that requires 100% inspection of all accessible areas of the shroud welds H3, H4, H5, and H7. If the cracking is found to be greater than 10 percent in any weld, then full inspections are required in accordance with the requirements for Category C plants.

Option B : inspection strategy for category B plants that allows inspection of a length of L'_{min} for the welds specified in Option A.[1]

t : the wall thickness

Table 1 : provides core shroud reinspection intervals for Category C plants. The reinspection interval is a function of the degree of cracking and the stress level in the weld.

Table 2 : provides a summary of the crack length assumption used in determining the reinspection intervals in Table 1.

3.0 STAFF EVALUATION

In general, the staff finds that the guidance presented in BWRVIP-07 ensures the structural integrity of the core shroud. The scope of the guidelines is comprehensive, covering circumferential, vertical, and ring segment welds, as well as repair components for both Category B and C plants. Industry experience shows that the inspection methodologies —

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[1] It should be noted that Options A and B vary slightly depending on which welds are being discussed. To more fully describe these Options would require publicly providing information that the NRC staff has determined to be proprietary in nature.

enhanced visual and ultrasonic methods — are appropriate for detecting cracking. In addition, the guidance is consistent with that in earlier staff-approved reports on the inspection and evaluation of core shrouds. The guidelines of BWRVIP-07 incorporate many of the recommendations in the referenced safety evaluations and also allow for revision on the basis of new data and lessons learned. The staff, therefore, finds much of the proposed guidance acceptable. However, the staff also finds the guidance complicated and difficult to implement and audit.

Sections 3.1 and 3.2 present the staff's detailed evaluation of the guidance regarding reinspections of BWRs with unrepaired and repaired core shrouds, respectively.

3.1 Reinspection of BWRs With Unrepaired Core Shrouds

Sections 3.1.1 and 3.1.2 discuss the staff's findings regarding the BWRVIP-07 guidelines for the reinspection of Category B and C plants with unrepaired core shrouds.

3.1.1 Category B Plants

For Category B plants with unrepaired core shrouds, the staff finds that the guidance is generally acceptable; however, several issues that the staff requests the BWRVIP to resolve are presented below.

Issue 3.1.1-1: Use of L_{min} to Determine Reinspection Intervals for Category B Shrouds

The guidelines state that if the limited inspections show that L_{min} is met, more than one cycle of operation may be justified. In earlier safety evaluations of core shroud cracking (Refs. 8 and 9), the staff required reinspection after just one cycle. The staff recommends that L_{min} be used to justify only one additional cycle of operation.

Issue 3.1.1-2: Use of Option A to Determine Inspection Intervals for Category B Core Shrouds

The staff considers that the recommended reinspection intervals of Option A[2] for unrepaired Category B core shrouds are not conservative.

The staff recommends the following modifications to BWRVIP-07:

(1) The inspection intervals (stated in years) should correlate reasonably well with cycle lengths.

(2) The 10-year inspection interval should be reduced to 6 years.

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[2] e.g., an inspection interval of 120 months if no cracking is found in the weld; 96 months if the weld cracking is less than or equal to 10 percent of the weld length inspected; and, if cracking is greater than 10 percent of the weld length inspected, then full inspections are required in accordance with the requirements for Category C plants

(3) The 8-year inspection interval should be reduced to 5 years.

These reductions reflect the rationale with regard to the staff's concerns about using Table 1 in BWRVIP-07 to determine inspection intervals. These concerns are addressed under Issue 3.1.2-3, below.

Issue 3.1.1-3: Use of Option B to Establish the Reinspection Intervals for Category B Core Shrouds

The staff's concerns about using the equation for L'_{min} given under the proposed Option B are as follows:

(1) In responding to items 6 and 7 in the staff's request for information (ref. 3), the BWRVIP stated that the assumption that the uninspected areas are fully cracked is conservative. Nonetheless, the responses still do not adequately address the frequency of crack initiations in the "unflawed region." The staff can accept the L'_{min} approach for a relatively short period of operation, (i.e., one cycle); however, the staff does not consider this approach adequate for the long term. Additional evaluation is needed from more rigorous inspections to justify longer intervals between inspections.

(2) The referenced documents do not justify the assumption that the inspection uncertainty is not to exceed 2t (t=wall thickness). Previous analyses have used other values, including 4t. All values used for inspection uncertainties should be demonstrated and technically justified.

(3) Unlike the equation for L_{min}, this equation does not include a factor of 2 to account for additional safety margin.

Because of these concerns, the staff does not consider it acceptable to use L_{min} and L'_{min} to justify more than one additional operating cycle. The staff therefore recommends that the BWRVIP revise the inspection strategies citing L_{min} and L'_{min} criteria so that they no longer justify multi-cycle operation without reinspection.

Issue 3.1.1-4 and 3.1.2-4: Inspection of Vertical Welds

The proposed reinspection of unrepaired core shrouds for both Category B and Category C plants is limited to horizontal (i.e. circumferential) welds only. This is consistent with the scope of the baseline inspection which is also limited to the inspection of circumferential welds. The reinspection guidelines do not require the inspection of the vertical welds even if the circumferential welds are found to be cracked. This appears to be based on the assumption that the core shroud will still function if the structural integrity of the flawed circumferential weld can be shown by fracture mechanics analysis and crack growth evaluation. However, the staff has concerns about the functionability of the shroud when a vertical weld has completely failed and an intersecting circumferential weld has extensively cracked. In such a case, the core shroud could open locally during postulated design-based accidents. This local opening of the shroud would create a large by-pass flow from inside the shroud which could cause inadequate core cooling. Therefore, the staff requests that BWRVIP-07 provide for inspecting the vertical welds to ensure their integrity when cracking of

the circumferential welds, as discussed above, is found. The provision should also be incorporated into the BWRVIP document that addresses the baseline inspection of core shrouds.

3.1.2 Category C Plants

The staff finds the reinspection guidance for Category C plants acceptable with the following exceptions:

Issue 3.1.2-1: Use of L_{min} to Determine the Reinspection Intervals for Category C Core Shrouds

The concern discussed under 3.1.1-1 of Section 3.1.1 of this SE also applies.

The guidelines stated that if a Category C plant cannot be inspected for L_{min}, the acceptability and inspection interval of the weld are to be determined on a case-by-case basis. The staff recommends that such determinations be submitted to the NRC for review and approval.

Issue 3.1.2-2: Use of the L'_{min} Equation to Determine the Reinspection Intervals for Category C Core Shrouds

This issue is covered by the discussion under Issue 3.1.1-3 of Section 3.1.1 of this SE.

In addition, the guidance given by the L'_{min} method and Table 1 (discussed below) may be inconsistent in terms of conservatism. For instance, for a shroud with 40 percent cracking, the table could require reinspection in about an operating cycle, whereas the L'_{min} method may not require reinspection before 6 years, i.e., potentially four or more operating cycles. The L'_{min} method requires much less inspection than that of the Table 1 method, but may allow a longer reinspection interval. The L'_{min} method is non-conservative relative to the Table 1 method.

Issue 3.1.2-3: Use of Table 1 to Determine Reinspection Intervals for Category C Core Shrouds

The staff has the following concerns about using Table 1 of BWRVIP-07 to determine the inspection intervals for Category C plants with unrepaired shrouds.

(1) The proposed 10-year reinspection interval for uncracked circumferential welds is too long because the welds would have been inspected only once since operation began. Moreover, the original inspections may not have been of a sufficiently high quality. Visual examinations could have missed some cracks if the surface was not properly prepared, or if the examination was not performed under optimal conditions of lighting and focal length. Similarly, original ultrasonic inspections (using what was then state-of-the-art equipment) may have missed some cracks because the equipment and techniques were not as good as they are now.

(2) The statistical model presented in BWRVIP-07 appears to be in error. The number of flaws assumed in Table 2, "Crack Length Assumptions Used To Determine

Reinspection Intervals," does not bound the industry-wide inspection results. The following tables present examples of this discrepancy:

Flaw Indications Reported in the Circumferential Welds of the Peach Bottom Unit 3 Core Shroud (Category C Plant, UT, Fall 1993)

Weld	Percent Inspected	Percent Cracked in Inspected Area	No of Flaws Observed	No. of Flaws Assumed in BWRVIP-07	Maximum Depth as a Percent of Wall Thickness
H1	84.5	4.8	7	7	36.5
H3	89.5	34.9	10	24	42.5
H4	89.2	32.2	36	24	50.0
H5	90.8	7.5	9	7	11.2

Flaw Indications Reported in the Circumferential Welds of the Susquehanna Unit 1 Core Shroud (Category B Plant, UT, Spring 1995)

Weld	Percent Inspected	Percent Cracked in Inspected Area	No. of Flaws Observed	No. of Flaws Assumed in BWRVIP-07	Maximum Depth as a Percent of Wall Thickness
H1	83.2	0.6	3	7	-
H2	83.2	7.9	29	7	-
H4	82.7	28.9	28	20	35
H5	72.7	29.4	27	20	32
H6A	83.9	1.8	8	7	12
H6B	84.2	10.2	19	14	33

The following conclusions can be deduced from a review of the above and the BWRVIP-07 report:

(a) The number of flaws reported in 7 of the 10 welds listed above exceeds the number assumed in BWRVIP-07. Therefore, it appears that the defect density assumed in the statistical model of BWRVIP-07 does not bound the actual inspection results to date with 95 percent confidence as suggested. According to the proposed methodology, the reinspection intervals would have been reduced if a larger number of flaws had been assumed for a weld. To resolve this discrepancy, a sensitivity study is needed

to show the effect of defect density on the proposed reinspection intervals presented in Table 1 of BWRVIP-07.

(b) Table 2 of BWRVIP-07 appears to contain an inconsistency between the amount of cracking assumed and amount of cracking actually reported. The staff requests that an explanation be provided of how, for example, 20 percent cracking observed in 55.3 inches of inspected weld can contain 14 flaws, whereas a statistical model for about the same length of an inaccessible weld assumes only 7 flaws with a 95 percent confidence level.

(c) In the staff's SE dated December 28, 1994 (Ref. 8), the staff recommended that licensees inspect the core shrouds in Category C plants during each refueling outage because these plants are highly susceptible to IGSCC. However, the staff allowed that licensees could reduce the reinspection frequency when the inspection findings support such a reduction. Table 1 of BWRVIP-07 proposes such a reduction; however, the staff finds that additional reinspection data are needed to permit confident prediction of the crack initiation and growth. Actual inspection data from at least two or three reinspections of the core shrouds ranked as Category C[3] are needed to permit evaluation of the reinspection intervals in Table 1 to determine the conservatism of the proposed flaw density.

(d) The input to the fracture mechanics analysis still requires careful review to determine its acceptability for generic application. Specifically, the staff has to review the proposed flaw density in inspected and uninspected regions, the assumed defect rate in inaccessible regions, and the use of the Distributed Ligament Length (DLL) computer code for calculating L_{min}.

The staff also has to review the assumption on which the BWRVIP-07 report's Table 3 is predicated. Data in Table 3, "Defect Rates in Inaccessible Regions of Core Shroud Welds," are derived from a probabilistic estimate of the amount of cracking that could occur in the inaccessible region. The analysis assumes that the mean defect rate in the inaccessible region equals that observed in the inspected region and calculates a defect rate that (based on a 95-percent probability) would not be exceeded. This assumption that every region of the weld is equally susceptible to cracking may not be entirely valid but may be acceptable given other margins in the analysis. Moreover, in its response to the staff's RAI , the BWRVIP stated that the number of flaws assumed in the fracture mechanics analyses was based on the UT results of a BWR-4 core shroud inspection. The staff will have to determine the validity of this statement in light of new data from recent inspections.

(e) BWRVIP-07 contains the findings of a new crack initiation study that supported a reinspection interval from 8.4 years to 8.9 years for cracking in no more than 40 percent of the weld circumference. However, this study assumed a crack growth rate

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[3] Includes those core shrouds originally ranked in Category B grouping that have been reclassified to Category C status as a result of inspections that revealed cracking in excess of 10 percent.

of 2×10^{-5} inch per hour. The staff has this crack growth rate under review. Until the staff can draw a conclusion regarding the acceptability of the proposed growth rate, the bounding crack growth rate currently accepted for fracture mechanics analyses is 5×10^{-5} inch per hour.

In light of the concerns about using Table 1 to predict the inspection intervals for Category C core shrouds and the time required to resolve them, the staff has developed an alternative (shown as Table A to this SE) for interim use in determining reinspection intervals for circumferential welds.

3.1.3 Acceptable Alternative to Table 1 of BWRVIP-07 for Interim Use in Determining Reinspection Intervals (Table A)

The staff's alternative (Table A) is acceptable if the circumferential welds have been inspected and determined to have a remaining uncracked ligament in excess of the minimum uncracked ligament. In addition, welds must have been inspected for at least 50 percent of the weld circumference.

The inspection frequencies given below may be used for the longer term. The staff has simplified and modified the criteria in Table 1 of BWRVIP-07 to make them sufficiently conservative, and have capped the reinspection interval at 6 years.

(1) For welds with no relevant crack indications found in the inspected length, the reinspection interval should not exceed 6 years or 4 fuel cycles, whichever comes first.

(2) For welds with cracking in no more than 10 percent of the inspected length, the reinspection interval should not exceed 5 years or 3 fuel cycles, whichever comes first.

(3) For welds with cracking in more than 10 percent and no more than 25 percent of the inspected length, the reinspection interval should not exceed 4 years or 2 fuel cycles, whichever comes first.

(4) For welds with cracking in more than 25 percent but not more than 40 percent of the inspected length, the reinspection should be performed during the next refueling outage.

(5) For welds with cracking in excess of 40 percent of the inspected length, a plant-specific analysis is required to support continued operation for one fuel cycle without repair.

3.2 Reinspection of Boiling-Water Reactors with Repaired Core Shrouds

With regard to the proposed guidelines for repaired core shrouds and the components used in the repair modifications, the staff finds the general guidance acceptable.

Sections 3.2.1 through 3.2.4 discusses the staff's concerns regarding the BWRVIP-07 guidelines for reinspection of components used in core shroud repair modifications, repair anchorages, horizontal and vertical welds in repaired core shrouds, and ring segment welds in repaired core shrouds, respectively.

3.2.1 Reinspection of Repair Components

The following items summarize the staff's concerns about the proposed guidance on reinspections of components used in repairs of shrouds :

Issue 3.2.1-1: Recommended Generic Criteria Regarding Reinspection of Core Shroud Repair Components

Recent industry experience suggests that the generic recommendations presented in BWRVIP-07 may not be adequate for reinspection of components used in core shroud repair modifications. For example, a licensee found that a repair, which was in place for only one cycle, had a failed retainer clip on one assembly and a loose nut on another. After finding the degradation, the licensee examined all the assemblies. In contrast, the guidance in BWRVIP-07 does not provide for expanding the inspection of the sample if degradation of the repair components is discovered. The staff recommends including such a provision in the guidelines of BWRVIP-07.

If that licensee had followed the guidelines and had selected a different assembly for sampling, it could have missed this degradation. The staff requests that licensees provide a technical justification for not performing a 100-percent inspection after one cycle. The staff recommends requiring such an inspection to discover installation errors and other problems.

It is not clear whether following the provisions of the guidelines would have revealed the loose nut on the degraded repair described above. The staff recommends that the guidelines specify ways of verifying bolt tightness when visual examination is not feasible or adequate.

Issue 3.2.1-2: Recommendation to Review Current Set of Assumptions Used in Analyses of Core Shroud Repair Modifications

The staff recommends that the guidelines specify reviewing the design assumptions for the repair. For example, repairs may be installed preemptively. Such repairs may be designed assuming that welds are fully cracked or by assuming that a minimum ligament is present. The staff recommends a pre-repair inspection for all future repairs to ensure the minimum ligaments assumed actually are present.

In light of the above concerns, the staff is recommending the alternative below and summarized in Table B of this SE. The alternative specifies more inspections early in the life of the repair to ensure that the repair is performing as designed.

Recommended Alternative for Reinspections of Components in Core Shroud Repair Designs

(1) After installing a repair, licensees should reinspect all repair components during the subsequent refueling outage. If any repair component is found to be degraded, the

licensee should perform reinspections of the same scope during the successive refueling outages until no degradation is found. (Degradation includes missing or degraded parts, misalignment of repair parts, loosening or relaxation of locking devices and deviation of critical gaps or contact areas.)

(2) After successfully completing the inspections in Item (1) above, the licensee may perform a 25 percent reinspection of the repair components during each of the next four refueling outages. If these inspections reveal degradation, the licensee should expand the inspection to 100 percent of the repair components, and inspect in accordance with the reinspection scope prescribed in Item (1) above.

(3) After successfully completing the inspections of the repair components as prescribed in Item (2), the licensee may perform a 25 percent inspection of the repair components every other cycle. If these inspections reveal degradation, the licensee should expand the sample to 50 percent of the repair components, and the scope of inspection in subsequent cycles should follow the scope prescribed in Item (1). If these subsequent inspections also reveal degradation in the expanded sample, the licensee should further expand the sample to 100 percent.

3.2.2 Reinspection of Repair Anchorages

The inspection guidelines for repair anchorages are generally acceptable. Specific issues regarding the provisions for reinspection of repair anchorages are summarized in Issues 3.2.2-1, 3.2.2.-2, 3.2.2-3 and 3.2.2-4, below.

Issue 3.2.2-1: Justification for Omitting Pre-Inspections of Components Used for the Core Shroud Repair Anchorages.

The guidelines state that, in the absence of a pre-repair inspection, a licensee should perform an inspection in the first operating cycle after the repair is completed. However, the guidelines did not justify not performing a pre-repair inspection. As discussed in Section 3.2.1, the staff recommends that licensees conduct a pre-repair inspection for all future repairs and then reinspect 100 percent of the repair in the subsequent outage. Welds H8 and H9 should be included in this inspection to ensure that all the repair design assumptions are met.

Issue 3.2.2-2: Recommended Practices for Reinspections of Gussets, Brackets and Repair Anchorages

In addition to the guidelines in BWRVIP-07, the BWRVIP is currently preparing another document to address inspection requirements for welds H8 and H9, as well as the shroud support structure (which includes gussets and brackets). The staff recommends that licensees reinspect the repair anchorages and set the reinspection intervals according to those of the repair components, because the anchorages must retain their integrity for the repair components to be effective. The staff recommends that it would be preferable to address welds in anchorages associated with the repair in the context of these BWRVIP-07 guidelines (rather than in a separate document). An alternative acceptable to the staff is given above under Section 3.2.1 of this SE and summarized in Table B to this SE. Licensees

can also take credit toward the required frequency for inspections done according to other NRC-approved documents.

Issue 3.2.2-3: Recommendations for Technical Analyses to Verify the Effect of Cracking on Gusset Stiffness

The guidelines state that minor cracking in a gusset weld would not have any impact on the effectiveness of a shroud repair and that the stiffness of the gusset is not significantly affected if there is no gross through-thickness cracking. This statement needs to be supported by technical analysis. The staff recommends that the guidelines of BWRVIP-07 be amended to recommend that technical analyses be performed to verify the validity of this assumption if minor cracking is detected in a gusset weld.

Issue 3.2.2-4: BWRVIP-07 Recommended Intervals for Reinspection of BWR Core Shroud Repair Assemblies

Repairs currently made by the industry typically consist of 4 to 10 assemblies. According to the guidelines, 10 years can pass before an assembly is reinspected. The staff considers this interval too long. Currently, not enough operating data regarding repaired shrouds has been collected to justify postponing reinspections of the repair assemblies for 10 years. The staff recommends that the reinspection schedule for repair anchorages be set equivalent to the intervals recommended in the staff's alternative in Section 3.2.1 of this SE unless the longer intervals can be technically justified. The staff also recommends adding a provision requiring sample expansion similar to that recommended in Issue 3.2.1-2 of Section 3.2.1 to this SE if degradation is found.

3.2.3 Reinspection of Core Shroud Horizontal and Vertical Welds

For horizontal welds structurally replaced by the repair, the staff concludes that BWRVIP-07 provides acceptable guidance. As previously approved in the staff's SE dated September 29, 1994 (Ref. 13), the horizontal welds structurally replaced by the shroud repair do not need to be reinspected. Those not replaced by the repair will be reinspected consistent with the requirements for an unrepaired shroud (Ref. 13). The staff's evaluation for unrepaired horizontal welds has previously been discussed in Section 3.1 to this SE. The staff has some concerns, however, about the guidelines proposed for vertical shroud welds as discussed below.

Issue 3.2.3-1: General Issue Regarding the Proposed Guidelines for Reinspection of Vertical Welds in Repaired Shrouds

The staff recommends reevaluating vertical welds needed to ensure structural integrity to confirm that the design assumptions remain valid. In the past, it was not considered necessary to inspect vertical welds, and such inspection was not required. Since then, some licensees have made repairs (some of them preemptive), without inspecting for L_{min} in vertical welds.

Cracking is now being reported in vertical welds with increasing frequency. Operating experience has now shown that a combination of residual stresses, fluence, and environment

can cause extensive cracking in vertical welds. These welds warrant inspections that are more extensive than those proposed. The staff recommends that licensees perform baseline inspections of the vertical welds (100 percent of accessible areas, at least 50 percent of the total weld length) when the repair is made or at the next outage if the repair has already been made. This recommendation replaces the existing guidelines which specify using Option A or Option B.

Issue 3.2.3-2: Use of Option A or Option B for Reinspections of Vertical Welds in Repaired Core Shrouds

Option A for vertical welds is acceptable except for its provision regarding the reinspection interval determined by L'_{min}. As stated earlier in this evaluation, the staff recommends that the L'_{min} method be used to support additional operating cycle.

In contrast, the staff considers Option B inadequate. The staff recommends that licensees successfully complete a baseline inspection and a 25 percent sampling of the total vertical weld length for every cycle. Welds should be inspected from both the ID *and* OD or volumetrically. Recent industry experience confirms the need for this type of inspection, having shown that cracks initiate on either ID or OD and do not extend through the wall.

The staff recommends that, in addition to the provisions of the guidelines regarding sample expansion, licensees inspect the same location of some other vertical welds at the same elevation.

The staff also recommends that licensees' samples include the vertical welds near the beltline exposed to high fluence. In selecting welds for reinspection, licensees should emphasize welds with high susceptibility to IGSCC or irradiation-assisted stress corrosion cracking (IASCC).

3.2.4 Reinspection of Ring Segment Welds

The staff has determined that the provisions in BWRVIP-07 for reinspections of core shroud ring segment welds are generally acceptable. However, the staff has some concerns as discussed under Issues 3.2.4-1 and 3.2.4-2, below.

Issue 3.2.4-1: Use of Option A for Reinspections of Ring Segment Welds

Option A for ring segment welds is based on an L'_{min} approach to determine inspection scope and intervals, as distinguished from the definition in Section 2.0, above. Option A is acceptable except, as discussed earlier, it should be used to support continued operation for only one additional fuel cycle without repair. The staff has not declared the crack growth rate to be acceptable as referenced (Ref. 11); it is still under review. The bounding crack growth rate of 5×10^{-5} in/hr should be used in flaw evaluations.

Issue 3.2.4-2: Use of Option B for Reinspections of Ring Segment Welds

The staff's evaluations of the methods have been discussed in Section 3.1.1 of this report.

Very little data exist regarding the integrity of ring segment welds. Therefore, the staff recommends that, to acquire baseline data, licensees should inspect all ring segment weld surfaces, including heat affected zones, unless there is adequate technical justification for not inspecting them. The inspection method should be capable of detecting flaws initiating from all surfaces (e.g., the OD, ID, bottom surface, and top surfaces), and should not be limited to inspecting just the OD or ID.

The staff considers that Option B requires examining too small a sample over too long a period. The accessible areas of these welds are very limited. Therefore, it is desirable that licensees apply this sampling only after performing a baseline inspection of all accessible regions at each ring segment weld, with the total inspected length at each weld encompassing at least 50 percent of the weld length and then sampling 25-percent of the total number of ring segment welds every cycle.

Finally, the staff recommends that licensees' samples include the ring segment welds near the beltline exposed to high fluence. In selecting welds for reinspection, licensees should emphasize welds with high susceptibility to IGSCC or IASCC.

4.0 RECOMMENDATIONS AND CONCLUSIONS

As previously stated, the staff finds much of the guidance in BWRVIP-07 acceptable. It is consistent with guidance in previous submittals and incorporates recommendations in previous safety evaluations. However, the staff also finds the guidance complicated and not sufficiently conservative. The staff recommends that the BWRVIP resolve the issues raised in this SE. These issues, discussed in Sections 3.1.1, 3.1.2, 3.2.1, 3.2.2, 3.2.3, and 3.2.4 of this SE, are restated (in bullet form) in Sections 4.1, 4.2, 4.3, 4.4, and 4.5 below for clarity.

4.1 Guidelines for Unrepaired Core Shrouds in Category B and C Plants

Issues 3.1.1-1, 3.1.1-3, 3.1.2-1, and 3.1.2-2

- Inspection strategies that cite L_{min} and L'_{min} to justify multi-cycle operation (for Category B and C plants) without inspection should be revised to justify operation for only one cycle.

Issue 3.1.1-2

- The 10-year reinspection interval used in Option A for Category B plants should be reduced to 6 years and the 8-year reinspection interval should be reduced to 5 years.
- The inspection intervals (stated in years) should correspond reasonably well to operating cycle lengths.

Issues 3.1.1-4 and 3.1.2-4

- The reinspection guidelines do not require the inspection of vertical welds even when inspection reveals cracking of circumferential welds. The staff has concerns about the functionability of the shroud when a vertical weld has completely failed and the

circumferential weld intersecting it has extensively cracked. The staff requests that BWRVIP-07 provide for inspecting vertical welds to assure their integrity when cracking in the intersecting circumferential welds is found. The provision should also be incorporated into the relevant BWRVIP document for baseline inspection of the core shrouds.

Issue 3.1.2-1

- The guidelines in BWRVIP-07 state that if a licensee cannot inspect Category C core shroud for L_{min}, the acceptability and inspection interval of the weld are to be determined on a case-by-case basis. The evaluation addressing these issues should be submitted for NRC review and approval.

Issue 3.1.2-3

- Issue 3.1.2-3 lists a number of concerns about Table 1 that should be resolved. The staff has provided an acceptable alternative to Table 1 of the BWRVIP-07 guidelines in its summary of Issue 3.1.2-3. This alternative is also summarized in Table A of this SE.

4.2 Guidelines for Reinspection of Components in Repaired Core Shrouds

Issues 3.2.1-1 and 3.2.1-2

The staff recommends incorporating the following changes into the existing BWRVIP-07 guidelines:

- Require a 100-percent inspection after one cycle to find installation errors.
- Specify other ways of verifying bolt tightness where visual examination is not feasible or adequate.
- Require expanding the sample if degradation is found in any of the repair components.
- Specify a review of the design assumptions regarding the repair.

Having considered the need for baseline data on how repairs are performing, and in view of recent operating experience, the staff finds that the scope and frequencies proposed in BWRVIP-07 are inadequate. Consequently, the staff recommends an alternative for reinspecting repair components, as discussed in Section 3.2.1 of this SE and summarized in Table B of this SE.

4.2 Guidelines for Reinspection of Repair Anchorages

Issues 3.2.2-1, 3.2.2-2, 3.2.2-3 and 3.2.2-4

The staff recommends incorporating the following changes into the existing BWRVIP-07 guidelines:

- Specify a pre-repair inspection for all future repairs and reinspection of 100 percent of the repair components after the first operating cycle following the installation of the repair modification.

- Address reinspections of the welds in anchorages associated with the repair in these BWRVIP-07 guidelines (rather than in a separate document). An alternative acceptable to the staff in scope and frequency is given under Section 3.2.1 and summarized in Table B of this SE.

- Require that, if minor cracking is detected in a gusset weld, technical analyses be performed to verify the assumption that the cracking at the gusset weld would not have any impact on the effectiveness of the shroud repair.

- Reduce the interval between inspections of the anchorages unless the longer interval can be technically justified.

- Add a provision requiring that licensees expand the sample if degradation is found.

4.4 Guidelines for Reinspection of Horizontal and Vertical Welds in Repaired Core Shrouds

Issue 3.2.3-1:

The staff recommends incorporating the following changes into the existing BWRVIP-07 guidelines:

- State that vertical welds needed for structural integrity be reevaluated to confirm that the original design assumptions remain valid.

- State that a baseline inspection (100 percent of accessible areas, at least 50 percent of the total weld length) of the vertical welds should be performed when the repair is installed, or at the next outage if the repair has already been made. (This inspection should be specified in place of those specified by Options A and B.)

Issue 3.2.3-2:

The staff recommends incorporating the following changes into the existing BWRVIP-07 guidelines:

- Specify that, for Option A for vertical welds, the inspection intervals determined from the L'_{min} methods prescribed in Section 4.4.1 of BWRVIP-07 be revised to justify only one additional cycle of operation.

- Require licensees to inspect 25 percent of total vertical weld length every cycle after the baseline inspections of the vertical welds are complete.

- Specify inspecting welds visually from both the ID *and* OD, or volumetrically.

- State that in expanding the vertical weld inspection sample, licensees should inspect the same location of some other vertical welds at the same elevation.

- State that licensees' sampling should include the vertical welds near the beltline which are exposed to high fluence.

4.5 Guidelines for Reinspection of Ring Segment Welds in Repaired Core Shrouds

Issue 3.2.4-1:

The staff recommends incorporating the following changes into the existing BWRVIP-07 guidelines:

- The NRC approved crack growth rate of 5 X 10^{-5} inches/hr should be specified in lieu of the growth rate specified in EPRI Topical Report TR-105873 (Ref. 11) .

Issue 3.2.4-2:

The staff recommends incorporating the following changes into the existing BWRVIP-07 guidelines:

- Use the staff's recommendations regarding the calculation methods for the unrepaired core shrouds for Category B plants.

- State that the inspection method should be capable of detecting flaws initiated from all surfaces (the OD, ID, bottom surface, and top surface), and not be limited to just the OD or ID.

- State that, after successfully completing a baseline inspection (100 percent of accessible areas, at least 50 percent of the weld length), licensees may apply a 25-percent sampling of the total number of ring segment welds during each subsequent refueling outage .

- State that licensees' sampling should include the ring segment welds near the beltline exposed to high fluence.

In addition, the staff recommends that the slowest crack growth rate used for analyses should be 5 x 10^{-5} inch per hour (the crack growth rate currently approved by the NRC).

5.0 BIBLIOGRAPHY AND REFERENCES

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2. July 17, 1996 — Letter from V. Wagoner, Chairman — BWRVIP Integration Committee, to C. Carpenter, U.S. Nuclear Regulatory Commission: "Clarification to Core Shroud Reinspection Guidelines."

3. May 21, 1996 — Letter from C.E. Carpenter, Lead Project Manager, U.S. Nuclear Regulatory Commission, to J.T. Beckham, Jr., Chairman — BWRVIP, submitting "Request for Additional Information - Review of BWR Vessel and Internals Project Proprietary Report, 'BWR Vessel and Internals Project, Guidelines for Reinspection of BWR Core Shrouds (BWRVIP-07).'"

4. October 21, 1996 — Letter from Larry Stewart, Acting on behalf of Robin Dyle, BWRVIP Technical Chairman, to the U.S. Nuclear Regulatory Document Control Desk, submitting the "BWRVIP Response to Request for Additional Information Regarding Proprietary Topical Report EPRI TR-105747."

5. January 8, 1997 — Letter from W. Bilanin, Acting on behalf of Robin Dyle, BWRVIP Technical Chairman, to the U.S. Nuclear Regulatory Document Control Desk, submitting the "Modification to BWRVIP Response to NRC Request for Additional Information on BWRVIP-07."

6. September 2, 1994 — General Electric Company Topical Report No. GENE-523-113-0894, "BWR Core Shroud Inspection and Evaluation Guidelines."

7. March 1995 — General Electric Company Topical Report No. GENE-523-113-0894, Rev. 1, "BWR Core Shroud Inspection and Flaw Evaluation Guidelines."

8. December 28, 1994 — Letter from B.W. Sheron, Director —Division of Engineering, U.S. Nuclear Regulatory Commission, to J.T. Beckham, Chairman — BWRVIP: "Evaluation of 'BWR Shroud Cracking Generic Safety Assessment', Rev. 1, GENE-523-A107P-0794, August 5, 1994 and 'BWR Core Shroud Inspection and Evaluation Guidelines,' GENE-523-113-0894, September 2, 1994."

9. June 16, 1995 — Letter from B.W. Sheron, Director — Division of Engineering, U.S. Nuclear Regulatory Commission, to J.T. Beckham, Chairman — BWRVIP: "Evaluation of 'BWR Core Shroud Inspection and Evaluation Guidelines,' GENE-523-113-0894, Rev. 1, dated March 1995, and 'BWRVIP Core Shroud NDE Uncertainty and Procedure Standard,' dated November 22, 1994."

10. October 1995 — Electric Power Research Institute Topical Report No. TR-105692, "Guide for Format and Content of Core Shroud Repair Design Submittals (BWRVIP-04)."

11. November 1995 — Electric Power Research Institute Topical Report No. TR-105873, 2nd Draft, "Evaluation of Crack Growth in BWR Stainless Steel RPV Internals."

12. September 13, 1994 — Letter from J.T. Beckham, Chairman — BWRVIP, to

D.S. Brinkman, Senior Project Manager, U.S. Nuclear Regulatory Commission: "Response to NRC Request for Additional Information (RAI) Regarding Boiling Water Reactor (BWR) Core Shroud Repair Design Criteria," including Attachment 1, "Response to NRC Questions," and Attachment 2, "Revision 1 of the Core Shroud Repair Design Criteria."

13. September 29, 1994 — Letter from B.W. Sheron, Director — Division of Engineering, U.S. Nuclear Regulatory Commission, to B. McLeod, Chairman — BWRVIP Repair Technical Subcommittee: "Safety Evaluation on Boiling Water Reactor BWR Core Shroud Repair Design Criteria."

Table A: Staff Alternative to Table 1 of the BWRVIP-07 Guidelines

Cracking (%)	Inspection Interval
% = 0	6 yrs or 4 cycles, whichever comes first
10 ≥ % > 0	5 yrs or 3 cycles, whichever comes first
25 ≥ % > 10	4 yrs or 2 cycles, whichever comes first
40 ≥ % > 25	Next outage
% > 40	plant-specific analysis needed to operate 1 cycle without repair

Table B: Staff Alternative for Inspecting Repair Components and Anchorages

What	When	Frequency	Expansion
(1)Inspect to ensure all design assumptions are met	Pre-repair inspection for all future repairs	N/A	N/A
100% of repair components and anchorages	Inspect after completion of repair or after one cycle if repair already done.	If no degradation is found, go to (2)	If any degradation is found, inspect 100% each outage until no degradation is found
(2) 25% sampling	After (1) is successful.	Each cycle for 4 cycles. If no degradation is found, go to (3)	If degradation is found, expand to all repair components and follow (1)
(3) 25% sampling	After (2) is successful	Every other cycle	If degradation is found, expand inspections to 50% of the repair components. If further degradation is found, expand to 100%.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

August 14, 1997

Carl Terry, BWRVIP Chairman
Niagara Mohawk Power Company
Post Office Box 63
Lycoming, NY 13093

SUBJECT: TRANSMITTAL OF NRC STAFF'S INDEPENDENT ASSESSMENT OF THE BOILING WATER REACTOR VESSEL AND INTERNALS PROJECT BWRVIP-05 REPORT AND PROPRIETARY REQUEST FOR ADDITIONAL INFORMATION (TAC NO. M93925)

Dear Mr. Terry,

This letter transmits the staff's independent assessment of the frequency of BWR reactor pressure vessel failure (Enclosure 1), a proprietary request for additional information (Enclosure 2), and additional guidance on information the staff will need to assess requests for relief from the inspection schedule in 10 CFR 50.55a(g)(6)(ii)(A) (Enclosure 3). The staff's independent assessment was performed in response to the Boiling Water Reactor Vessel and Internals Project (BWRVIP) submittal of the proprietary report, "BWR Vessel and Internals Project, BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations (BWRVIP-05)[1]," dated September 28, 1995, as supplemented by letters dated June 24 and October 29, 1996, May 16, June 4, and June 13, 1997. The staff's assessment, a preliminary, multi-disciplined, risk-informed review of the BWRVIP-05 report, drew upon the resources of the Offices of Nuclear Reactor Regulation (NRR), Nuclear Regulatory Research (RES), Analysis and Evaluation of Operational Data (AEOD), and International Programs (OIP).

The staff has determined that, since the BWRVIP-05 report did not perform a risk-informed analysis, nor did it adequately consider beyond design basis transients (which appear to dominate the estimated frequency of BWR reactor vessel failure), the BWRVIP-05 report, as written and presently supplemented, is not sufficient for deferring permanently inspections of the circumferential welds. However, based on the staff's independent assessment, any significant RPV vulnerabilities would not be expected until relatively late in life of the reactor vessels when higher levels of embrittlement are reached. Based on this observation, the staff will consider plant-specific requests, if technically justified, to delay inspections of the circumferential welds for up to 40-months. Further, based on the staff's assessment, the requested additional information should be sufficient to determine if permanently deferring inspections of the RPV circumferential welds is acceptable.

---

(1) The BWRVIP-05 report proposed to reduce the scope of inspection of the BWR reactor pressure vessel (RPV) welds from essentially 100 percent of all RPV shell welds to 50 percent of the axial welds and zero percent of the circumferential welds. The October 29, 1996, letter modified the BWRVIP-05 original proposal to increase the examination of the axial welds to 100 percent (from 50 percent) while still proposing to inspect essentially zero percent of the circumferential RPV shell welds (except that the intersections of the axial and circumferential welds would have been included; approximately 2 - 3 percent of the circumferential welds would be inspected under this proposal).

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On May 12, 1997, the Commission was briefed by representatives of the BWRVIP and the staff on the issues related to the requirements for a full inspection of RPV shell welds. The Commission, in a Staff Requirements Memorandum (SRM M970512B), dated May 30, 1997, requested that the staff consider a tiered approach in gathering additional baseline information and/or implementing the Rule [10 CFR 50.55a(g)(6)(ii)(A)]. The SRM recommended that the staff's assessment (a) should address the BWRVIP proposal to examine 100 percent of the axial welds which would include examinations of some circumferential weld lengths near the intersections of the weld types to determine if this proposal could provide an appropriate level of sampling of the circumferential welds, (b) should provide a comprehensive evaluation of the probabilistic analysis contained in the BWRVIP proposed alternative in determining the acceptability of a proposed technical alternative and/or in pursuing changes to the rule, and, (c) should receive appropriate review, including review by the Advisory Committee on Reactor Safeguards (ACRS).

In SECY 97-152, dated July 18, 1997, the staff reported to the Commission its preliminary results of a broader risk informed assessment it had initiated and described its proposed course of action in response to the SRM. In particular, the staff reported that preliminary assessments indicated that beyond design basis transients, which had not been considered in the BWRVIP-05 submittal, appeared to dominate the estimated frequency of BWR reactor vessel failure. It was further noted that, although additional studies were necessary in this area, any significant vulnerabilities would not be expected until relatively late in life of the reactor vessels when higher levels of embrittlement are reached. Based on this latter observation, the staff indicated that plant specific requests to delay inspections of the circumferential welds would be considered, if technically justified, by the staff. On August 7, 1997, the staff issued Information Notice (IN) 97-63, "Status of NRC Staff's Review of BWRVIP-05," regarding licensee requests for relief.

Since issuance of SECY 97-152, the staff has completed additional analyses of the potential for BWR RPV failure considering beyond design basis events. These analyses were discussed with the BWR VIP and industry representatives during a meeting on August 8, 1997 and are documented in Enclosure 1. The additional work documented in Enclosure 1 supports the observations in SECY 97-152. However, these analyses are generic in nature and were performed utilizing more limited data than should be available to the industry. Therefore, the staff requests that the BWRVIP supplement its original BWRVIP-05 submittal with a broader, risk-informed evaluation of the proposed change in RPV weld inspection requirements.

Enclosure 2 is a request for additional information (RAI) that is intended to provide guidance on the areas that should be addressed in the industry's assessment. As indicated in SECY 97-152, the staff plans to complete its final evaluation of this issue within six months. Meeting this schedule is, of course, dependent upon receiving the industry's evaluation. In order to meet this schedule, we request that you provide the additional information requested in Enclosure 2 within 90 days. Please inform us within 30 days if

you can not meet this schedule. We are available for any meetings that would assist in facilitating this schedule.

With regard to plant specific relief requests, Enclosure 3 provides some additional guidance on what information the staff will need to assess requests for relief from the inspection schedule in 10 CFR 50.55a(g)(6)(ii)(A). As indicated in IN 97-63, the staff will consider delays in inspections of circumferential welds of up to 2 operating cycles or 40 months (whichever is longer). The acceptability of such requests will be based on plant-specific information submitted by the licensee that demonstrates that the expected frequency of beyond design basis events and the level of embrittlement of the RPV are low enough to assure low probability of vessel failure during the period of relief. It is the staff's expectation that granting relief for up to two operating cycles will allow sufficient time for the industry to complete its broader risk informed evaluation and for the NRC to reach a position on the acceptability of longer term relief from inspection of circumferential welds.

Also, a briefing of the ACRS subcommittee on this issue has been scheduled for August 26, 1997, which the industry has been invited to participate. A meeting with the main committee of the ACRS will be scheduled for early in 1998, after the industry has completed and submitted its risk informed evaluation, and the staff has completed its final Safety Evaluation.

Please contact C. E. (Gene) Carpenter, Jr., of my staff at (301) 415-2169 if you have any further questions regarding this subject.

Sincerely,
ORIGINAL SIGNED BY
for Brian W. Sheron, Director
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Enclosures: as stated
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R

INDEPENDENT ASSESSMENT BY THE OFFICE OF NUCLEAR REACTOR REGULATION RELATED TO THE REVIEW OF THE TOPICAL REPORT BY THE BOILING WATER REACTOR VESSEL AND INTERNALS PROJECT: "BWR REACTOR PRESSURE VESSEL SHELL WELD INSPECTION RECOMMENDATIONS" (BWRVIP-05)

Enclosure 1

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INDEPENDENT ASSESSMENT BY THE OFFICE OF NUCLEAR REACTOR REGULATION RELATED TO THE REVIEW OF THE TOPICAL REPORT BY THE BOILING WATER REACTOR VESSEL AND INTERNALS PROJECT: "BWR REACTOR PRESSURE VESSEL SHELL WELD INSPECTION RECOMMENDATIONS" (BWRVIP-05)

1.0 INTRODUCTION

By letter dated September 28, 1995, as supplemented by letters dated June 24 and October 29, 1996, and May 16, June 4, and June 13, 1997 (References 1, 2, 3, 4, 5, and 6, respectively), the Boiling Water Reactor Vessel and Internals Project (BWRVIP), a technical committee of the BWR Owners Group (BWROG), submitted the proprietary report, "BWR Vessel and Internals Project, BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations (BWRVIP-05)," which proposed to reduce the scope of inspection of the BWR reactor pressure vessel (RPV) welds from essentially 100 percent of all RPV shell welds to 50 percent of the axial welds and zero percent of the circumferential welds. Reference 2 provided supplemental information in response to two Nuclear Regulatory Commission (NRC) staff requests for additional information (RAIs) dated April 2 and May 20, 1996 (References 7 and 8). In Reference 3, the BWRVIP modified the BWRVIP-05 original proposal to increase the examination of the axial welds to 100 percent (from 50 percent) while still proposing to inspect essentially zero percent of the circumferential RPV shell welds (except that the intersections of the axial and circumferential welds would have been included; approximately 2 - 3 percent of the circumferential welds would be inspected under this proposal). Reference 4 provided additional supplemental information in response to a third RAI dated May 20, 1997 (Reference 9). References 5 and 6 provided the BWRVIP's VIPER computer code and detailed programming information for the VIPER code; the VIPER code was used by the BWRVIP in obtaining the results detailed in the BWRVIP-05 report.

The staff met with members of the BWRVIP and their consultants on several occasions to discuss issues related to the review of the BWRVIP-05 report. These meetings were summarized in the following meeting summaries: summary of July 18, 1995, meeting dated July 25, 1995; summary of March 19, 1996, meeting dated March 26, 1996; summary of October 15, 1996, meeting dated December 10, 1996; and, summary of January 16, 1997, meeting dated February 13, 1997 (References 10, 11, 12, and 13, respectively). Additionally, by letter dated April 18, 1997 (Reference 14), the BWRVIP requested to meet with the Commission on this issue. On May 12, 1997, the Commission was briefed by representatives of the BWRVIP and the staff on the issues related to the requirements for a full inspection of reactor pressure vessel shell welds. The above meeting summaries, as well as the transcript of the May 12, 1997, Commission meeting (Reference 15) and the Commission's May 30, 1997, Staff Requirements Memorandum (SRM M970512B, Reference 16) are available in the Commission's Public Document Room, 2120 L Street, N.W., Washington, D.C. 20555.

1.1 Background

In January 1991, the NRC published in the *Federal Register* (56 FR 3796) a proposed Rule to amend Section 50.55a to Title 10 of the *Code of Federal Regulations* [10 CFR 50.55a], "Codes and Standards." One purpose of this amendment was to incorporate by reference a later edition and addenda to Section XI of the American Society of Mechanical Engineers (ASME) Code. This included the 1989 Edition of the ASME Section XI, Division 1, and addenda through 1988. In addition, the Rule proposed to create Section 50.55a(g)(6)(ii)(A) to 10 CFR 50.55a [10 CFR 50.55a(g)(6)(ii)(A)], "Augmented examination of reactor vessel," which required that all licensees perform an inspection of RPV welds in accordance with Section XI of the ASME Code on an expedited schedule, and revoked all previously granted reliefs for RPV weld examinations.

As noted in the Statement of Considerations for the Rule, the primary reason for the augmented examination requirement was that few examinations had been performed up to that time on either boiling water reactor (BWR) or on pressurized water reactor (PWR) RPV shell welds. Therefore, there was a concern on the part of the staff regarding the existence of manufacturing flaws in the RPV shell welds, and the initiation and propagation of flaws during service. The staff had no assurance that ASME Section XI flaw acceptance criteria were being satisfied in the RPV shell welds, which had never been completely examined as part of a Section XI inservice inspection program. The augmented examination was needed for BWRs because of evidence demonstrating a viable mechanism for initiating environmentally-assisted cracks in the RPV cladding, evidence that the cladding cracks can propagate into the ferritic steel of the RPV base material and, evidence that BWR reactor vessels may be embrittled more by neutron irradiation than would be predicted by Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials."

The intent of 10 CFR 50.55a(g)(6)(ii)(A) was to require that licensees perform an expanded RPV shell weld examination, as specified in the 1989 Edition of ASME Section XI, on an "expedited" basis. "Expedited," in this context, effectively meant during the inspection interval when the Rule was approved or the first period of the next inspection interval. The final Rule was published in the *Federal Register* on August 6, 1992 (57 FR 34666).

By incorporating into the regulations the 1989 Edition of the ASME Code, the staff was requiring that licensees perform volumetric examinations of "essentially 100 percent" of the RPV pressure-retaining shell welds during all inspection intervals. This represented an expansion by the ASME of the requirements from previous editions of Section XI of the ASME Code which, as far back as the Winter 1975 Addenda, had required examination of 100 percent of the RPV pressure-retaining welds during the first inspection interval, then limited examinations in the intervals thereafter. Requiring every licensee to perform an extensive volumetric examination of every RPV shell weld at least once during the service life of the RPV was therefore consistent with the philosophy that had been expressed in the ASME Code for more than 20 years. In those 20 years, however, BWR licensees had requested and been granted extensive relief from performing RPV shell weld examinations. As a result, only a small percentage of the beltline welds in BWR RPVs had been examined,

projected to be reached later in life at some plants. This additional work includes further studies of the potential precursor events in order to better quantify the potential for cold overpressure events in BWRs, and additional probabilistic fracture mechanics analysis to both understand the sensitivities to various parameters and to support an uncertainty analysis.

This assessment provides, in Section 2.0, a description of the techniques used by the several vendors to fabricate the reactor vessels. Section 3.0 contains a discussion of the history of non-destructive examinations, and Section 4.0 has a discussion of the two degradation mechanisms which have the potential to initiate RPV cracking or to cause existing flaws to grow. In Section 5.0 and Appendix C there is a discussion of limiting transients of concern. The probabilistic fracture mechanics model is discussed in Section 6.0, and an integrated probabilistic assessment is presented in Section 7.0. Section 8.0 contains a summary of foreign inspection strategies and inspection capabilities, and the staff's interim conclusions are presented in Section 9.0. A final Safety Evaluation Report (SER) will be completed and provide additional evaluations of non-design basis events. A final conclusion regarding the need for circumferential weld examinations, or other actions that may be suggested by the risk informed assessment, will be reached when the final SER is completed.

2.0 REVIEW OF REACTOR PRESSURE VESSEL FABRICATION TECHNIQUES

Reactor pressure vessels were fabricated to very high standards, as evidenced by preservice and inservice inspections to date. However, different processes were used to fabricate RPV welds under a number of varying conditions. Additionally, several vendors constructed BWR RPVs. The staff reviewed the various fabrication methods associated with RPV fabrication in order to support a statistical assessment of a sampling approach to RPV weld inspections. In particular, the staff focused on whether it could be assumed that, considering the various fabrication methods, that all RPV welds represent one statistical population, or if the population of RPV welds could be divided into well defined subpopulations. Section 2.1 discusses the review of welding techniques and differences in the types of welds (Reference 17). Section 2.2 discusses the statistical evaluation of a sampling approach to inspection.

2.1 Summary of Fabrication Methods

A number or variables had to be considered when determining which weld process to use for fabrication of an RPV. One variable was economic impact. During the late 1960's and early 1970's when RPVs were fabricated, shielded metal arc welding (SMAW) was the most expensive of the three weld processes. Submerged arc welding (SAW) was less expensive than the SMAW process. Electroslag welding (ESW) was the least expensive process of the three, however, it could only be used to fabricate axial welds. The weld processes used in fabricating BWRs are described below. Relative strengths and weaknesses for each weld process are also outlined.

2.1.1 Shielded Metal Arc Welding (SMAW)

SMAW is a process wherein coalescence is produced by heating with an electric arc between a coated electrode and the plate or forging. SMAW is a manual process in which the electrode is fed into the weld joint by the welder. Shielding is obtained from decomposition of the electrode coating and the filler metal is obtained from the electrode. Using the SMAW process, weld metal can be deposited in any orientation. Therefore, this process was used for some field-fabricated beltline welds.

Welding is interrupted frequently to change the electrode which results in a slower rate of deposition and smaller weld beads. More than 100 passes could be needed to complete an SMAW weld. Since the process is manual, there is a greater probability of weld defects. These defects are expected to be relatively smaller than defects associated with other weld processes.

2.1.2 Submerged Arc Welding (SAW)

SAW is a process wherein coalescence is produced by heating with an electric arc or arcs between a bare metal electrode or electrodes and the plate or forging. SAW is an automatic process in which the electrode is a spool of wire that is continuously fed into the weld joint at a constant rate. The welding is shielded by a blanket of granular, fusible material that is on the plate or forging, and the filler material is obtained from the electrode or electrodes. The granular material, or molten flux, that submerges the arc, serves to protect the weld from oxidation.

Since SAW is an automatic process, it is less subject to operator error and would be less likely to produce discontinuities such as cracks, entrapped slag or porosity. However, competent welders should be able to produce quality SMAW welds that are equivalent to those produced using the SAW process. The number of passes needed to complete an SAW weld could be 50 to 100.

2.1.3 Electroslag Welding (ESW)

ESW is a process that is unique in comparison to the other weld processes since the entire weld thickness is deposited in one pass. ESW is an automatic process used to fabricate axial welds (not applicable to circumferential welds). An arc is present throughout the process, and the melted weld flux forms a molten slag which provides the heat needed to melt the weld metal. The slag floats above the molten weld metal and protects the weldment from atmospheric contamination. One or more weld wires can be fed into the weld, so high weld deposition rates can be achieved. The molten weld metal and slag are contained in the weld region by water-cooled copper shoes that are braced against the plates being joined. The structure of the ESW weld is as-cast throughout and must be heat treated to achieve the required properties. The ESW process closely resembles a continuous casting operation.

Due to the one-pass nature of the ESW process, the probability of producing a defect is small, but any defects would be larger than those found in welds that are fabricated using the other weld processes. It should be noted that there are welding variables such as heat and contamination that affect the quality of the resulting weld.

2.1.4 Division of BWR Welds into Sub-Populations for Statistical Analysis

It is important to note that each reactor vessel can have unique characteristics that distinguish it from other reactor vessels, even if the vessels were fabricated by the same vendor. For example, differences in the fit up of the vessel shell courses or "roundness" of the vessel can exist among vessels that were fabricated by the same vendor. Accurately categorizing sub-populations of reactor vessel welds is difficult due to the many variables involved in the welding process. However, the staff attempted to divide the welds according to available weld process information.

2.1.4.1 Manufacturers of BWR RPVs

Table 2-1 shows the BWR welds grouped by vessel fabricator and weld process. The axial welds that were fabricated by Chicago Bridge and Iron (CB&I) are double V-groove welds made with a spacer bar. The SAW process was used from the shell inner diameter (ID). The spacer bar was removed by back gouging from the outer diameter (OD), then the SMAW process was used to complete fabrication of the axial weld.

The circumferential welds that were fabricated by CB&I are typically single V-groove welds with a closed root gap. In most cases, the SAW process was used from the shell OD. Back gouging was used to remove the root pass in order to reach the sound metal. The SMAW process was used from the shell ID to complete fabrication of the circumferential weld.

The axial welds that were fabricated by Combustion Engineering (CE) are double U-groove welds with a closed root gap geometry. The SAW process was used from both the shell ID and OD. After one side was completed, back gouging was used to reach the sound weld metal. The SMAW process was used as needed to shape the back gouged cavity. The SAW process was used to complete fabrication of the axial weld.

The circumferential welds that were fabricated by CE are square groove welds with a carbon steel backing strip. The SAW process was used from the shell OD. The backing strip was removed and back gouging was used to reach the sound weld metal. The SMAW process was used from the shell ID to complete fabrication of the circumferential weld.

The ESW process was used for axial welds that were fabricated by Babcock & Wilcox (B&W). The exception is the Big Rock Point axial welds which were fabricated using the SAW process[1]. The SAW process was used from the shell ID. The spacer bar was removed by back gouging from the shell OD. The SAW process was used from the shell OD to complete fabrication of the axial weld.

The circumferential welds that were fabricated by B&W are a single V-groove welds with a backing strip. The SAW process was used from the shell OD. The backing strip was removed and back gouging was used to reach the sound weld

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(1) Big Rock Point is scheduled to be permanently shut down by Fall 1997, and is not considered in this assessment.

metal. The SMAW process was used from the shell ID to complete fabrication of the circumferential weld.

Dresden unit 2 was the only BWR fabricated by New York Shipbuilding (NYS). The ESW process was used to fabricate the axial welds, and the SAW process was used to fabricate the circumferential welds. The NYS fabricated vessel is comparable to the B&W vessels.

Hope Creek was the only BWR fabricated by Hitachi. The SMAW process was used to fabricate the axial welds, the circumferential welds, and a low pressure coolant injection (LPCI) nozzle weld that is in the beltline. The Hitachi fabricated vessel is comparable to the CB&I vessels.

2.1.4.2 Weld Categorization

Sources of variance to consider when categorizing welds include: 1) whether the welds were field or shop fabricated, 2) details of weld repair procedures, 3) whether a single or double groove process was used to fabricate the weld, 4) details of back gouging procedures, 5) details of cladding procedures, and 6) stress relief processes.

The BWRVIP reviewed plant-specific documents and reported that the following CB&I vessels were field completed:

Clinton	Duane Arnold	Limerick 1 & 2
Monticello	Susquehanna 1 & 2	Vermont Yankee

All other BWR vessels were shop fabricated. Details on which CB&I welds were field or shop fabricated are not readily available. This level of detail would require a plant-specific search. With regard to weld repair, locations of the repairs would vary; but all major repairs were required to be documented as part of the ASME Code procedures.

With regard to back chipping and weld refill, double and single groove welds with a backing strip or spacer bar require grinding, chipping, machining and/or air-arc gouging to remove root defects. Lack of fusion and porosity were common defects that occurred because production of a fully fused root was not an objective of the weld procedure. Back gouging the root to sound weld metal was an important part of the weld procedure. When air-arc gouging is used, an additional 0.03125 inches (0.79375 mm) of weld metal was required to be removed by mechanical means. This was required to assure that the arc-heat hardened surface was removed. After the metal is removed, the weld groove is restored to the original groove dimensions. Weld repair may be required if the weld groove is not acceptable for continued welding. The weld groove is surface inspected to assure removal of defects before welding.

With regard to back cladding, the axial and circumferential welds were back clad using SMAW, SAW, or the semi-automatic gas metal arc welding (GMAW), or some combination of weld processes. The plate material and axial welds of individual shell courses were machine clad as one step. The vessel assembly drawing usually specified more than one process. In general, there are a minimum of two layers of back clad. The first layer is E309 type electrode and the remaining layers are E308L. E309 and E308 type electrodes were used

in older vessels. For B&W fabricated vessels, the SMAW, SAW and GMAW processes were used for back cladding. For CE and CB&I fabricated vessels, the SMAW process or a combination of the SMAW and GMAW processes were used for back cladding.

With regard to stress relief processes, the standard temperature and time used for stress relief was 1150°F (621°C) for a cumulative period of approximately 35 hours. The guidelines for the amount of stress relief time are given by Section III of the ASME Code. A shell course may have gone through multiple stress relief cycles as a sub-assembly before experiencing another process cycle during treatment of the circumferential weld. It should be noted that the ASME Code contains alternate requirements for stress relief temperatures that are lower than the minimum of 1100°F (593°C).

2.2 Statistical Analysis of A Sampling Approach to RPV Weld Examinations

Of the total length of axial and circumferential welds in a BWR RPV, approximately 60 percent are circumferential welds and axial welds make up the remaining 40 percent. The staff has concluded that no meaningful statement about the circumferential welds can be made based only on a statistical analysis of the axial welds. Any such statement must be based on the assumption that both the axial and circumferential welds are random samples from some parent population of welds. If this assumption is made, an inference can be made about the expected number of defective circumferential welds based on an inspection of the axial welds. For the best possible situation in which *all* axial welds are inspected and *no* defective welds are found, an inference can be made about the expected number of defective circumferential welds based on the inspection of the axial welds:

> Denote the expected number of defective[2] circumferential welds by d, the proportion of defective welds in the parent population by p, the number of circumferential welds by m and the number of axial welds by n. Assuming that circumferential welds comprise 60 percent and axial welds 40 percent of the total weld length, then $m = 1.5n$ and $d = mp$. Further, with the standard confidence level of 95 percent, it can be concluded that $p < 3/n$. Therefore, it follows that, with 95 percent confidence, $d < 4.5$. Based on this bound for d, a bound on the probability that at least one of the circumferential welds is defective can be derived. The distribution of the number of defective circumferential welds is given by a Poisson distribution with mean d. Therefore, the probability that at least one circumferential weld is defective is equal to $1 - \exp(d)$. It follows that this probability is less than $1 - \exp(-4.5) = 0.99$, with 95 percent confidence.

Based on the above evaluation, the staff has concluded that no useful inference regarding the condition of the circumferential welds can be made based on a 100 percent inspection of the axial welds only. The staff also considered the effect of including a small percentage of the circumferential

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(2) Note that for the purpose of this evaluation, "defective" is defined as containing a crack-like indication that could severely jeopardize RPV integrity.

welds to in the inspection sample by assuming a sample size of 50 percent of total RPV welds (axial and circumferential). Including the small percentage of the circumferential welds does not alter the above conclusion.

The above statistical analysis is based on the assumption that both the axial and circumferential welds are random samples from the same parent population of welds. In reality, as discussed in Section 2.1, the circumferential welds are not from the same population due to the different weld processes and other sources of variances (i.e., field vs. shop fabrication); therefore, a sampling approach is not valid.

An example of the above is the results of the Tennessee Valley Authority's Browns Ferry Nuclear Plant Unit 3 RPV shell welds augmented examination (Reference 17). The augmented examination consisted of 15 axial welds and 5 circumferential welds. The axial welds were ESW and the circumferential welds were SAW. The axial welds contained no reportable indications. The circumferential welds contained over 100 recordable indications[3], 15 of which required evaluation to be determined acceptable by ASME Code Section XI, IWB-3600 (1986 Edition). Since all the recordable flaws were in the SAW type welds, this examination indicates that the SAW type welds have a greater propensity for flaws than the ESW type welds. Hence, examination results from the ESW axial welds are not representative of the SAW circumferential welds.[4]

The above evaluation of a statistical sampling approach to RPV weld examinations indicates that relatively large sample sizes e.g., on the order of 50%, do not allow a meaningful conclusion with regard to the possibility of a single or a small number of isolated unacceptable defects existing in the remainder of the population. The practical implication of this evaluation is that a sampling approach cannot provide reliable protection against the possibility of an isolated, unacceptable conditions existing in the population of uninspected welds. An example of this situation would be a large defect existing in a substandard weld repair. This is an important conclusion relative to assessment of the RPV which must have an extremely high reliability. However, this evaluation should not be interpreted as meaning that no useful information can be gathered from inspection of a sample of welds. For example, a sampling inspection can provide useful inferences regarding the possible existence of a more wide-spread mechanistic mode of degradation. An example of this might be a service induced cracking mechanism. Thus it is possible that additional insights may be available through a statistical assessment of the sample of welds that has been or will be inspected in BWRs under the VIP-05 proposal. As indicated in Section 2.1,

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(3) "Reportable" indications are those that are required to be reported to the NRC; "recordable" indications are those of a sufficient size to be recorded by the testing equipment.

(4) During a meeting on August 8, 1997, the staff was informed that the axial welds at Hatch did non-reportable indications and that the inspection methods used for the axial welds may have been different from those used for the circumferential welds. The staff will need additional details in order to assess this new information.

care must be taken in any such evaluation to make sure that differences in sub-populations of welds are taken into account, and it may be difficult to define such sub-populations. The staff intends to perform further evaluations and to pursue this area in greater detail with the industry.

TABLE 2-1
Processes Used to Fabricate Beltline Welds in BWR Reactor Vessels

		-----PROCESS USED TO FABRICATE-----	
PLANT	FABRICATOR	AXIAL WELDS	CIRCUMFERENTIAL WELDS
BIG ROCK POINT	B&W	SAW	SAW
BROWNS FERRY UNITS 1/2/3	B&W	ESW	SAW
DRESDEN UNIT 3	B&W	ESW	SAW
PEACH BOTTOM UNITS 2/3	B&W	ESW	SAW
QUAD CITIES UNITS 1/2	B&W	ESW	SAW
BRUNSWICK UNITS 1/2	CB&I	SAW	SAW
CLINTON	CB&I	SAW	SMAW
DUANE ARNOLD	CB&I	SMAW	SMAW
GRAND GULF	CB&I	SMAW	SAW
LASALLE UNIT 2	CB&I	SAW	SAW
LIMERICK UNIT 1	CB&I	SMAW	SAW
MONTICELLO	CB&I	SMAW	SMAW
NINE MILE POINT UNIT 2	CB&I	SAW	SAW
PERRY	CB&I	SMAW	SAW
RIVER BEND	CB&I	SMAW	SMAW
SUSQUEHANNA UNITS 1/2	CB&I	SAW	SMAW
VERMONT YANKEE	CB&I	SMAW	SMAW
WNP UNIT 2	CB&I	SAW	SAW
LIMERICK UNIT 2	CB&I	SMAW	SMAW
COOPER	CE	SAW	SAW
FERMI UNIT 2	CE	SAW	SAW
FITZPATRICK	CE	SAW	SAW
HATCH UNITS 1/2	CE	SAW	SAW
LASALLE UNIT 1	CE	SAW	SAW
MILLSTONE UNIT 1	CE	SAW	SAW
NINE MILE POINT UNIT 1	CE	SAW	SAW
OYSTER CREEK	CE	SAW	SAW
PILGRIM	CE	SAW	SAW

TABLE 2-1
Processes Used to Fabricate Beltline Welds in BWR Reactor Vessels

		-----PROCESS USED TO FABRICATE-----	
PLANT	FABRICATOR	AXIAL WELDS	CIRCUMFERENTIAL WELDS
DRESDEN UNIT 2	NYS	ESW	SAW
HOPE CREEK	HITACHI	SMAW	SMAW*

*Also SMAW for a nozzle weld

3.0 EVALUATION OF NON-DESTRUCTIVE EXAMINATION (NDE)

Reactor pressure vessels were constructed to ASME Section III requirements, which include the requirements for NDE fabrication inspection. Many reactor pressure vessels were constructed in the U.S. from the mid-1960s through the mid-1970s. To evaluate the effectiveness of the NDE performed during this time, several relevant studies of inspection reliability should be assessed.

The only significant study conducted in the 1960s was the work of the Pressure Vessel Research Committee (PVRC) of the Welding Research Council. In the late 1960s, PVRC Specimen 201 (CB&I vessel 8" thick, butt weld, manual metal arc process) was inspected, using several techniques and procedures. This specimen contained a total of 10 flaws (3 cracks, 5 slags, and 2 lack-of-fusion flaws). The effectiveness of the inspection procedure was computed as the ratio (in percent) of the number of flaws detected plus the number of false indications to the total number of flaws present. The average performance of the radiographic testing (RT) teams was only 26 percent, and ranged from a high of 33 percent to a low of 17 percent. The performance of the ultrasonic testing (UT) teams from the unclad surface was about 50 percent, with a high of 65 percent and a low of 43 percent. For UT from the clad surface, the UT average was 14 percent, with a high of 43 percent and a low of 0 percent (3 teams made no detections at all). With advanced 1960s NDE technology, such as acoustic holography and focused probes, UT inspections from the clad surface achieved an average of 65 percent, with a high of 70 percent and a low of 60 percent.

This work was followed by inspections of PVRC Specimen 251J (CE vessel, butt weld, with submerged arc weld process), through the mid-1970s. This plate contained 10 cracks and 5 slags. The RT results were similar to those described above, with an average of 21 percent (a high of 27 percent and a low of 15 percent). The UT inspections of Specimen 251J produced similar performance to that for Specimen 201 and even included a specially developed PVRC procedures intended to address some problem areas identified in the earlier PVRC work.

The next significant reliability study was a round robin conducted under the international Plate Inspection Steering Committee (PISC) program (References 19 and 20). The focus of PISC was on UT, and the PISC procedure was based on the ASME 1974 Code, using a recording threshold of 20 percent DAC (distance-amplitude-correction) and a reporting threshold of 50 percent DAC. Any UT signal exceeding the recording threshold must be recorded and those signals

exceeding the reporting threshold must be fully documented, dispositioned and reported. The defect detection probability exceeded 90 percent for vertical defects 50 mm (1.97 inches) and larger in depth. However, for sets of defects it was only 40 percent for defects up to 150 mm (5.91 inches) in through-wall size.

The PISC studies continued with a second phase in the early to mid-1980s (Reference 21). This PISC II database is the largest ever collected on the capabilities of NDE procedures and techniques for thick section steel. The results for PISC Plate No. 3 show that UT procedures based on a 50 percent DAC sensitivity achieve, on average, a 15 percent probability of detection and correct sentencing,[5] for defects up to 60 mm (2.36 inches) in through-wall size. To put this in perspective, this is the sensitivity level specified in the ASME Code up until the 1986 edition; and the changes introduced with the 1986 edition were based on these PISC II results.

When the sensitivity was increased to 10 or 20 percent DAC, the effectiveness of NDE increased, on average, to 75 percent, for defects up to 60 mm (2.36 inches) in through-wall size. This PISC II result is the basis for the ASME Code requirement going to a sensitivity of 20 percent DAC. The PISC II studies also showed that effective performance for the near surface zone can be achieved only with special NDE techniques; and that a 70° dual longitudinal transducer is an effective tool for inspections of the near surface zone through the vessel cladding.

In summary, these PVRC and PISC studies showed that RT and UT inspections conducted through the mid-1980s, in accordance with ASME Code minimum requirements, were not very effective. Inspections based on the procedures, equipment, and personnel requirements specified in ASME Section XI Code, Appendix VIII, are expected to perform significantly better than those based on ASME Code prior to Appendix VIII. It is difficult to state quantitatively how much better, since Appendix VIII describes a screening test, and does not quantify the probability of detection (POD). The design basis for reactor pressure vessel testing specified in Appendix VIII is that an inspection result that combines a true value of 70 percent POD with a false call probability of 20 percent should have only a few percent chance of passing the performance demonstration test. This is a significantly higher performance than that demonstrated in the earlier studies cited above.

The industry is using highly effective performance plots of POD versus flaw size for input to the VIPER program (Fig. 2-11 of Reference 5). This performance is significantly better than those reported in the PVRC and PISC studies (including the results for special procedures). Furthermore, the industry plots are the composite performance results for several separate and independent inspection techniques. Up through the 1986 edition, the ASME Code called for vessel examinations using two inspection angles, each larger than 30°, and separated by at least 15°. Most inspections were conducted at 45° and 60°, including the PVRC and PISC studies. For inspections based on the

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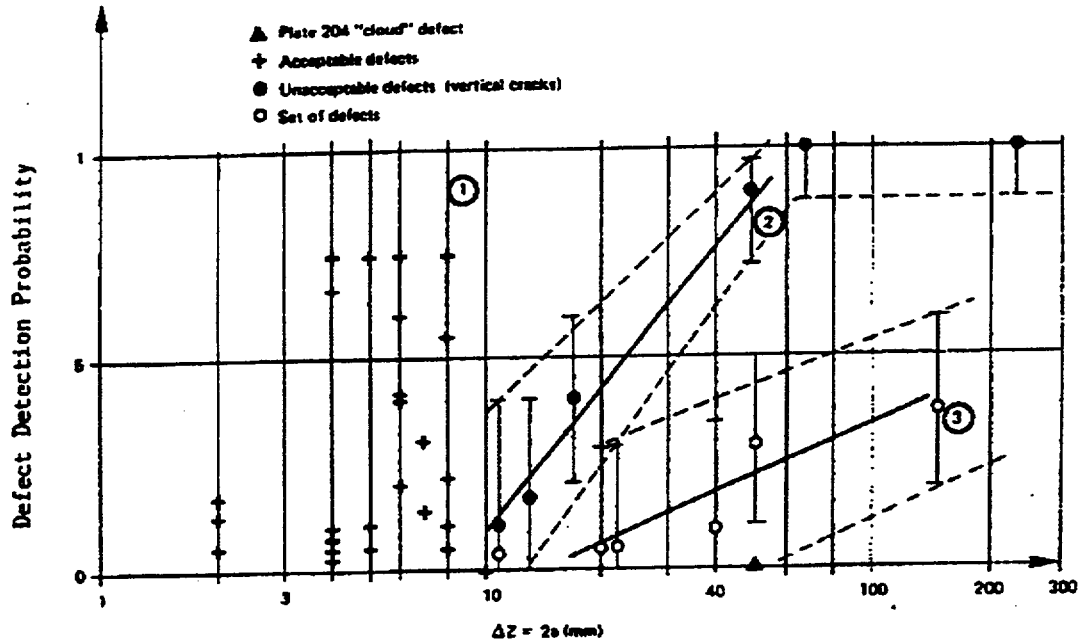
(5) Correct sentencing means that a defect is detected and is also correctly sized as an acceptable defect. If the defect is rejectable, it must be sized as rejectable, based on ASME Section XI, IWB-3500.

1986 and later editions of the ASME Code, performance would be expected to conform to the PISC II results for 20 percent DAC. This is substantially less effective than the performance the BWRVIP is using for the VIPER program.

The POD curves (Figure 2-11 of Reference 5) for use in the VIPER program are based on the assumptions that the techniques are complimentary and independent. Regarding the NDE procedures industry is using in the performance demonstration tests, the staff does not know if these are complementary or independent techniques. Furthermore, the PISC and other round robin studies have shown that the largest contributor to variability in POD performance is the human factor. These uncertainties are not reflected in the POD curves that industry is using (Figure 2-11 of Reference 5). The results shown there should be regarded as an upper bound on performance, and are not typical of inspections conducted using normal inservice inspection (ISI) techniques.

As an example, the PISC-II results on Plate No. 3 (shown in Figure 3-1), for a through-wall flaw size of 50 mm (1.97 inches) and a sensitivity of 20 percent DAC, the average performance is 75 percent with a variability from 50 percent to 100 percent probability of detection and correct sizing. To represent inspection performance more realistically, results should be shown separately for the early (pre-1986 Code) inspections with a sensitivity of 50 percent DAC, those with a sensitivity of 20 percent DAC, and those based on special procedures. The latter would be a good first approximation for those successfully passing the demonstration tests specified in Appendix VIII of the ASME Code.

In summary, the POD curves industry is using in the VIPER program are considered to be an optimistic upper bound on POD performance. It should be noted that assuming a high POD maximizes the benefit of inspection and is conservative for evaluating the ability of inspection to reduce the probability of RPV failures. However, such an assumption may not be conservative when assessing other inspection issues. Furthermore, the staff's position is that the reference case assessment should be performed using best estimate (i.e., realistic) assumptions. More realistic POD curves, based on round robin data (shown in Figure 3-1), should be used in the VIPER calculations. The proper curve should be selected in Figure 3-1, depending on the era of the NDE testing being simulated.



Defect size : dimensions of the defect in the through wall section

Thickness of the plates : from 200 mm to 250 mm

Figure 1 PISC I detection results for defect detection probability versus defect depth using 1974 ASME Section XI Code type procedures at sensitivity of 50% DAC (Reference 10).

4.0 RPV DEGRADATION MECHANISMS

Based on approximately 700 reactor-years of U.S. operating experience of BWR vessels (approximately 1700 reactor-years world-wide), the BWRVIP identified that fatigue and stress corrosion cracking (SCC) are the two degradation mechanisms which have the potential to initiate RPV cracking or to cause existing flaws to grow. There are two fatigue mechanisms significant to the BWR RPV: system cycling fatigue and rapid cycling fatigue. The system cycling fatigue, associated with plant start-up, shutdown, SCRAM and safety relief valve (SRV) blowdown, had been evaluated for the vessel shell as part of the vessel design and, because the resulting usage factors were around 0.1, found by the BWRVIP to be insignificant. The rapid cycling fatigue associated with plant operation when feedwater sparger and nozzle cracking was initiated was discounted because the vessel shell is remote from these nozzles.

Based on the above, the BWRVIP narrowed its investigation of the degradation mechanisms to only SCC, which was suggested by RPV cracking experience (e.g, cracks found in some clad and unclad steam generator shells of PWR vessels and in some recirculation nozzles of BWR vessels). Cracking observed in the manually backclad region also suggested the involvement of SCC.

The staff reviewed the qualitative arguments in the BWRVIP-05 report supporting the elimination of fatigue as a significant degradation mechanism, and determined that additional analysis was required. As a result, the BWRVIP provided (Reference 2) a quantitative comparison of the crack growths due to these two mechanisms. The low alloy steel (LAS) SCC growth rate relation of $da/dt = 1.18 \times 10^{-13} K^4$ and the ASME Section XI reference bi-linear fatigue crack growth law for carbon and low alloy steels in water environment were used in this study. The crack growth history for both mechanisms were presented for crack depths of 1.27, 2.54, and 7.62 cm (0.5, 1.0, and 3.0 inches). The results indicate that, when the cracks were increased into the range of larger sizes of the Marshall distribution (e.g., 7.62 cm or 3.0 inches), the crack growth from stress corrosion is more dominant. This difference is even more pronounced if the staff's LAS SCC growth rate, which is approximately 10 times larger than the BWRVIP's value, is used in this comparison. Hence, the staff has concluded that the fatigue crack growth does not need to be considered in this application.

5.0 DISCUSSION OF LIMITING TRANSIENTS

The BWRVIP-05 report discusses the operating characteristics of a BWR with respect to design transients and their effects on the vessel. These transients generally occur when a large steam region exists (i.e., a steam bubble is present). According to the BWRVIP-05 report, the most limiting operational transients with respect to the vessel are loss of feedwater or single safety relief valve (SRV) blowdown events for normal and upset conditions. Normal operating temperature and pressure for a BWR RPV is 260°C (500°F) and 6.9 MPa (1000 psig). During the hydrostatic test, RPV temperature and pressure is approximately 65.5°C to 93.3°C (150°F to 200°F) and 6.9 MPa (1000 psig) and is maintained on the pressure-temperature (P-T) curve for that particular vessel. The most limiting transients for emergency and faulted operating conditions are any transient which causes or results in a rapid cooldown, and rapid depressurization, of the vessel shell welds. These transients are limiting for pre-existing cracks in the vessel shell welds. In comparison, the BWRVIP-05 report states that the water-solid leak test condition, or hydrostatic test, is limiting for small flaws in RPV ID.

To assess the probability of failure (POF) for rapid depressurization, the staff assumed a hypothetical transient and performed a parametric study using the VISA-II code. The transient was characterized by a rapid depressurization to 10 psi (69 kPa) with the fluid temperature decreasing from 550°F to 206°F (288°C to 97°C) in 12 seconds (i.e., equivalent to a cooling rate of 1720°F or 956°C per minute). Use of this essentially instantaneous cooldown transient was meant to bound all real transients. The staff examined both axial and circumferential welds in this study. In one case, the staff intentionally increased the fluence to 5.0×10^{18} n/cm² and the initial RT_{NDT} for the axial weld to 40°F to increase the probability of producing vessel failure. However, the staff's results indicated no failure for all cases subjected to rapid depressurization transient. As a result, the severe foreign transient discussed in Appendix C.2.1 was used in the staff's independent simulations reported in Section 7.

The BWRVIP-05 report was limited to design basis accident (DBA) events. To provide a broader risk-informed assessment, the staff performed a sampling review of approximately 17 years of licensee event reports (LERs) and event notifications (ENs) to determine if other events (i.e., shutdown events) could be potentially more limiting to the vessel (see Appendix C). The staff's sampling review does not encompass all LERs and ENs, which will be more fully reviewed in the staff's final SER. The staff also notes that some overpressure events during shutdown, as described in Appendix C.1, may not be reportable under 10 CFR 50.72. The preliminary staff sampling review is discussed in Appendix C.2.

Table C-1 includes the results of the staff's sampling review of LERs and ENs. The table provides the maximum pressure and temperature reached at the end of the transient, where known. Where the information was not provided, an NP is listed. It should be noted that, although the transients discussed in Appendix C are examples of the types of events that could result in cold overpressure events, these events occurred at temperatures that are high relative to the reference temperature of the vessel weld material, as required by plant Technical Specifications (TS). Therefore, these events did not, in and of themselves, represent significant challenges to the RPV. This is more thoroughly discussed in Appendix C.

The staff's review also identified an actual low-temperature overpressure event that occurred at a foreign plant. A more detailed discussion of the staff's independent assessment of the potential for beyond design basis challenges to the RPV is presented in Section 7.1.

6.0 EVALUATION OF PROBABILISTIC FRACTURE MECHANICS MODEL

Conventional vessel analysis codes, such as VISA-II and FAVOR (References 22 and 23), are codes based on probabilistic fracture mechanics (PFM) methodology that perform millions of deterministic vessel simulations to determine the conditional POF for a vessel subjected to a specific transient. The vessel conditional POF is the ratio of the number of failed vessels to the number of simulations. For each simulation, the random variables (e.g., crack size, copper, nickel, and fluence) are assigned according to the prescribed distribution with the form and parameters of the distributions specified by the user. Deterministic fracture mechanics analyses are then performed, and the vessel conditional POF is determined.

The BWRVIP-05 report used the VIPER code (Reference 5) to perform the PFM analyses. Section 6.1 describes the underlying methodology and the special features in the VIPER code. Section 6.2 evaluates technical issues and input parameters. All major technical issues related to the VIPER code have been reviewed by the staff, with special attention to the flaw size distribution and the flaw density in vessel welds. New information in these two areas has been accumulated in the past 5 years. The staff has used the new information to thoroughly address the projected flaw size distribution and density. The staff's assessment of the VIPER's importance sampling technique is in Section 6.3 and the staff's assessment of the BWRVIP's POF results is presented in Section 6.4.

6.1 Structure and Underlying Methodology

The structure and the underlying methodology in VIPER is similar to those of VISA-II and FAVOR. In these type of codes, parameters which are not highly variable (e.g., vessel and clad geometry, transient pressure and temperature definition, weld volume and orientation) are treated as deterministic. Highly variable parameters (e.g, clad stress, fluence, copper and nickel contents, initial RT_{NDT}, and fracture toughness) are treated as random variables. The VIPER code differs from VISA-II or FAVOR by having implemented the following additional features to its core PFM methodology: (1) an importance sampling technique was added to the conventional Monte-Carlo simulation to significantly reduce the number of simulations needed to obtain convergent results, (2) a cladding SCC initiation curve was added to simulate crack growth in cladding, and (3) an SCC growth rate for LAS was added to simulate further crack growth from cladding into base metal or shell welds.

The VIPER code assigned initiated and manufacturing flaws and sizes in accordance with the appropriate distributions. Crack initiation in cladding was first determined by comparing the service time with the predicted time for cladding crack initiation under a certain cladding stress according to the cladding SCC initiation law. Once initiation occurs, the flaw is assumed to crack through the clad instantly and line up with a manufacturing flaw in the LAS. Material properties (e.g., fluence, copper and nickel contents, RT_{NDT}) are sampled for each flaw, and the flaws are grown according to the probabilistic crack growth law. Inspections are performed for the inspection intervals and percentage of weld specified, and the process continued until either the flaw grows to failure (based on fracture mechanics analysis), or until the end of vessel life is reached. The program continues to loop through the analysis in this manner, until the specified number of iterations have been completed. Like the VISA-II code, VIPER can define POD curves and treat the percent inspection of the vessel as an input parameter. Detailed discussions regarding these three areas and the POD curves used in the VIPER are discussed in the following sections, along with some other basic technical issues and input parameters.

6.2 Technical Issues and Input Parameters

6.2.1 Flaw Size Distribution

For cases without preservice inspection (PSI), the VIPER code used the following function to describe the flaw size distribution:

$$N(x) = (6.94)\exp(-6.94x) \quad \ldots \quad (1)$$

where x is the crack length in inches and N(x) has been normalized. The staff used the following flaw size distribution derived from the pressure vessel research user facility (PVRUF) data (Reference 24). The distribution, in the cumulative distribution form, is:

$$F(.0787) = .9054 \text{ and,}$$

$$F(x) = 1 - (.1616)\exp(-6.94x) - (.00139)\exp(-4.06x) \text{ for } x > .0787 \quad \ldots \quad (2)$$

The distribution used in the BWRVIP-05 report and in the staff's assessment were based on the Marshall flaw size distribution. The Marshall flaw size distribution (Reference 25), after normalization, is (page 41, Reference 22):

$$N(x) = (0.0346)\exp(-4.06x) + (6.88)\exp(-6.94x) \quad (3)$$

with ultrasonic inspection, and

$$N(x) = (4.06)\exp(-4.06x) \quad (4)$$

without ultrasonic inspection. Although Marshall did not mention (Reference 25) whether the inspection was preservice or inservice, the staff interprets "ultrasonic inspection" as PSI.

The VIPER manual (Reference 5) indicates that the Marshall distribution *with inspection* (i.e., Equation 3), was used for the as-manufactured (*without PSI*) flaw size distribution. If this were the case, the BWRVIP would have double counted the benefit of inspection for the case when inspection was considered *and* an associated POD curve was utilized. This is because Equation (3) has already considered the benefit of PSI inspection. However, the staff found through examining the source coding of VIPER that instead of Equation (3), the VIPER code employed Equation (1) in its PFM calculations for the as-manufactured flaw size distribution. The BWRVIP did not provide a basis for using Equation (1). However, for large flaw sizes, Equation (1) is a good approximation to Equation (3).

The staff's flaw size distribution was calibrated against the PVRUF data and follows the Marshall distribution in Equation (3) for $x > 0.0787$ inch. Section 7.2.4 contains the derivation of the staff's flaw distribution. Since the staff's flaw size distribution has a probability of 91 percent for flaws under 0.0787 inch, and for larger flaws, the exponent in the major term of the staff's equation is the same as that in the BWRVIP's equation, the staff's flaw distribution may give lower vessel conditional POFs.

6.2.2 Flaw Density

The range for flaw density as stated in the Marshall report is 0.4 to 40 flaws per cubic meter, with the most likely value at the low end of this range. The BWRVIP used a flaw density of 30 flaws per cubic meter. Recent information (Reference 24) on flaw density indicates that the flaw density in RPV welds may be more than 30 times higher than that used by the BWRVIP. Therefore, the staff used a flaw density of 995 flaws per cubic meter in its assessment, which was based on detailed examinations on the Midland vessel and the PVRUF vessel and is discussed in Section 7.2.4.

6.2.3 Flaw Initiation and Growth Data

The VIPER code assumed that after an SCC initiated in the clad, it would penetrate the clad instantly and line up with underlying fabrication defects. The time to initiate the crack in the clad was developed by the BWRVIP and is discussed below. The staff assumed that service would induce SCC cracks in the BWR vessel cladding, but did not consider the initiating time of the cladding cracks. The staff then adopted the same assumption as in VIPER code

that the SCC would penetrate through the clad and combine with the fabrication flaws.

The VIPER code used a cladding SCC initiation curve that was derived from five "cast/weldment" data points. The VIPER code did not consider data pertaining to sensitized 304 stainless steels because BWRVIP believes that cast/weldment materials should have far greater resistance to SCC than sensitized 304 stainless steels. However, Figure 8-4 of the BWRVIP-05 report indicates that two of the five data points for cast/weldment material have the same initiation rate as that for sensitized 304 stainless steels in the BWR environment. Consequently, the BWRVIP's approach of forming a separate group for these five cast/weldment data points and using their best-fit curve in its PFM analyses may not be prudent. The BWRVIP later performed additional PFM analyses in its response (Reference 2) to the staff's RAIs (References 7 and 8) using the conservative SCC initiation curve for sensitized 304 stainless steel. These results indicated that the POFs became an order of magnitude higher because the crack initiated earlier in the cladding. However, the general trend regarding the effect of percent inspection on POFs remained unchanged. Nevertheless, when absolute values of POF become important and no additional cast/weldment data are available to show a definitive trend, the more conservative cladding SCC initiation curve should be used.

The BWRVIP developed its LAS SCC growth curve from low and high oxygen environments in Reference 26 and recent General Electric (GE) data from References 27 and 28. The recent GE data shows a lower crack growth in high oxygen, which falls reasonably in line with the old low oxygen data, and virtually no crack growth in a low oxygen environment. To take advantage of the lower crack growth rate exhibited by the new high oxygen data while discarding the zero crack growth exhibited by the new low oxygen data, the BWRVIP used an engineering approach to derive the growth rate for the low oxygen environment of BWR vessels. The BWRVIP found the ratio of the low to high oxygen crack growth rates from the old data, then applied this ratio to a fitted curve from the new high oxygen data. The BWRVIP's approach of deriving the LAS SCC curve for low oxygen environment gives heavy weight to the new GE data and to the ratio of the high oxygen growth to low oxygen growth rates from the old data. The staff considers the proposed LAS SCC growth curve not acceptable because there is no evidence that the new data is more accurate than the old data. Further, the test time for the GE new low oxygen data is only a few hundred hours, which is relatively short for such tests. Reference 29 showed that crack growth is virtually zero for hundreds of hours, and yet there was a quite significant average crack growth rate over much longer periods of time.

6.2.4 Fracture Toughness

The VIPER code has four options for the fracture toughness curve: ASME K_{Ic}, ASME K_{Ir}, the mean K_{Ic} (VISA K_{Ic}), and the mean K_{Ir} (VISA K_{Ir}). The ASME K_{Ic} and the ASME K_{Ir} curves in the Viper code are the lower bound curves in Section XI of the ASME code. The mean K_{Ic} and the mean K_{Ir} curves are from NRC analyses (Reference 30) and are also used in the VISA-II code. According to Reference 2, the BWRVIP used the mean K_{Ic} curve without sampling in all analyses performed in the BWRVIP-05 report. The mean K_{Ic} curve is truncated at 200 ksi $(in)^{1/2}$. Although VIPER treats the upper-shelf fracture toughness (i.e.,

truncated K_{Ic} value) as a random variable, this feature was deactivated in the BWRVIP's analyses. Consequently, there was no sampling on the upper-shelf fracture toughness either. The staff used the FAVOR code to perform the vessel simulations defined in Section 7. Unlike VIPER, FAVOR employs a mean K_{Ic} curve and samples about this curve. The mean K_{Ic} in FAVOR is different from the mean K_{Ic} in VIPER and VISA-II. FAVOR's mean K_{Ic} was derived from the bounding K_{Ic} curve in the ASME Code in which the bounding curve was considered to be a mean-minus-two standard deviation curve.

The BWRVIP's approach of using a mean K_{Ic} without sampling is not appropriate. Since the limiting transient that the BWRVIP evaluated occurs on the upper shelf, the BWRVIP essentially eliminated one of the key random variables from their PFM analyses. No sampling about the fracture toughness in the transition and lower-shelf regions also limits the application of the VIPER code. For instance, for the PFM analysis presented in Section 7, the foreign transient occurs below the upper-shelf region. Analyses using the mean K_{Ic} curve without sampling would give lower vessel POFs, and therefore, is non-conservative. The staff believes that sampling about the mean K_{Ic} is necessary to produce accurate POF.

The fracture toughness is calculated as a function of $T-RT_{NDT}$; where T is the temperature of the metal and RT_{NDT} is the reference temperature of the vessel weld material. The RT_{NDT} is the sum of the initial RT_{NDT}, the increase in RT_{NDT} resulting from neutron irradiation (ΔRT_{NDT}), and a margin term to account for uncertainty in the initial RT_{NDT}, copper and nickel contents and calculational procedures. The details will be presented in Section 6.2.6.

6.2.5 Probability of Detection Function

The VIPER manual indicates that four inspection methods can be specified by the user: eddy current (EDDY), full-V ultrasonic angle beam examination (FULLV), multi-mode ultrasonic examination (SLIC), and combined techniques (COMBINED), for both PSI and ISI. The BWRVIP's corresponding POD curves are shown in Figure 2-11 of Reference 5. The BWRVIP performed sensitivity studies on the effect of lower POD on the POF. The results summarized in Figure 8-14 of the BWRVIP-05 report shows that, for high POD curves, the POF for 50 percent inspection is about 7 times the POF for 90 percent inspection. For a low POD curve, the POF for 50 percent inspection is only 2 times the POF for 90 percent inspection. This is the basis that the BWRVIP used to apply only high POD curve to real vessel cases. Because if results from a POD curve that reflects high benefit of ISI can justify a relaxation of ISI requirements, the results from a POD curve that reflects less benefit of ISI can certainly justify the relaxation. The staff agrees that a lower POD would show a reduced benefit of ISI. However, as discussed previously, a realistic POD should be used when performing a probabilistic risk assessment. Assumptions regarding more or less optimistic PODs are useful for performing uncertainty or sensitivity studies.

6.2.6 Embrittlement Function and Reference Temperature

Embrittlement is measured as an increase in the reference temperature resulting from neutron radiation, ΔRT_{NDT}, and is a function of copper and nickel content and fluence. In the VIPER and FAVOR codes the copper and

nickel contents and the neutron fluence are random variables, and the mean ΔRT_{NDT} is calculated according to the procedure of Regulatory Guide (RG) 1.99, Revision 2 (Reference 31). The RT_{NDT} of the embrittled vessel material is the sum of the initial RT_{NDT}, the mean ΔRT_{NDT}, and a term to account for uncertainty in the initial RT_{NDT}, copper and nickel contents, and calculational procedures. Although in theory the uncertainty in ΔRT_{NDT} has been partially accounted for by treating the initial RT_{NDT} and copper and nickel contents as random variables, VIPER conservatively adopted the same approach as FAVOR by specifying a standard deviation for ΔRT_{NDT}. The square of this standard deviation for ΔRT_{NDT} is added to the square of the standard deviation for the initial RT_{NDT}. The term is calculated as the square root of this sum.

6.2.7 Other Input Parameters

The BWRVIP reported its summary of results for real vessel cases in Table 8-9 of the BWRVIP-05 report. The results for axial welds are based on a "typical" BWR vessel with an initial RT_{NDT} of 40°F, a copper content of 0.30 percent, a nickel content of 0.33 percent, a room temperature clad stress of 32 ksi, and a fluence of 4.9×10^{17} n/cm². The corresponding values for circumferential welds are -32°F for the initial RT_{NDT}, 0.06 percent for the copper content, and 0.97 percent for the nickel content. The mean fluence and the mean room temperature clad stress are the same for both welds. The standard deviation values for both welds are 17°F for the initial RT_{NDT}, 0.042 percent for the copper content, 0.05 percent for the nickel content, 5 ksi for the room temperature clad stress, and 20 percent for the fluence.

Although the effect due to high variability in copper and nickel contents has been explored by the BWRVIP through making additional calculations using standard deviations of copper (0.045 percent, 0.075 percent) and nickel (0.0165 percent, 0.075 percent), the effort was not extensive enough to determine the uncertainty associated with the reported POF results.

The staff surveyed the reactor vessel integrity database (RVID) and found that there are approximately 258 BWR RPV welds that have fluence 10 times larger than 4.9×10^{17} n/cm², and have a copper content and an initial RT_{NDT} for the circumferential weld much higher than those of the BWRVIP's. The staff does not believe that one reference case for axial welds and one reference case for circumferential welds can be used to represent all BWR reactor vessel welds. The staff surveyed the RVID database, and grouped the vessels according to fabrication processes for different vendors. Based on the difference in materials used in the fabrication of the CE, B&W, and CB&I vessels, the staff determined that three reference cases are necessary to comprehensively evaluate all BWR reactor vessels. The reference cases and the results from the PFM analysis of the reference cases and the associated uncertainty analyses are reported in Section 7. Other inputs, such as the mean (32 ksi) and standard deviation (5 ksi) of the clad stress, are considered adequate by the staff.

6.3 Importance Sampling

Importance sampling was used to shift the random variables of the Monte-Carlo simulation in the unfavorable direction to reach a convergent POF solution

with many fewer simulations. Each vessel failure was adjusted by multiplying the ratio of the probability density function for that random variable in the shifted distribution versus that in the unshifted distribution. For importance sampling with multiple variables such as the current case, the probability ratios are multiplied or summed, depending on how the associated variables occur in the problem. The vessel POF was obtained by first summing the final ratio for each failed vessel until the specified number of simulations has been performed, then dividing this sum by the specified number of simulations. Examples showing the convergence of POF with respect to the number of iterations were provided in the BWRVIP-05 report for sample single-variable problems. In Reference 2, the BWRVIP provided similar convergence information for sample multi-variable problems, which are more pertinent to the current application. A detailed parametric evaluation of sampling strategies and shifting parameters was also provided there. However, the independent simulations performed by the staff using VIPER indicated the contrary.

Some other concerns with the importance sampling scheme used in VIPER have also been identified. When the distributions for the number of manufacturing flaws and SCC flaws have importance sampling performed, the ratios of the density functions corresponding to manufacturing flaws and SCC flaws are multiplied. Since either an SCC flaw or a manufacturing flaw may fail in a given vessel simulation, it appears inappropriate to multiply these two ratios. It would appear that it is more appropriate to use only the ratio for the type of flaw which fails (i.e., manufacturing or SCC).

Another concern involves the importance sampling weights when a very low POF is being estimated. In such a case, the observed value of the shifted distribution will tend to lie very far out in the tail of the unshifted distribution, so that the ratio of these density functions will tend to be very large. Unless there are a large number of simulated failures, there is a danger that the sum of the ratios (i.e., the importance sampling weights) will be dominated by a small number of very large ratios. In such a case, the sum of the ratios and hence the estimated POF would have a large variance. Thus, if the POF being estimated is very low, the importance sampling estimate may have a large variance.

Hence, the staff concludes that the accuracy and repeatability of the VIPER code is yet to be demonstrated. The staff plans to conduct additional reviews in this area.

6.4 Evaluation of the BWRVIP's Vessel POF Results

The BWRVIP benchmarked the VIPER code against the VISA-II code using some test problems, and summarized its findings in Figure 8-12 of the BWRVIP-05 report. The staff concludes that the results from using VIPER are in reasonable agreement with those from using the VISA-II code when the same inputs are used. However, as discussed in Sections 6.2.1 to 6.2.7, the staff does not believe the inputs used by the BWRVIP will produce proper results.

As described in Section 6.2.1 to 6.2.7, some of the BWRVIP's assumptions in its VIPER simulations were inappropriate. Further, the BWRVIP's PFM analysis identified only two reference cases -- one for axial welds and one for

circumferential welds. It did not consider the variability between vessels. To determine the failure probability for all the BWR RPVs, the PFM analysis must consider the variability between vessels or present the failure probabilities and associated uncertainties as a function of adjusted RT_{NDT} or other appropriate indexing factor over the range applicable to BWRs. The variability between vessels will be addressed in the staff's uncertainty analysis in Section 7. The BWRVIP used the pressure test transient in all its analyses. To provide a more risk-informed perspective, the staff examined extensively BWR transients and selected a foreign transient as the input transient for all the staff's analyses reported in the assessment. The staff's discussion on the limiting transients is provided in Sections 5 and 7 and Appendix C.

7.0 INTEGRATED PROBABILISTIC ASSESSMENT

The BWRVIP-05 report identified loss of feedwater or single safety/relief valve blowdown events as the most limiting operational transients which could challenge RPV integrity. The most limiting transients were identified as those which result in a rapid cooldown and rapid depressurization of the vessel shell welds.

Since the BWRVIP-05 report only considered risk from design basis events, the staff performed an independent review of LERs and ENs to search for events outside of design basis that could be potentially more limiting to the RPV. This review is discussed in Section 5.0 and in Appendix C. Events that occur at cold shutdown, for example, were not identified as significant in the BWRVIP-05 report. However, the staff review identified 35 potential precursors to cold overpressure events at U.S. BWRs and one actual cold overpressure event which occurred at a foreign BWR. Accounting for these precursor and actual events, the staff performed a preliminary probabilistic assessment of events that could challenge the RPV integrity at cold shutdown.

7.1 Frequency Estimation of Potential Initiating Events and Vessel Integrity Challenges

Based on the staff findings of past shutdown events as discussed in Appendix C.2, four types of events have been identified that could potentially lead to RPV cold overpressurization conditions during shutdown. These include (1) inadvertent injection (e.g., via feedwater, HPCS) as a result of spurious actuation, or failures of operator to follow procedures (act of commission), (2) excessive cold water injection into the RPV prior to or during hydrostatic testing or vessel leak testing, (3) loss of reactor water cleanup (RWCU) system while the control rod drive pumps continue to operate, and (4) intentional injection as an act of commission without procedures or management or regulatory oversight. The likelihood of these events leading to a reactor pressure vessel low temperature overpressurization challenge, based on a limited set of historical data, is discussed below.

7.1.1 Inadvertent Injection During Shutdown

As discussed in Appendix C.1 and shown in Table C-1, several inadvertent injections, including both high pressure and low pressure injections, have

occurred during cold shutdown in U.S. BWRs. An uncontrolled injection of feedwater or High Pressure Core Spray (HPCS) during shutdown could potentially result in a water-solid condition in the reactor and inadvertent repressurization of the RPV could occur with the potential to exceed the P-T curve limits. One feedwater injection event and one HPCS injection event were identified in the staff LER search. In addition, one HPCI injection event during a HPCI automatic initiation surveillance test at cold shutdown was reported. It should be noted that none of these events resulted in a violation of TS P-T limits.

The LER search revealed a more frequent occurrence of inadvertent low pressure injections during shutdown via Low Pressure Coolant Injection (LPCI) or Core Spray. According to Appendix C.1, these injection modes can also repressurize the RPV while maintaining or lowering reactor vessel temperature. As shown in Table C-1, a total of 14 low pressure injections were identified in the LER search.

Assuming that each one of the above injections (both high pressure and low pressure) were precursors, the staff estimated the probability of an inadvertent injection to be 0.023. This estimate accounted for 17 reported inadvertent injection events during an estimated 755 reactor years of U.S. BWR operation. To arrive at a probability of 0.023, 755 reactor years of BWR operation was interpreted as 755 "plant states" during which a cold overpressurization event could occur. The ratio of time at shutdown to total reactor years was accounted for by the initiating condition frequency estimated in the previous section. However, the staff notes that this probability estimate of inadvertent injection has large uncertainty and that a thorough review of industry data would be needed to obtain a more accurate estimate.

Once an inadvertent injection occurs during a cold shutdown, the last barrier to prevent such an event from leading to cold overpressurization and thus challenging the vessel integrity is the operator/system recovery action. To estimate the probability of non-recovery, the above 17 precursor events were reviewed to note reactor conditions (e.g., high moderator temperature), operator actions (e.g., operator stopping injection), and/or system recovery actions which prevented these events from propagating to a cold overpressurization condition.

Due to the complexity of incorporating these and other factors such as instrumentation failure into estimating a non-recovery probability, a bounding probability of 1/18, or 0.055, was assumed. This non-recovery probability value assumes that the next inadvertent actuation (18th of known/reported event) would result in no recovery action. This non-recovery value, in comparison, is similar to a value of 0.05 suggested in NUREG/CR-1278 to depict a human error probability reflecting moderately high stress condition for dynamic interplay between the operator and system indications.[6]

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(6) NUREG/CR-1278, "Handbook of Human Reliability Analysis With Emphasis on Nuclear Power Plant Applications," assigned "...moderately high level of stress in a reliability analysis to transients that involve shutdown of (continued...)

Based on the estimation of above parameters, the accident sequence leading to challenging the vessel integrity was estimated to be about 1.0 x 10^{-4}/Year for a U.S. BWR. This sequence involves an initiating condition (cold shutdown with a frequency of 0.075/Year)[7], inadvertent injection (0.023), and operator/system failure to recover or restore the reactor to a stable condition (0.055).

7.1.2 Hydrostatic Tests or Vessel Leak Tests

A vessel hydrostatic test is performed about once every ten years and a vessel leak test is performed once every refueling cycle. As discussed in Appendix C, during a water solid leak test or a hydrostatic test, the RPV temperature and pressure are maintained at approximately 65.5°C to 93.3°C (150°F to 200°F) and 6.9 MPa (1000 psig), respectively, and are within the P-T curve limits. These temperature and pressure conditions are required by the plant TS; the specific limits are dependent upon the level of neutron embrittlement of the RPV.

The estimated frequency of hydrostatic injections is simply related to the test frequency. However, because of the strict temperature conditions and oversight required for conducting hydrostatic tests, it is assumed that the likelihood that these tests could lead to an undesirable temperature and cold overpressurization of the RPV is negligible. As shown in Table C-1, none of the reported precursor events related to performing a hydrostatic or vessel leak test lead to a condition exceeding the P-T limits.

The difficulty in quantifying the low temperature overpressurization challenge to a BWR vessel lies in estimating a probability of operator failing to heat the vessel in preparation for its leak or hydro test. However, if this human error is assumed to have a mean probability of 1.0 x 10^{-3} per event[8], then

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[6](...continued)
the reactor and turbine, to certain tasks during startup and shutdown which must be performed within time constraints..." The human error probability of 0.05, suggested as a high end value for dynamic tasks involving interplay between the operator and system indications, was assumed for this analysis.

[7] The initiating condition reflects the plant state at the beginning of the event tree. Thus, the initiating event frequency was estimated to be the percentage of time that the reactor would be considered in a cold shutdown condition with the vessel head tensioned. The staff assumed for this study an average plant capacity factor of 0.7 with 25 percent of the shutdown period being in a cold shutdown condition with the reactor vessel head tensioned. Thus, the frequency of plant state during which a cold overpressurization event could occur was estimated to be about 0.075/Year. The staff notes that this frequency is plant specific and could vary substantially across industry.

[8] Human error probability of 1.0 X 10^{-3} was estimated from NUREG/CR-4550, Vol. 6, "Analysis of Core Damage Frequency: Grand Gulf, Unit 1 Internal Events."

estimating a frequency of RPV low temperature overpressurization challenge due to these tests can be derived by multiplying this value with the frequency of leak (and hydrostatic) tests. Assuming that a leak test is performed every 1.5 years[9], the overall frequency from both leak and hydro tests results in about 7.0×10^{-4}/Year. The estimated uncertainty range of RPV low temperature overpressurization challenge frequency associated with the RPV hydrostatic or leak test is as follows: Assuming a log normal distribution of the frequency with an error factor of 10, the median sequence frequency was estimated at 2.5×10^{-4}/Year, the 5th percentile value at 2.5×10^{-5}/Year, and the 95th percentile value at 2.5×10^{-3}/Year.

7.1.3 Loss of Reactor Water Cleanup System

Another way that a vessel could experience cold overpressurization involves a loss of RWCU system which is used for vessel water letdown operation, while the control rod drive (CRD) pumps continue to inject water into the vessel. Using a loss of RWCU frequency of 1.5×10^{-3}/Year (assumed in NUREG/CR-6143, "Grand Gulf Low Power and Shutdown Study"), the loss of RWCU during cold shutdown was estimated by multiplying this value by the fraction of the time the plant is in cold shutdown, 0.075. The probability of an operator failing to control the level via CRD was assumed to be 0.011 (also assumed in NUREG/CR-6143 to depict same accident scenario). The point estimate of this sequence frequency was calculated to be 1.2×10^{-6}/Year.

7.1.4 Intentional Injection to Support Maintenance Activities

A foreign BWR experienced a cold overpressurization event during shutdown. As described in Appendix C.2.1, a series of operator errors of commission resulted in the CRD pump continuing to run until the vessel went water-solid with no outflow from the reactor. Unlike the above precursor events, this event challenged the reactor vessel, which resulted in a conditional vessel challenge probability of 1.0.

If a total of about 1700 reactor years of world-wide BWR operation is assumed to be from the same population (i.e., no differences between operator training, plant management, maintenance, operations, and hardware) it can be assumed that the frequency of cold overpressurizing the reactor from this class of event is about 6×10^{-4}/Year. The staff notes that a thorough review of industry (world-wide) data and information is needed to confirm the actual number of world-wide years of BWR operation and to determine if other low temperature overpressurization events have occurred.

7.1.5 Summary of Transient Frequency Estimation

Based on the information and data collected via the staff LER search of past U.S. BWR events and staff assumptions, a frequency of cold overpressurization challenging vessel integrity for U.S. BWRs due to inadvertent injection was

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(9) A refueling outage is assumed to occur once per 18 months. A hydrostatic test of RPV is performed once every 10 years. However, for the purpose of this analysis, it is assumed to replace the RPV leak test for the year that hydrostatic test is performed.

estimated to be about 1.0×10^{-4}/Year. The sequence frequency from the RPV leak and hydrostatic tests was estimated to be about 7.0×10^{-4}/Year. The sequence frequency from the RWCU isolation was estimated to be about 1.0×10^{-6}/Year. The total of these frequencies, about 8.0×10^{-4}/Year, accounts for the theoretical estimation of cold overpressurization frequency which is based on limited industry data and assumptions which include leak/hydrostatic test frequency, past inadvertent injection events, fraction of time vulnerable to cold overpressurization, and human error probabilities.

In addition to the above estimates, the staff estimated a cold overpressurization frequency based on one reported cold overpressurization event occurred at a foreign reactor. The estimated sequence frequency from this event was 6.0×10^{-4}/Year.

Although the sequence frequency (6×10^{-4}/Year) estimated from an actual cold overpressurization event is in close proximity to the theoretical cold overpressurization frequency (8×10^{-4}/Year), it is noted that the two results represent different types of scenarios leading to cold overpressurization. Because of the characteristics of the single foreign event (as a potential outlier), it would be premature to conclude that the theoretical estimation of cold overpressurization frequency realistically represents frequency estimate from world-wide operating data.

However, if the comparison is made within the general context or the general class of cold overpressurization events, the theoretical estimate appears to be consistent with an estimate derived from the world-wide operating data. As mentioned previously, to better estimate the frequency of cold overpressurization events, a thorough review of industry data and information would be needed to reduce the uncertainties associated with the selected parameter values and to improve the modelling capability.

7.2 Background for Probability of Failure and Uncertainty Analyses

Integrated PFM analysis was performed by the staff to determine the POF for potential BWR transients that may challenge irradiation embrittled BWR reactor vessel integrity. This involves identifying the BWR transients that have non-negligible event frequencies, P(E), that may cause appreciable conditional probability of vessel failure, $P(F|E)$, given that the transient under consideration occurs. The frequency of vessel failure, P(F), is determined by multiplying the estimated P(E) by the estimated $P(F|E)$.

To determine the POF of all BWR RPVs, the analysis must consider the variability between reactor vessels. An uncertainty analysis was performed to determine the effect of the variability of the input parameters (e.g., flaw size distribution, neutron fluence, unirradiated reference temperature, weight percent copper, weight percent nickel, and K_{Ic}) in the PFM analysis on the vessel conditional POF.

The uncertainty analyses are performed by shifting the mean value of a given random variable by 2 sigma (σ) in the adverse direction, while holding the other random variables at the reference case level, and determining the effect on POF.

7.2.1 Reference Case Models

Since there are significant differences in material properties for welds fabricated by the three vendors, each vendor (B&W, CE, and CB&I) was considered a reference case to determine the conditional POF for BWR reactor vessels. The material properties for the welds fabricated by each vendor are discussed in Section 7.2.2.

7.2.2 Mean and Standard Deviation of Chemistry and Material Data

The probability of vessel failure and the 95 percent confidence uncertainty bound between vessels are dependent upon the mean value and the standard deviation within a vessel and the standard deviation between vessels of the random variables in the analysis. The mean value and standard deviation within a vessel are used in determining the reference case failure probability. The standard deviation between vessels is used in determining the 95 percent confidence uncertainty bound. The method of determining the 95 percent confidence bound is discussed in Section 7.2.4.

Table 7-1 shows the three reference cases for the staff's fracture analysis of the BWR reactor vessels that were fabricated by CE, B&W, and CB&I. The mean values, the standard deviation within a vessel and the standard deviation between vessels for the unirradiated reference temperature, percent copper and percent nickel for CE and B&W fabricated welds were determined from the data in an SER of the impact of increased variability in chemistry on the RT_{PTS} value of PWR reactor vessels (Reference 32). The corresponding values for the CB&I vessels were derived by analysis of the data contained in the RVID. Based on the similarity of materials, the analysis results for the NYS-fabricated vessel should be similar to the results from the B&W fabricated vessels, and the analysis results for the Hitachi-fabricated vessel should be similar to the results from the CB&I fabricated vessels.

BWR reactor vessels beltline welds that were fabricated by CE used the SAW process. The mean copper content from the RVID for SAW welds is virtually the same as the generic copper value for CE fabricated welds. The mean nickel content from the RVID for the SAW welds is lower than the generic nickel value for CE fabricated welds. Hence, utilizing the generic chemistry data for CE fabricated SAW welds is reasonable.

BWR reactor vessels beltline welds that were fabricated by B&W used the SAW and ESW processes. The ESW weld chemistries for each BWR beltline weld are contained in the RVID. The mean copper content from the RVID for ESW welds is virtually the same as the generic copper value for B&W fabricated SAW welds. The mean nickel content from the RVID for the ESW welds is lower than the generic nickel value for B&W fabricated SAW welds. Hence, combining data for these two processes appears reasonable.

BWR reactor vessels beltline welds that were fabricated by CB&I used the SAW and SMAW processes. CB&I generic chemistry values were derived from data contained in the RVID. The mean values for copper and nickel contents were the average of all data from both SAW and SMAW welds. The data from both of these welds are low in copper and high in nickel, and appear to overlap. Hence, combining data for these two processes seems reasonable. Table 7-1

shows the various input parameters for the three reference cases, and provides a list of six uncertainty cases that are expected to provide information on uncertainties in the analysis model.

7.2.3 Mean and Standard Deviation of Neutron Fluence

The mean values for the neutron fluence, the standard deviation of the neutron fluence within a weld, and the standard deviation of the neutron fluence between vessels were determined from data reported in the RVID and from the Cooper surveillance program (Reference 33). The RVID indicates that the mean value of the peak neutron fluence are 0.174×10^{19} n/cm², 0.073×10^{19}n/cm², 0.263×10^{19}n/cm², and the standard deviation of the neutron fluence are 0.0929×10^{19}n/cm², 0.0327×10^{19}n/cm², 0.2224×10^{19}n/cm², for CE, B&W and CB&I fabricated vessels, respectively. The Cooper surveillance report indicates that the minimum value in a BWR is 45 percent of the peak value. Using this data, the staff determined the mean value for the neutron fluence and the standard deviation within a weld for the generic CE, B&W and CB&I vessels.

The standard deviation of the analysis method was assumed to be 20 percent of the peak neutron fluence. The standard deviation of the values of neutron fluence between vessels was assumed to be the square root of the sum of the squares of the standard deviation of the analysis method and standard deviation of the vessel fluence for the CE, B&W and CB&I vessels.

7.2.4 Flaw Density and the Flaw Size Distribution

The staff determined the best-estimate flaw size distribution from the inspection of the PVRUF vessel welds (Reference 24). The inspections were performed using the synthetic aperture focusing technique for ultrasonic testing (SAFT-UT). The PVRUF inspection data was from welds that had not been placed in service, but were subject to PSI. Hence, the defects reported were those resulting from manufacturing. From Table 7-2, there are a total of 148 indications at weld and fusion locations in the near surface zone. Given that a flaw exists, the distribution of crack size is determined in two stages. From Table 7-2, 134 of the 148 near surface cracks were less than 0.0787 inch (2 mm) in length. Accordingly, the probability of a crack size that is equal to 0.0787 inch is assumed to be the ratio of 134 to 148, or 0.9054. For cracks larger than 0.0787 inch, data for deeper indications was combined with the near surface data, because there was relatively little near surface data (14 near surface and 42 deep cracks) and because there is no reason to believe that the crack size distributions for cracks larger than 0.0787 inch are significantly different for near surface and deep cracks.

The staff plotted the frequencies of crack sizes from Table 7-2 on a semi-log paper and found that the best fit line has a slope approximately equal to -7.62. For the values of crack size considered, the Marshall distribution with PSI is essentially a single exponential with a slope of -6.94, and this also fits the data quite well when plotted. Accordingly, it was assumed that the crack size distribution for near surface flaws greater than 0.0787 inch is given by a Marshall distribution with PSI. The flaw size distribution for "x" greater than 0.0787 inch (0.2 cm) is tabulated in Table 7-3. Equation (2) in Section 6.2.1 represents this flaw size distribution in the cumulative form.

The staff's best-estimate flaw density is based on the observed number of indications in the near surface zone from the PVRUF data. Here, only weld and fusion locations are counted. Table 7-2 indicates that there are a total of 148 such indications. Because 0.14875 cubic meter of weld volume was inspected, the flaw density in the near surface zone is calculated to be 995 flaws per cubic meter.

Table 7-4 provides the upper 95 percent confidence bound flaw size distribution derived from analysis of the PVRUF inspection data. The uncertainty of the assumed flaw density and flaw size distribution is determined as follows:

> Because the assumed flaw density and flaw size distribution are based on counts, which have a Poisson distribution, confidence bounds for them can be derived from confidence bounds on a Poisson mean. The best-estimate flaw density is based on 148 indications that were observed in the near surface of the PVRUF data. An upper 95 percent confidence bound is found by replacing 148 by one-half of the 95th percentile of a chi-square distribution with 2(148) + 2 = 298 degrees of freedom. This quantity is equal to 170. Therefore, an upper 95 percent confidence bound for the flaw density is 170 divided by the volume of the PVRUF welds (170/0.14875 = 1143 flaws/m^3). This is a 14.9 percent increase over the best-estimate value.
>
> The uncertainty in the crack size distribution is driven by the uncertainty in the probability of a crack size less than 2 mm (0.0787 inches) in length. Because this probability is close to one, its uncertainty is driven by its complement, the probability of a crack size greater than 2 mm. From Table 7-2, this probability is estimated at 14/148 = 0.0946. From a table for Poisson confidence bounds, an upper 95 percent confidence bound for this probability is 21.89/148 = 0.1479. A lower 95 percent confidence bound for the probability of a crack size less than 2 mm is therefore, 1 - 0.1479 = 0.8521. This value is used for F(.0787) in the formula for F(a) to calculate the upper 95 percent confidence bound on the flaw size distribution, which is given by Table 7-4.

Since BWRs operate in an environment that is susceptible to SCC, the staff assumed that service would induce SCC in the cladding. The staff assumed that the SCC would penetrate through the clad and combine with the manufacturing flaws. As a result of this assumption, the flaw sizes in Table 7-3 and 7-4 were increased by 0.20 inch (0.508 cm, "typical" thickness of BWR clad) to account for service induced stress corrosion cracking.

Generally the assumption that clad cracks would line up with manufacturing defects would be considered a conservative assumption. However, in reactor vessels that have been subject to repair on the inside of the weld or repairs to the cladding, it is possible that the repairs could result in cracks in the weld and increase the susceptibility for SCC. In this case, it would be likely that the SCC would line up with the cracks initiated by the repair. Cracks penetrating a repair in the clad and progressing into the adjacent base metal were observed in the reactor vessel head of Quad Cities Unit 2, (see NRC Information Notice 90-29, "Cracking of Cladding and Its Heat-Affected Zone in the Base Metal of a Reactor Vessel Head," dated April 30, 1990).

The number of flaws assumed in the PFM analysis was a product of the flaw density and the volume of a generic BWR weld. The volume of a generic BWR circumferential weld was calculated to be 0.109 cubic meter (for a generic vessel of internal radius 112.6 inches (286 cm), vessel beltline region thickness of 5.45 inches (13.8 cm) and weld thickness of 1.75 inches or 4.44 cm). For the reference case, 108 flaws per weld (994 X 0.109) was assumed in each analysis. In the uncertainty study for the effect of the change in flaw size and density, the upper 95 percent confidence bound for the number of flaws was increased by 14.9 percent.

7.2.5 Residual Stresses and Crack Driving Forces for a Limiting Transient in BWR Vessels

A limiting cold overpressure transient described in Appendix C (Section C.2.1), is considered for analyses. This event consists of an internal pressure of 1150 psi at an isothermal temperature of 88°F. "Typical" BWR vessel geometric dimensions are assumed, i.e., an internal radius of 112.5 inches, vessel wall base-metal thickness of 5.25 inches, and a clad thickness of 0.2 inch. Thermo-mechanical properties of the vessel are assumed as following:

Thermal conductivity (BTU/hr.ft.°F) = 24 (base metal), 10 (cladding)
Specific heat (BTU/lb.°F) = 0.12 (base-metal, cladding)
Density (lb/ft^3) = 489 (base metal, cladding)
Elastic Modulus (ksi) = 28,000 (base-metal), 22,000 (cladding)
Thermal expansion coefficient (/°F) = 6.9 X 10^{-6} (base-metal), 9 X 10^{-6} (cladding)
Poisson's ratio = 0.3 (base metal, cladding)
Cladding stress-free reference temperature (°F) = 515
Heat-transfer coefficient (BTU/hr.ft.^{2}°F) = 1000

Due to isothermal internal pressure conditions in the vessel, hoop and axial stresses are due to internal pressure, clad base-metal differential thermal expansion, and the residual stresses due to the weld in process for the axial and circumferential welds. Residual stress distribution through vessel thickness obtained from measurements in an actual PWR vessel axial welds are assumed to be applicable for the BWR vessels. The weld residual stress has a maximum amplitude of 6.5 ksi at the inner surface of the vessel and can be approximated with a third degree polynomial, for 5.45 inches $\geq x \geq 0$ inch, as $\sigma_{residual\ stress} = [8.437 - 11.699x + 3.565x^2 - 0.292x^3]$ ksi.

The crack driving force, $K_{1Applied}$, is calculated for axial and circumferential semi-elliptical surface cracks of aspect ratios (total crack length/crack depth) of 2, 6, and 10. The $K_{1Applied}$ values are determined in FAVOR code at 10 locations along the crack periphery at 10 degrees elliptical angle apart between the free-surface point of the crack and the maximum depth. The condition for cleavage crack growth ($K_{1Applied} \geq K_{1c}$) is checked at each of the 10 locations in the base-metal between the free-surface and the maximum depth points. In FAVOR computations, a statistical sampling is performed on the crack aspect-ratios of 2, 6, and 10, based on the assumption of a uniform distribution for the aspect ratios. This will make the conditional POF, $P(F|E)$, to be smaller for the FAVOR code as compared to VIPER which assumes that the flaw is infinitely long.

7.2.6 Uncertainty Distribution for the Conditional Probability of Vessel Failure

The between-vessel uncertainty of the conditional probability of vessel failure, $P(F|E)$, is best described by an uncertainty distribution (explained below in Section 7.4) because the mean values of the parameters which determine $P(F|E)$ all vary from vessel to vessel. There are five such parameters: neutron fluence, unirradiated reference temperature, percent copper, percent nickel and K_{Ic}. These parameters are assumed to be independently and normally distributed with means and standard deviations between vessels given by Table 7-5. (The standard deviations within welds given in Table 7-5 are used in the Monte Carlo calculation of $P(F|E)$).

7.3 Conditional Failure Probabilities

Tables 7-6, 7-7 and 7-8 present the conditional POF, $P(F|E)$ for axial and circumferential welds in CE, B&W and CB&I fabricated vessels for the reference cases as well as for the uncertainty cases with shifting of mean values one at a time by 2σ for the fluence, copper, nickel, initial RT_{NDT}, K_{Ic} fracture toughness , and new flaw density and depth distributions from PVRUF vessel data for cleavage crack growth. For the CB&I fabricated vessels with circumferential cracks, no failure was predicted for 10 million simulations performed for the various cases, and as such the conditional failure probability $P(F|E)$ is less than 1.0×10^{-7}.

7.4 Uncertainty Analysis

A response surface approach is used to obtain an uncertainty distribution for $P(F|E)$. As an approximation, $\ln\{P(F|E)\}$ is assumed to be a linear function of the first five parameters listed in Section 7.3 (this is the response surface). Therefore, $\ln\{P(F|E)\}$ has a normal distribution or, equivalently, $P(F|E)$ has a log-normal distribution. The coefficients in the response surface are estimated by performing sensitivity runs one at a time for each of these five parameters and comparing the resultant failure probabilities with the reference case failure probability. Based on these coefficients and the between-vessel standard deviations from Table 7-5, the standard deviation of $\ln\{P(F|E)\}$ is calculated. Between-vessel uncertainty bounds for $P(F|E)$ are then calculated from this standard deviation and normal distribution tables. For example, a multiplier of 1.645 for the standard deviation is used to calculate a 95 percent uncertainty bound.

The 95 percent confidence bounds between vessels that results from the uncertainty analysis are reported in Table 7-9. The best-estimate failure probabilities and the 95 percent confidence bounds for the reference case circumferential welds could not be determined accurately because insufficient failures resulted from these analyses. These analyses were based on 10 million vessel simulations.

The failure frequency due to cold overpressure events is denoted by $Fr(E,F)$, where E denotes the class of overpressure events and F denotes vessel failure. This is decomposed by writing $Fr(E,F) = Fr(E)P(F|E)$, where $Fr(E)$ is the frequency of the class E and $P(E|F)$ is the conditional probability of vessel failure given that an initiating event from E has occurred.

Based on the foreign cold overpressure transient, a point estimate of Fr(E) is $1/1700 = 5.88 \times 10^{-4}$, where the denominator is the total number of BWR reactor years of world-wide experience. From tables for confidence limits for the expectation of a Poisson variable, a 95% upper confidence bound for Fr(E) is $4.74/1700 = 2.79 \times 10^{-3}$. This uncertainty bound can be combined with an uncertainty bound for P(F¦E) to obtain an overall uncertainty bound for Fr(E,F).

7.5 Sensitivity to Flaw Density and the Flaw Size Distribution

In contrast to the uncertainty in P(F¦E) due to the five parameters listed in Section 7.2.6, the uncertainty in P(F¦E) due to flaw density and the flaw size distribution is best described by a sensitivity study. While the five parameters vary from vessel to vessel, the flaw density and flaw size distribution are assumed to be constant from vessel to vessel. They are uncertain because they are estimated from data as described in Section 7.2.4. Accordingly, the uncertainty in P(F¦E) due to flaw density and the flaw size distribution is described by the ratio of P(F¦E) based on 95 percent upper confidence bounds for flaw density and the flaw size distribution to the reference case value of P(F¦E). The pertinent POF results have been reported in Tables 7-6 to 7-8. They have been reproduced in Table 7-10 along with the results of the sensitivity study. However, since insufficient failures were reported for the reference cases for the circumferential welds, their sensitivity to flaw size and density could not be determined.

7.6 Sensitivity to Inservice Inspection

The staff performed sensitivity studies to determine the effect that inservice inspection would have on the POF. Two POD of flaws were evaluated. They are the POD of flaws resulting from the PISC studies and the POD of flaws defined by Method C in the VIPER Code. The POD from the PISC studies was developed by Pacific Northwest Laboratories (PNL). Figure 7-1 compares the POD from the PISC study and Method C. The staff believes that the PISC based POD should be considered a lower bound for ultrasonic inspection methods that have been qualified to Appendix VIII of ASME Code Section XI. The conclusion is based on discussions with technical specialist at Pacific Northwest Laboratories. The VIPER Method C would be considered an upper bound POD for ultrasonic inspection methods.

PNL developed flaw size distributions based on the PISC and Method C PODs. The flaw size distributions included flaw sizing error of ±0.2589 inch (±0.658 cm) and flaw acceptance of 0.2071 inch (0.526 cm). Flaws less than 0.2071 inch (0.526 cm) would be acceptable to ASME Code inspection standards and would not be required to be removed following inspection. The best-estimate flaw size distributions resulting from PISC and Method C and the 95% confidence bound flaw size distribution resulting from PISC are shown in Table 7-11. The best-estimate flaw distributions were developed from the staff's flaw size distribution that is shown in Table 7-3, the 95 percent confidence flaw distribution was developed from Table 7-4.

The POF for the flaw size distributions from PISC and Method C are compared to the POF resulting from the reference cases in Table 7-12. The reference cases used the flaw size distributions in Table 7-3. The sensitivity to inservice

inspection is the ratio of the POF from the PISC or Method C PODs to the POF from the reference case. Lower ratios would indicate that the inservice inspection method would have greater impact on reducing the POF of the reactor vessel. Table 7-12 also contains the results from additional sensitivity studies performed to determine the effect of using the POD from the PISC studies on the staff results which were based on a 95 percent confidence bound flaw size distribution.

Table 7-1
Three Reference Cases for Staff's BWRVIP-05 Fracture Analysis of RPVs

Random Variable (μ=Mean), (σ=Std. Dev.)	CE Vessels (Ref. Case 1)	B&W Vessels (Ref. Case 2)	CB&I Vessels (Ref. Case 3)
μ (Initial RT_{NDT})	- 56°F	- 5°F	- 56°F
σ (Initial RT_{NDT})	16.7°F	19.8°F	16.7°F
μ (Fluence x10^{19})	0.126 n/cm²	0.053 n/cm²	0.19 n/cm²
σ (Fluence x10^{19})	0.024 n/cm²	0.01 n/cm²	0.036 n/cm²
μ (Copper)	0.226 wt %	0.287 wt %	0.04 wt %
σ (Copper)	0.062 wt %	0.06 wt %	0.019 wt %
μ (Nickel)	0.76 wt %	0.6 wt %	0.93 wt %
σ (Nickel)	0.032 wt %	0.0155 wt %	0.079 wt %
σ (ΔRT_{NDT})	24°F	24°F	24°F
μ (K_{Ic})	μ from ASME	μ from ASME	μ from ASME
σ (K_{Ic})	0.147xμ(K_{Ic})	0.147xμ(K_{Ic})	0.147xμ(K_{Ic})
μ (K_{Ia})	μ from ASME	μ from ASME	μ from ASME
σ (K_{Ia})	0.1xμ(K_{Ia})	0.1xμ(K_{Ia})	0.1xμ(K_{Ia})
Flaw Size & Density	μ and σ from PVRUF data	μ and σ from PVRUF data	μ and σ from PVRUF data
Sensitivity Studies by Shifting to Following Mean Values "One at a Time"			
μ (Fluence x E19)	0.326 n/cm²	0.125 n/cm²	0.651 n/cm²
μ (initial RT_{NDT})	-50°F	1°F	-50°F
μ (Copper)	0.256 wt %	0.317 wt %	0.05 wt %
μ (Nickel)	0.776 wt %	0.608 wt %	0.96 wt %
μ (K_{Ic})	0.94xASME μ	0.94xASME μ	0.94xASME μ
Flaw Size & Density	(μ + 2σ) from PVRUF data	(μ + 2σ) from PVRUF data	(μ + 2σ) from PVRUF data

Table 7-2
Summary of Data from SAFT-UT Inspection of PVRUF Vessel Welds

PVRUF: Number of Near Surface Zone Indications by Category															
09/15/95	Through-Wall Extent of the Indications (DZ)														
	< 2 mm	2-3 mm		3-4 mm		4-5 mm		5-6 mm		6-7 mm		7-8 mm		TOTAL > 2 mm	
Location		V	P	V	P	V	P	V	P	V	P	V	P	V	P
Clad	978	2	0	2	0	0	0	0	0	0	0	0	0	4	0
Weld	87	0	5	0	2	0	0	0	0	2	1	0	1	2	9
Fusion (HAZ)	47	0	0	1	1	0	0	0	0	0	0	1	0	2	1
Base	392	4	14	1	4	0	2	0	0	1	0	0	1	6	21
Total	1504	6	19	4	7	0	2	0	0	3	1	1	2	14	31
				Total Number Characterized > 2 mm 45											
							Total Number < 2 mm 1504								
							Total Number 1549								

Table 7-3
NRC Staff's Flaw Size Distribution
(Best Estimate)

FLAW SIZE a (inches)	CDF	Probability Distribution Pr{a}
0.0787	0.905405	0.905405
0.2159	0.963304	0.057899
0.3532	0.985740	0.022436
0.4904	0.994435	0.008695
0.6276	0.997817	0.003382
0.7649	0.999138	0.001321
0.9021	0.999656	0.000518
1.0394	0.999861	0.000205
1.1766	0.999942	0.000081
1.3138	0.999976	0.000034
1.4511	0.999989	0.000013
1.5883	0.999995	0.000006
1.7225	0.999998	0.000003
1.8628	0.999999	0.000001
2.0000	1.000000	0.000001

TABLE 7-4
Upper 95 percent Confidence Bound Flaw Size
Distribution from PVRUF Data

FLAW SIZE a (inches)	CDF	Probability Distribution Pr{a}
0.0787	0.852095	0.852095
0.2159	0.942628	0.090533
0.3532	0.977703	0.035075
0.4904	0.991299	0.013596
0.6276	0.996587	0.005288
0.7649	0.998652	0.002065
0.9021	0.999462	0.000810
1.0394	0.999783	0.000321
1.1766	0.999909	0.000127
1.3138	0.999962	0.000053
1.4511	0.999983	0.000021
1.5883	0.999992	0.000009
1.7225	0.999997	0.000005
1.8628	0.999998	0.000001
2.0000	1.000000	0.000002

TABLE 7-5
Values of Material Property Parameters Used in the BWR RPV Uncertainty Analysis

	Combustion Engineering RPV's			Babcock & Wilcox RPV's			Chicago Bridge & Iron RPV's		
Variables	Mean Value	Standard Deviation Within A Vessel	Standard Deviation Of Means Between Vessels	Mean Value	Standard Deviation Within A Vessel	Standard Deviation Of Means Between Vessels	Mean Value	Standard Deviation Within A Vessel	Standard Deviation Of Means Between Vessels
Neutron Fluence ($X\ 10^{19}$ n/cm2) (E>1MeV)	0.126	0.024	0.10	0.053	0.010	0.036	0.191	0.036	0.23
Unirradiated Reference Temperature (°F)	-56	16.7	3	-5	19.8	3	-56	16.7	3
Weight Percent Copper	0.226	0.062	0.015	0.287	0.060	0.015	0.04	0.019	0.005
Weight Percent Nickel	0.76	0.032	0.008	0.60	0.0155	0.004	0.93	0.079	0.015
Static Fracture Toughness (K_{Ic})	ASME Mean	14.7% of ASME Mean	3% of ASME Mean	ASME Mean	14.7% of ASME Mean	3% of ASME Mean	ASME Mean	14.7% of ASME Mean	3% of ASME Mean

TABLE 7-6
Conditional Probability of Failure P(F|E) for CE Fabricated BWR Vessels

SN	Case Description	Inner Surface (RT_{NDT} + 2σ)°F	Conditional POF Circumferential Flaws	Conditional POF Axial Flaws
1	Reference Case 1	93.2	$< 1.0 \times 10^{-7}$	1.5×10^{-3}
2	Mean Fluence = 0.326×10^{19} n/cm²	137.5	2.1×10^{-5}	3.7×10^{-2}
3	Mean Copper = 0.256 wt %	97.4	$< 1.0 \times 10^{-6}$	2.9×10^{-3}
4	Mean Nickel = 0.776 wt %	94.4	$< 1.0 \times 10^{-6}$	2.5×10^{-3}
5	Mean Initial RT_{NDT} = -50°F	99.2	$< 1.0 \times 10^{-6}$	3.8×10^{-3}
6	(μ+2σ) Flaw Density & Flaw Depth Distribution from PVRUF Vessel Data	93.2	6.0×10^{-7}	3.1×10^{-3}
7	Mean K_{IC} = 0.94 x ASME Mean K_{IC}	93.2	1.0×10^{-6}	4.3×10^{-2}

<u>Reference Case definition:</u>
Mean Fluence = 0.126×10^{19} n/cm²
Mean Copper = 0.226 wt %
Mean Nickel = 0.76 wt %
Mean Initial RT_{NDT} = -56°F

TABLE 7-7
Conditional Probability of Failure P(F|E) for B&W Fabricated BWR Vessels

SN	Case Description	Inner Surface (RT_{NDT} + 2σ)°F	Conditional POF Circumferential Flaws	Conditional POF Axial Flaws
1	Reference Case 1	114.5	1.0×10^{-6}	9.5×10^{-3}
2	Mean Fluence = 0.125×10^{19} n/cm²	145.1	2.5×10^{-5}	5.6×10^{-2}
3	Mean Copper = 0.317 wt %	135.1	1.0×10^{-6}	1.1×10^{-2}
4	Mean Nickel = 0.608 wt %	114.9	$< 1.0 \times 10^{-6}$	9.7×10^{-3}
5	Mean Initial RT_{NDT} = 58.3°F	120.5	$< 1.0 \times 10^{-6}$	9.8×10^{-3}
6	(μ+2σ) Flaw Density & Flaw Depth Distribution from PVRUF Vessel Data	114.5	4.0×10^{-6}	1.5×10^{-2}
7	Mean K_{IC} = 0.94 x ASME Mean K_{IC}	114.5	6.0×10^{-6}	2.0×10^{-2}

<u>Reference Case definition:</u>
Mean Fluence = 0.053×10^{19} n/cm²
Mean Copper = 0.287 wt %
Mean Nickel = 0.6 wt %
Mean Initial RT_{NDT} = -5°F

TABLE 7-8
Conditional Probability of Failure P(F|E) for CB&I Fabricated BWR Vessels

SN	Case Description	Inner Surface (RT_{NDT} + 2σ)°F	Conditional POF Circumferential Flaws	Conditional POF Axial Flaws
1	Reference Case 1	32.7	N/A	4.6×10^{-6}
2	Mean Fluence = 0.651 x E19 n/cm²	50.0	N/A	1.7×10^{-4}
3	Mean Copper = 0.05 wt %	40.5	N/A	---
4	Mean Nickel = 0.96 wt %	32.7	N/A	---
5	Mean Initial RT_{NDT} = -50°F	38.7	N/A	---
6	(μ+2σ) Flaw Density & Flaw Depth Distribution from PVRUF Vessel Data	32.7	N/A	3.8×10^{-6}
7	Mean K_{IC} = 0.94 x ASME Mean K_{IC}	32.7	N/A	---

<u>Reference Case definition:</u>
Mean Fluence = 0.191×10^{19} n/cm²
Mean Copper = 0.04 wt %
Mean Nickel = 0.93 wt %
Mean Initial RT_{NDT} = -56°F

Table 7-9
Results of Uncertainty Analyses

Fabricator	Weld Orientation	Reference Case Failure Probability	95% Conf. Uncertainty Bound Between Vessels
Babcock & Wilcox (B&W)	Axial	9.5×10^{-3}	4.6×10^{-2}
	Circumferential	1.0×10^{-6} [a]	$> 1.4 \times 10^{-5}$ [a]
Combustion Engineering (CE)	Axial	1.5×10^{-3}	2.9×10^{-2}
	Circumferential	$<1.0 \times 10^{-7}$ [a]	--- [a]
Chicago Bridge & Iron (CB&I)	Axial	4.3×10^{-6}	$> 8.9 \times 10^{-5}$ [b]
	Circumferential	$<1.0 \times 10^{-7}$ [a]	--- [a]

Notes:

[a] Insufficient or no failures to accurately determine reference case failure probability and sensitivity to flaw size, flaw density and inservice inspection.

[b] Failure Probability based on analyses of the neutron fluence only. Other variables were not analyzed for this case. However, the other variables in the other cases only made a small impact in the increase in the 95 percent confidence uncertainty bound.

Table 7-10
Results of Sensitivity Analyses

Fabricator	Weld Orientation	Reference Case Failure Probability	Sensitivity to Flaw Size and Density[b]
Babcock & Wilcox (B&W)	Axial	9.5×10^{-3}	1.6
	Circumferential	1.0×10^{-6} [a]	--- [a]
Combustion Engineering (CE)	Axial	1.5×10^{-3}	2.1
	Circumferential	$<1.0 \times 10^{-7}$ [a]	--- [a]
Chicago Bridge & Iron (CB&I)	Axial	4.3×10^{-6}	>1.0 [c]
	Circumferential	1.0×10^{-7} [a]	--- [a]

[a] Insufficient or no failures to accurately determine reference case failure probability and sensitivity to flaw size, flaw density and inservice inspection.
[b] The sensitivity to flaw size and density is the ratio of the probability of failure using the 95 percent confidence bound for flaw size and density to the probability of failure using the best estimate flaw size and density. (Conditional POF values taken from Tables 7-7, 7-8, and 7-9).
[c] Observed ratio was less than 1. Insufficient failures to accurately determine ratio.

Table 7-11
NRC Staff's Flaw Size Distribution With Adjustment for Inservice Inspection

	Using POD from PISC		Using POD from VIPER Method C		Using POD from PISC	
FLAW SIZE a (inches)	CDF (best-estimate)	Probability Distribution Pr{a} (best-estimate)	CDF (best-estimate)	Probability Distribution Pr{a} (best-estimate)	CDF (95% confidence)	Probability Distribution Pr{a} (95% confidence)
0.0787	0.950135460	0.950135460	0.964561300	0.964561300	0.922032500	0.922032500
0.2159	0.988873790	0.038739190	0.993327320	0.028766020	0.982606520	0.060574020
0.3532	0.998149148	0.009275358	0.999282796	0.005955476	0.997107020	0.014500500
0.4904	0.999548076	0.001398928	0.999721720	0.000438924	0.999294464	0.002187444
0.6276	0.999881868	0.000333792	0.999890820	0.000169100	0.999816371	0.000521907
0.7649	0.999956837	0.000074696	0.999956870	0.000066050	0.999933563	0.000117192
0.9021	0.999982737	0.000025900	0.999982770	0.000025900	0.999974063	0.000040500
1.0394	0.999992987	0.000010250	0.999993020	0.000010250	0.999990113	0.000016050
1.1766	0.999997037	0.000004050	0.999997070	0.000004050	0.999996463	0.000006350
1.3138	0.999998737	0.000001700	0.999998770	0.000001700	0.999999113	0.000002650
1.4511	0.999999387	0.000000650	0.999999420	0.000000650	1.000000163	0.000001050
1.5883	0.999999687	0.000000300	0.999999720	0.000000300	1.000000613	0.000000450
1.7255	0.999999837	0.000000150	0.999999870	0.000000150	1.000000863	0.000000250
1.8628	0.999999887	0.000000050	0.999999920	0.000000050	1.000000913	0.000000050
2.0000	0.999999937	0.000000050	0.999999970	0.000000050	1.000001063	0.000000100

Table 7-12
Results of Sensitivity Analyses due to ISI

Fabricator	Weld Orientation	Sensitivity to Inservice Inspection[a] using POD from PISC (best-estimate)	Sensitivity to Inservice Inspection using POD from VIPER Method C[a] (best-estimate)	Sensitivity to Inservice Inspection using POD from PISC (95% confidence)
Babcock & Wilcox (B&W)	Axial	0.33	0.26	0.31
Combustion Engineering (CE)	Axial	0.29	0.20	0.24
Chicago Bridge & Iron (CB&I)	Axial	0.11	---[b]	---[b]

Notes:

[a] The sensitivity to inservice inspection is the ratio of the probability of failure using the flaw distribution derived from the probability of detecting flaw during inservice inspection to the probability of failure using the best estimate flaw size and density.

[b] No failure in 12 million simulations.

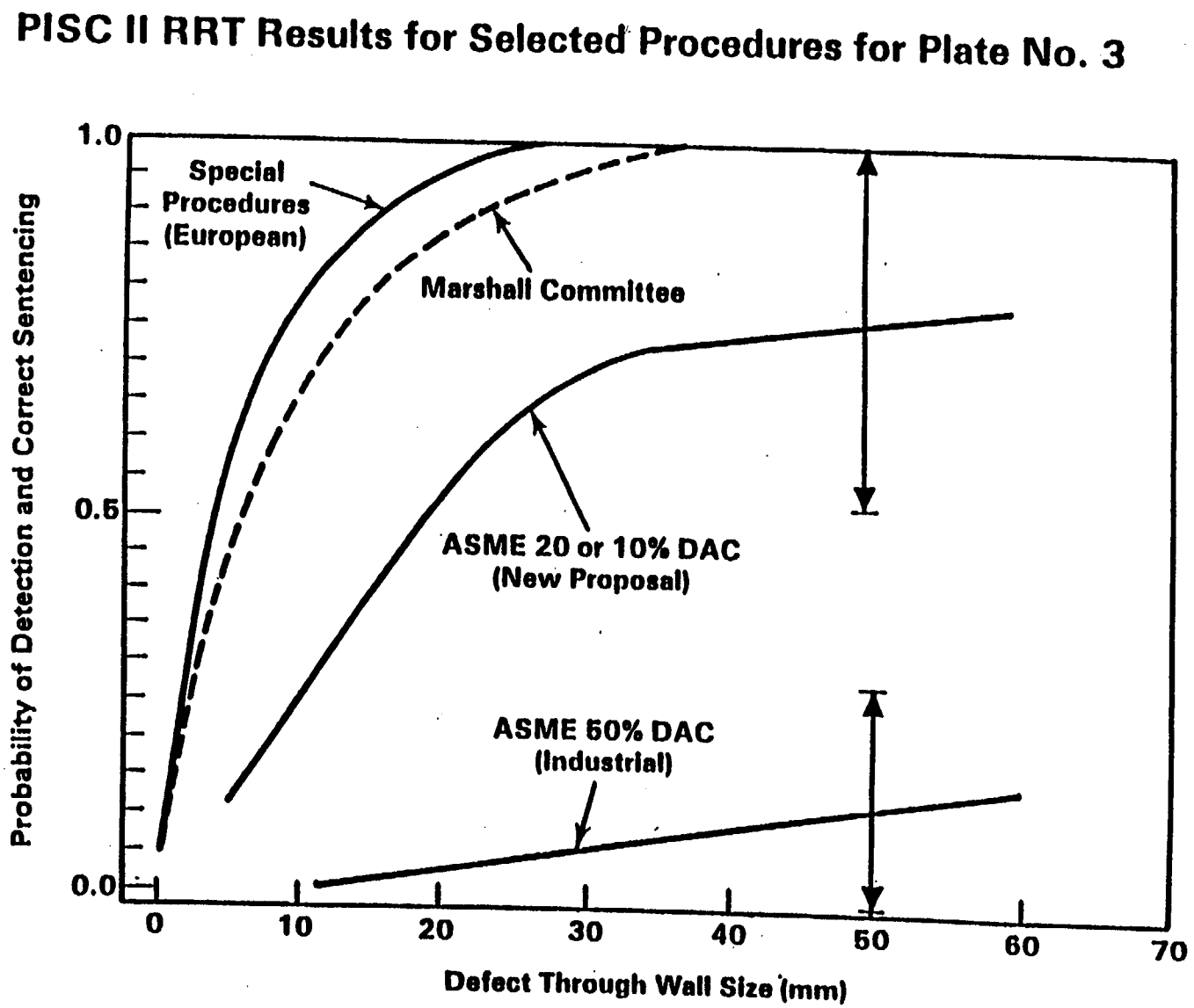

Figure Probability of Detection and Correct Sentencing Curves from PISC-II Studies (Nichols and Crutzen 1993)

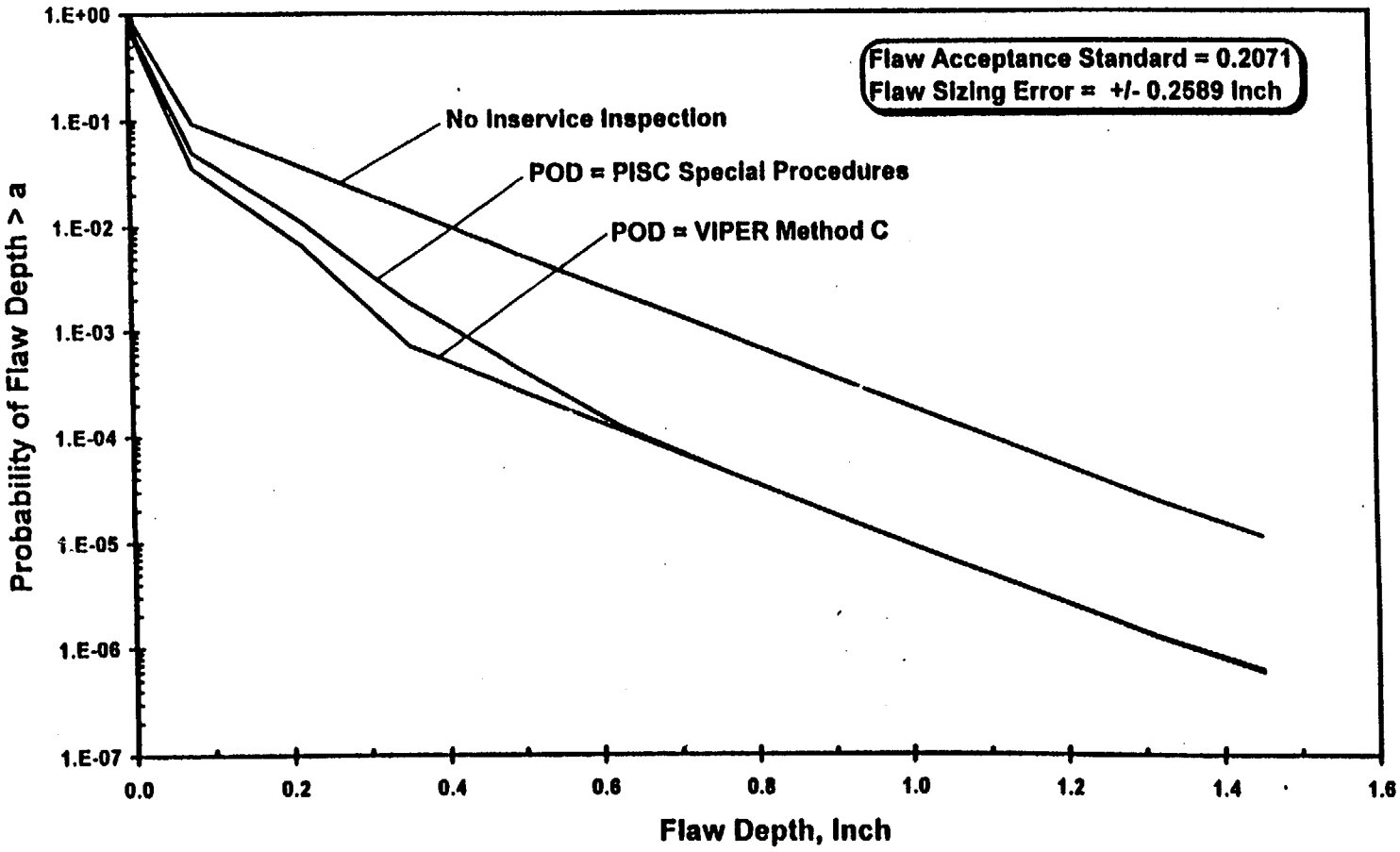

Figure Effect of Inservice Inspections with Differing POD Curves on Flaw Depth Distribution

APPENDIX A
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APPENDIX B
ACRONYMS AND INITIALISMS

°C	degrees Celsius
°F	degrees Fahrenheit
μ	mean
σ	standard deviation
ACRS	Advisory Committee on Reactor Safeguards
ARI	Alternate Rod Injection
ASME	American Society of Mechanical Engineers
ATWS	Anticipated Transient Without Scram
B&W	Babcock & Wilcox
BWR	boiling water reactor
BWRVIP	Boiling Water Reactor Vessel and Internals Project
CB&I	Chicago Bridge and Iron
CE	Combustion Engineering
CFR	*Code of Federal Regulations*
cm	centimeters
CRD	control rod drive
CS	core spray
CST	condensate storage tank
d	expected number of defective circumferential welds
DAC	distance-amplitude-correction
E	class of overpressure events
EDDY	eddy current inspection
EN	event notifications
ESW	electroslag welding
F	vessel failure
Fr(E)	frequency of the class E
Fr(E,F)	failure frequency due to cold overpressure events
FULLV	full-V ultrasonic angle beam examination
GE	General Electric
GMAW	gas metal arc welding
gpm	gallons per minute
HPCS	high pressure core spray
HPCI	high pressure core injection
ID	inner diameter
ISI	in-service inspection
K_{Ic}	static fracture toughness
K_{Ir}	reference stress intensity
LAS	low alloy steel
LER	licensee event reports
LOCA	loss-of-coolant accident
LPCI	low pressure coolant injection
L/s	liters per second
m	number of circumferential welds
mm	millimeters
MPa	megaPascals
n	number of axial welds
n/cm²	fluence, neutrons per square centimeter
NDE	non-destructive examination
NRC	Nuclear Regulatory Commission

NYS	New York Shipbuilding
N(x)	function to describe flaw size distribution
OD	outer diameter
p	proportion of defective welds in the parent population
P(E)	non-negligible event frequencies
P(E|F)	conditional probability of vessel failure given that an initiating event from E has occurred
P(F)	frequency of vessel failure
P(F|E)	conditional probability of vessel failure
PFM	probabilistic fracture mechanics
PISC	Plate Inspection Steering Committee
POD	probability of detection
POF	probability of failure
PSI	preservice inspection
psig	pounds per square inch gage
P-T	pressure-temperature curve
PVRC	Pressure Vessel Research Committee
PVRUF	pressure vessel research user facility
PWR	pressurized water reactor
RAI	request for additional information
RHR	residual heat removal
RPV	reactor pressure vessel
RT	radiographic inspections
RT_{NDT}	reference temperature of vessel weld material
RWCU	reactor water cleanup
RVID	reactor vessel integrity database
SAFT-UT	synthetic aperture focusing technique for ultrasonic testing
SAW	submerged arc welding
SCC	stress corrosion cracking
SER	safety evaluation report
SI	Surveillance Instruction
SLIC	multi-mode ultrasonic examination
SMAW	shielded metal arc welding
SRM	Staff Requirements Memorandum
SRV	safety relief valve
TS	technical specification
UT	ultrasonic testing
x	crack length in inches

APPENDIX C
LIMITING TRANSIENTS

The BWRVIP-05 report discusses the operating characteristics of a BWR with respect to design transients and their effects on the vessel. These transients generally occur when a large steam region exists (i.e., a steam bubble is present). According to BWRVIP-05, the most limiting operational transients with respect to the vessel are loss of feedwater or single safety relief valve (SRV) blowdown events for normal and upset conditions. Normal operating temperature and pressure for a BWR RPV is 260°C (500°F) and 6.9 MPa (1000 psig). During the hydrostatic test, RPV temperature and pressure is approximately 65.5°C to 93.3°C (150°F to 200°F) and 6.9 MPa (1000 psig) and is maintained on the pressure-temperature (P-T) curve for that particular vessel. The most limiting transients for emergency and faulted operating conditions are any transient which causes or results in a rapid cooldown, and rapid depressurization, of the vessel shell welds. These transients are limiting for pre-existing cracks in the vessel shell welds. In comparison, the BWRVIP-05 report states that the water-solid leak test condition, or hydrostatic test, is limiting for small flaws in the inside diameter of the vessel.

Based on the types of events discussed above, the staff performed a sampling review of approximately 20 years of licensee event reports (LERs) and event notifications (ENs) to determine if other events (i.e., shutdown events) could be potentially more limiting to the vessel. The staff's sampling review does not encompass all LERs and ENs due to the time restraints. The staff also notes that some overpressure events during shutdown, as described in Appendix C.1, may not be reportable under 10 CFR 50.72. The preliminary staff sampling review is discussed in Appendix C.2.

Table C-1 includes the results of the staff's sampling review of LERs and ENs. The table provides the maximum pressure and temperature reached at the end of the transient. Where the information was not provided, an NP is listed.

It should be noted that, although the following are examples of the types of events that could result in cold overpressure events, these events occurred at temperatures that are high relative to the reference temperature of the vessel weld material, as required by TS. Therefore, these events did not, in and of themselves, represent significant challenges to the RPV.

C.1 Description of Rapid Depressurization and Shutdown Events

The staff evaluated the following classes of transients and shutdown events with the potential to be more limiting than the operational transients discussed in BWRVIP-05.

C.1.1 Rapid Depressurization

Rapid depressurization of the RPV during operation can occur as a result of a large break loss-of-coolant accident (LOCA), or lifting of a SRV. Most facilities have technical specifications (TSs) for the rate of cooldown (i.e., cooldown limit of 37.8°C (100°F) per hour). These TSs are in place to protect

the vessel from pressurized thermal shock. Examples of known rapid depressurization events at U.S. BWRs are as follows:

1. Hope Creek 1 had an event which cause reactor pressure to decrease from 5.3 MPa to 689.5 kPa (765 psig to 100 psig), 266.7°C to 164.4°C (512°F to 328°F) respectively, in 51 minutes on October 10, 1987.

2. Perry had an event which exceeded the cooldown rate of 37.8°C (100°F) per hour on October 27, 1987.

3. Quad Cities 1 had a stuck open relief valve at 10 percent power which caused a cooldown rate of 93.3°C (200°F) per hour for the first hour and 46.1°C (115°F) for the second hour on April 17, 1989.

To assess the POF for rapid depressurization, the staff assumed a hypothetical transient and performed a parametric study using the VISA-II code. The transient was characterized by a rapid depressurization to 10 psi (69 kPa) with the fluid temperature decreasing from 550°F to 206°F (288°C to 97°C) in 12 seconds (i.e., equivalent to a cooling rate of 1720°F or 956°C per minute). Use of this essentially instantaneous cooldown transient was meant to bound all real transients. The staff examined both axial and circumferential welds in this study. In one case, the staff intentionally increased the fluence to 5.0×10^{18} n/cm² and the initial RT_{NDT} for the axial weld to 40°F to increase the probability of producing vessel failure. However, the staff's results indicated no failure for all cases subjected to rapid depressurization transient. As a result, the severe foreign transient discussed in Appendix C.2.1 was used in the staff's independent simulations reported in Section 7.

C.1.2 Hydrostatic Testing and/or Vessel Leak Test

As discussed in Section 5.0, the hydrostatic test (or vessel leak test), is performed in the water solid condition with RPV pressure and temperature maintained on the vessel P-T curve, as is generally specified in the licensee's TSs. With the vessel in the water solid condition, any addition of water to the vessel increases RPV pressure while maintaining temperature. An example of potential water addition to the vessel is control rod drive (CRD) pumps running with no reactor water cleanup (RWCU) letdown. This condition has the potential to exceed the P-T curve limits.

C.1.3 Feedwater Injection and/or High Pressure Core Spray Injection

An uncontrolled injection of feedwater or high pressure core spray (HPCS) during shutdown could potentially result in a water solid condition in the reactor. The staff notes that HPCS is applicable only to BWR 5 and 6. The HPCS pump is capable of delivering at least 97.8 L/s (1550 gpm) at 7.9 MPa (1147 psig) reactor pressure, 385.4 L/s (6110 gpm) at 1.4 MPa (200 psig), and a maximum of 492 L/s (7800 gpm) at run-out conditions. HPCS normally is aligned to the condensate storage tank (CST) for suction. Depending on the temperature and pressure at the time of the injection, normally 48.8°C to 93.3°C (120°F to 200°F) and 0 MPa (0 psig), an inadvertent repressurization of the RPV could occur with the potential to exceed the P-T curve limits. The staff notes that HPCS injection during shutdown is usually avoided due to precautionary tagouts during maintenance of the systems.

The rest of the BWR fleet have a high pressure core injection (HPCI) System. HPCI is turbine driven and is not designed to operate below 689.5 kPa (100 psig). Since the reactor is subcooled at atmospheric pressure during shutdown, inadvertent actuation of HPCI is generally not possible. However, the staff did identify one LER which describes an inadvertent HPCI injection during cold shutdown. This LER is discussed in Section 5.2.

C.1.4 Low Pressure Core Injection and/or Core Spray Injection

A more common event during shutdown is a LPCI or core spray (CS) injection. Injections of this type can also repressurize the RPV while maintaining reactor vessel temperature. If the injection is for a long duration, the potential exists that the injection may cause a reactor temperature decrease. Both LPCI and core spray take suction from the suppression pool as the primary source of water. An alternate alignment could be the CST. The suppression pool water temperature ranges from 18.3°C to 35°C (65°F to 95°F), depending on the plant-specific TSs and the time of the year. In general, each LPCI pump (RHR pump) is designed to produce a flow rate of 458 L/s (7260 gpm) at 1.2 MPa (172 psig) and a maximum flow rate of 537.5 L/s (8520 gpm) for run-out conditions. For this pump, shutoff head is less than 2.3 MPa (329 psig). For some facilities, core spray is capable of delivering 394.3 L/s (6250 gpm) at 2.3 MPa (329 psig) and 482.9 L/s (7655 gpm) at run-out.

C.1.5 Unvented Condition During Shutdown

An unvented reactor vessel during shutdown with reactor temperature below a certain value is a violation of some plants' TSs. In this condition, any addition of water to the vessel, from any source, without letdown, has the potential to repressurize the vessel. However, this generally is a violation of plant TSs and, therefore, it is assumed that the licensees make every effort to avoid violation of their TSs.

C.2 Discussion of staff LER and Event Notification Review

The staff performed a sampling review of LERs and ENs. The LER search involved reviewing potential BWR overcooling or overpressurization events since 1980. A total of 229 LERs met the initial search criteria. The EN search consisted of the same review of notifications since 1985. Approximately 81 ENs were identified which met the initial search criteria. Of the 310 events identified, none of which were overlapping, 35 events were singled out as potential precursors to cold overpressure events at U.S. BWRs. These events are identified in Table 5-1. In addition, the staff is aware of a cold overpressure event which occurred at foreign BWR. The following paragraphs highlight the circumstances of some of the more significant events and what factors prevented a complete overpressurization and potential failure of the vessel.

C.2.1 Cold Overpressure Event at a Foreign Reactor

A cold overpressure event occurred at a foreign reactor during shutdown which appears to be more limiting than the leak test condition. The foreign reactor had nearly completed its sixth refueling outage with the reactor vessel head tensioned and RHR-A running in shutdown cooling mode. At the beginning of the

event, reactor water level was approximately 15 cm (5.9 inches) below the vessel flange and the main steam line isolation valves were in the shutoff position. In addition, the SRV, the vessel head vent pipe, and RWCU were administratively blocked. A weld overlay on a recirculation loop was in progress and required the "B" recirculation pump to be aligned to enhance heat removal capability. The CRD pump was started to establish sufficient seal injection flow for the recirculation pump. CRD flow was approximately 1.9 L/s (30.1 gpm).

As reactor water level gradually increased, operators aligned a RHR pump and related valves to letdown the excess water. Later on that day, maintenance personnel were informed of inaccurate level indication due to reference leg leakage. Almost concurrently, a reactor operator shift turnover occurred, shift 2 to shift 3. Records showed that the information regarding the inaccurate level indication was passed to shift 3. At shift turnover, the level indication was reading 700 cm (275.6 inches) based on the instrument zero line with a system pressure approximately 0 MPa (0 psig). Actual reactor water level was approximately 500 cm (196.8 inches).

This condition continued until reactor system pressure reached 917 kPa (133 psig) and, on high pressure, tripped the RHR pump. The operators were not aware that the RHR pump had tripped. The CRD pump continued to run until the vessel went water-solid with no outflow from the reactor. Pressure continued to increase at a higher rate until reactor system pressure reached 7.9 MPa (1150 psig) and tripped the CRD pump due to low suction head. The "B" recirculation pump also tripped at 7.9 MPa (1150 psig). During this event, reactor coolant temperature was maintained around 26°C to 31°C (79°F to 88°F).

C.2.2 Hydrostatic Tests and Vessel Leakage Tests at U.S. Plants

On March 19, 1989, the Clinton RPV was water solid for the leakage test and the mode switch was set to REFUEL. Concurrently, scram time testing was being performed. At the completion of the scram time testing, the reactor mode switch was repositioned to SHUTDOWN which initiated an expected scram signal. This signal caused the scram inlet and exhaust valves to open which resulted in the control rod drive bypass flow being directed to the water-solid reactor. This in turn caused the reactor pressure to increase at a rate of approximately 48.2 kPa (7 psig) per second and resulted in 4 SRVs lifting per design. Operators attempted to reduce pressure, prior to the SRVs lifting, by using the discharge paths established for the RPV pressure test but were unsuccessful. The maximum pressure recorded during the event was 7.8 MPa (1130 psig). Temperature at the beginning of the event was 71.1°C (160°F). No change in temperature was reported.

On May 11, 1993, a system leakage test of the RPV was being performed at Browns Ferry Unit 2. Pressure was being maintain using a CRD pump and RWCU reject flow control valve. In parallel, a "Functional Line Flow Check Valve" Surveillance Instruction (SI) was being performed. Due to isolations with both tests, operators received indications that reactor pressure was decreasing. The unit operator attempted to maintain pressure in the band prescribed by the ASME Section XI test by lowering RWCU reject flow. The resultant pressure increase resulted in an Anticipated Transient Without Scram (ATWS) scram signal that tripped the reactor recirculation system pumps and

initiated an Alternate Rod Injection (ARI) signal. The highest pressure encountered during the event was 7.7 MPa (1120 psig). Moderator temperature at the beginning of the event was 87.8°C (190°F). No change in temperature was reported.

C.2.3 Feedwater Injection and High Pressure Injection Events at U.S. Plants

On January 25, 1984, Peach Bottom Unit 3 was set up for long path recirculation (feedwater system flush to the condenser) after completion of maintenance on the reactor feed pump bypass valve. The operator failed to close the feedwater inlet valves to the RPV as required in the system procedure. With a condensate pump in service, the operator opened the 5th heater outlet valve and inadvertently injected condensate into the reactor vessel. Reactor water level increased approximately six feet and a minimal pressure increase was noted on the wide range reactor pressure strip chart recorder. This pressure increase was estimated to be less than 69 kPa (10 psig). Minimum reactor vessel temperatures recorded were 42.2°C (108°F) in the A recirculation suction line and 46.1°C (115°F) at the feedwater nozzle. This transient violated the TS limits of 48.9°C (120°F) and 0 kPa (0 psig) as referenced in Paragraph 3.6.A, Figure 3.6.2 of the Peach Bottom Unit 3 TSs.

On July 27, 1990, Nine Mile Point Unit 1 in an extended refuel outage with the core loaded and with the mode switch in the REFUEL position. The HPCI System initiated while the licensee was preparing to perform a HPCI Automatic Initiation Surveillance Test. The feedwater isolation valves were opened by procedure. Due to a worn cam in the feedwater flow control valve positioner, feedwater leaked into the reactor vessel. A high reactor water level, 241.3 cm (95 inches), signal was received which initiated a turbine trip signal. HPCI initiated and injected into the reactor vessel for approximately 3 seconds. At the time of the event, the reactor vessel water temperature was about 65.6°C (150°F).

On May 3, 1995, during the performance of the LaSalle Unit 2 Reactor Vessel Water Level Reference Leg Continuous Backfill Panel 2C11-P002 Operation, instrument maintenance personnel returned a continuous backfill panel to service. The backfill panel provides a low continuous flow of high pressure water to the reactor vessel instrument reference leg piping to ensure that the reference column is filled to its proper height and that any non-condensable gases are flushed from the reference leg piping. During the valving sequence specified in the procedure, a pressure line spike caused the HPCS diesel generator and HPCS pump to initiate. The HPCS pump started and injected into the vessel. Reactor water level rose approximately 20.3 cm (8 inches). If the reactor operator had not secured the HPCS pump, the HPCS system would have stopped injecting due to high level closure of the HPCS injection valve. The addition of suppression pool water did not reduce the reactor water temperature to below the minimum bolt-up temperature, 30°C (86°F), or the minimum temperature used in shutdown margin calculations, 20°C (68°F). Reactor water temperature remained above 48.9°C (120°F) during the event.

C.2.4 Low Pressure Core Injection and/or Core Spray Injection at U.S. Plants

On May 15, 1985, instrument and control personnel were in the process of performing the "Channel Logic Response Time" procedure at Hatch Unit 2. The

procedure mistakenly called for a jumper to be placed between AA3 and AA4 in panel 2H11-P617; proper placement was between AA3 and AA4 in panel 2H11-P627. RHR pump C was running in shutdown cooling mode. The improperly installed jumper initiated a loop "A" LOCA signal which caused RHR pumps "B" and "D" to start in the LPCI mode and inject water from the torus into the RPV. RPV level increased from approximately 91.4 to 254 cm (36 to 100 inches), referenced to instrument zero, as a result of the injection. The RPV pressure and temperature at the time of the event was not reported in the LER.

On June 9, 1988, during performance of the Cooper Surveillance Procedure 6.3.4.3 on the emergency diesel generators, approximately 56,775 liters (15,000 gallons) of suppression pool water were injection into the reactor vessel. The surveillance procedure is performed once per cycle to functionally test the emergency start of each emergency diesel generator, RHR pump and core spray pump, as well as the loading sequence of safety-related equipment on the associated diesel generator. Upon simulating the low reactor vessel water level, the DC powered RHR loop "B" inboard injection valve began to open. When the critical bus was re-energized, the RHR loop "B" outboard injection valve began to open and the RHR loop "B" outboard to suppression pool valve began to close. These valve operations, along with RHR pump "D" starting when the bus was re-energized, created an injection path into the reactor vessel via RHR loop "B." Additionally, the core spray loop "B" outboard injection valve and the core spray loop "B" inboard injection valve began to open. When the core spray pump "B" started ten seconds later, an additional flow path via core spray loop "B" was created. The total injection time for RHR loop "B" was 97 seconds, and 53 seconds for core spray. Approximately 41,635 to 45,420 liters (11,000 to 12,000 gallons) of water was injected by RHR and 11,355 to 15,140 liters (3,000 to 4,000 gallons) of water was injected by core spray to the RPV. Had the injection continued with no operator action, the reactor vessel would have completely filled and pressure would have stabilized at approximately 2.3 MPa (340 psig), shutoff head for core spray. Actual reactor vessel metal temperatures ranged from 76.7°C to 87.8°C (170°F to 190°F) in the flange area, with no vessel temperatures below 51.7°C (125°F).

On July 6, 1994, instrument and control personnel were back filling instrument lines to support excess flow check valve testing at Washington Nuclear Plant Unit 2. Due to a lineup error, the differential pressure sensed by the inservice reactor vessel level detectors was increased creating an invalid low level indication which caused several automatic actions including a low pressure core spray system actuation and injection. The residual heat removal (RHR) system was in a shutdown cooling lineup, and therefore, no LPCI system injection to the RPV occurred via the RHR system. Reactor water level increased approximately 50.8 cm (20 inches) due to the core spray injection. The injection water temperature was 21.1°C (70°F) with the RPV at 54.4°C (130°F) and 730.9 kPa (106 psig).

TABLE C-1 - Sampling Review of LERs and ENs

Plant	Date	Type of Event	P (psig)	T (°F)
Nine Mile Point 1	06/21/84	Hydrostatic test	1068	NP
Peach Bottom 2	05/30/85	Hydrostatic test	1030	NP
Shoreham	06/06/85	Hydrostatic test	1047	NP
Clinton	03/19/89	Hydrostatic test	1130	160
Browns Ferry 2	05/11/93	Hydrostatic test	1120	190
Peach Bottom 3	01/25/84	Feedwater inj.	10	108
Nine Mile Point 1	07/27/90	HPCI inj.	NP	150
LaSalle 2	05/03/95	HPCS inj.	0	120
Limerick 1	03/26/85	LPCI inj.	NP	NP
Hatch 2	05/15/85	LPCI inj.	NP	NP
Shoreham	05/05/87	LPCI inj.	125	160
Nine Mile Point 2	02/02/89	LPCI inj.	0	Amb.
Brunswick 2	10/21/86	Core spray inj.	NP	NP
FitzPatrick	04/07/87	Core spray inj.	NP	NP
Cooper	05/26/88	Core spray inj.	NP	NP
Vermont Yankee	04/12/92	Core spray inj.	NP	NP
WNP2	07/06/94	Core spray inj.	106	130
Brunswick 2	05/30/86	Core spray & LPCI inj.	NP	150
Dresden 2	02/05/89	Core spray & LPCI inj.	NP	> 100
Hope Creek	03/08/92	Core spray & LPCI inj.	0	NP
Cooper	06/09/88	RHR & Core spray inj.	< 340	170/190
Hatch 2	11/14/92	RHR & Core spray inj.	NP	NP
Peach Bottom 3	09/28/81	Unvented condition	78	<120
Dresden 3	07/11/84	Unvented condition	NP	130/140
Dresden 2	10/08/85	Unvented condition	NP	133
Dresden 2	05/24/86	Unvented condition	NP	130/140
Peach Bottom 3	10/26/89	Unvented condition	NP	155
Nine Mile Point 1	08/03/86	Pot. HPCI inj.	NP	NP

TABLE C-1 - Sampling Review of LERs and ENs

Plant	Date	Type of Event	P (psig)	T (°F)
Perry	07/11/86	Pot. HPCS inj.	NP	NP
WNP2	06/18/89	Pot. HPCS inj.	122	NP
WNP2	06/18/89	Pot. HPCS inj.	NP	NP
WNP2	06/18/89	Pot. HPCS inj.	NP	NP
WNP2	07/04/94	Pot. HPCS inj.	NP	NP
Nine Mile Point 2	10/08/88	HPCS init. - no inj.	35	104
Nine Mile Point 2	02/19/89	HPCS init. - no inj.	2	129

Pot. = Potential inj. = injection init. = initiated