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	3

RESPONSE TYPE — FINAL √ PARTIAL

REQUESTER: Paul Gunter

DATE: JUL 2 0 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.
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- [ ] 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 Carol Ann Reed

APPENDIX E
RECORDS BEING RELEASED IN THEIR ENTIRETY

NO.	DATE	DESCRIPTION/(PAGE COUNT)
14.	04/22/99	Ltr to A. Kenny, EPRI from J. Wilson, NRC Re: Request for withholding information from public disclosure TR-110172 Technical justification for the extension of the interval between inspections of weld overlay repairs February 1999 (4 pages)
15.	Undated	Memorandum to C Grimes from W Bateman, Subject: Acceptance for Referencing of Report, "BWR Vessel and Internals project, BWR Lower Plenum Inspection and Flaw Evaluation Guidelines (BWRVIP-47)" for Compliance with the License Renewal Rule (10 CFR Part 54) (12 pages)
16.	8/10/99	Memorandum to C Grimes from W Bateman, Subject: Safety Evaluation of the BWRVIP Vessel and Internals Project, EPRI Report TR-107286, for Compliance with License Renewal Requirements (10 pages)

APPENDIX E
RECORDS BEING RELEASED IN THEIR ENTIRETY

NO.	DATE	DESCRIPTION/(PAGE COUNT)
1.	12/26/96	Press release from Nuclear Power Safety Administration Division, ANRE/MITI (11 pages)
2.	03/25/97	Ltr to C. Carpenter, NRC from W. Bilanin, BWRVIP Re: Transmittal of EPRI Interim Report "Underwater Wet Welding for the Repair of Reactor Pressure Vessel Internals," NP7481, January 23, 1992 (33 pages)
3.	05/12/97	Public meeting between NRC and Boiling Water Reactor Vessel and Internals project (33 pages)
4.	05/12/97	View graphs on NRC Staff's position on augmented Examination requirements for Boiling Water Reactor Pressure Vessels pursuant to 10 CFR 50.55 (11 pages)
5.	05/21/97	Ltr to G. Carpenter, NRC from S. Tang, Structural Integrity Assoc. Inc. Re: Transmittal of VIPER source code (1 page)
6.	06/03/97	Ltr to C. Carpenter, NRC from C. Terry, Niagara Mohawk Power Co. Re: BWRVIP Vessel Inspection Program Evaluation for Reliability (VIPER) Software and Users Manual (4 pages)
7.	08/05/97	View graphs on Staff evaluation of Industry proposal to eliminate inspection of BWR RPV circumferential welds (9 pages)

APPENDIX E
RECORDS BEING RELEASED IN THEIR ENTIRETY

NO.	DATE	DESCRIPTION/(PAGE COUNT)
8.	08/07/97	View graphs on the Discussion of Independent Assessment of BWRVIP-05 Conditional Failure Porbability (19 pages)
9.	10/10/97	Ltr to C. Terry, Niagara Mohawk Power Co. from G. Lainas, NRC Re: Request for additional information regarding BWRVIP-05 (5 pages)
10.	12/14/97	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, BWR Standby Liquid Control System/Core Plate Inspection and Flaw Evaluation Guidelines BWRVIP-27" (2 pages)
11.	03/27/98	Ltr to C. Terry, Niagara Mohawk Power Co. from G. Lainas, NRC Re: Closeout for BWR Vessel and Internals Project (BWRVIP), Update of Bounding Assessment of BWR/2-6 Reactor Pressure Vessel Integrity issues BWRVIP-46 (3 pages)
12.	06/17/98	Ltr to L. Lund, G. Carpenter, NRC, L. Willertz, BWRVIP and R. Dyle, Inservice Engineering from B. McLeod, BWRVIP Re: Minutes of meeting with NRC regarding weldability of irradiated materials (4 pages)
13.	03/16/99	View graphs of meeting regarding Process to Revise NUREG-0313, Rev. 2 (23 pages)

Press Release Information	Nuclear Power Safety Administration Division, ANRE/MITI
Cause and countermeasure of troubles discovered during the periodic inspection on Unit-1 of the Fukushima Daiichi Nuclear Power Station, Tokyo Electric Power Company	

December 26, 1996

Unit-1 (BWR, rated power 460MWe) of the Fukushima Daiichi Nuclear Power Station, Tokyo Electric Power Company has been under the periodic inspection since August 18, 1996. After the construction work for replacement of a part of the reactor recirculation system piping had been finished, the check on the conditions around the jet pumps in the reactor pressure vessel was performed. As a result, slight cracks were recognized in the vicinity of welded part of two of 10 jet pump inlet pipes.

Therefore, it was decided to perform the check and investigation.

Besides, no external radiation effect was found.

(This was reported on November 26, 1996.)

As the result of the check, cracks with branches and so on were recognized at a part around each welding part on the concerned piping.

On investigation, the cause of the occurrence of these cracks was supposed as follows: the intergranular stress corrosion cracking occurred at the grain boundary under the circumstances of the water including dissolved oxygen gas at the high temperature during the operation, due to the residual tension by welding.

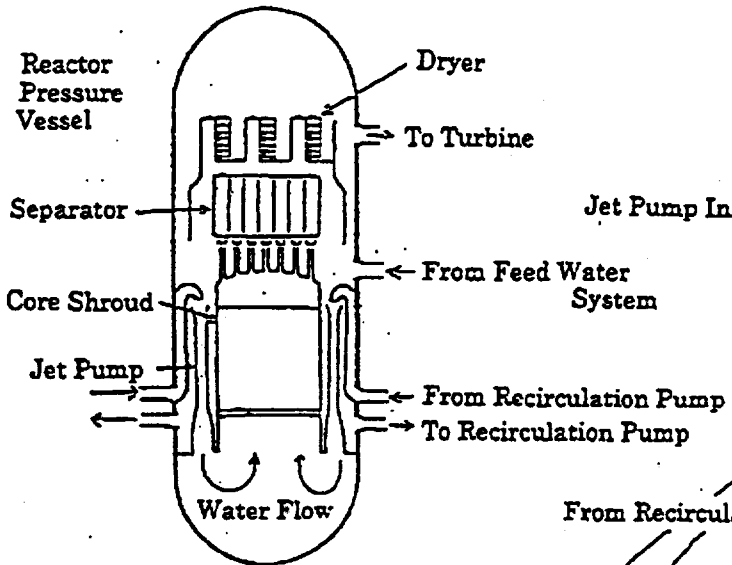
Therefore, it was decided to prevent the growth of cracks with the method, pouring hydrogen gas, introduced from this periodic inspection. It was also decided to repair by setting clamps at the concerned part.

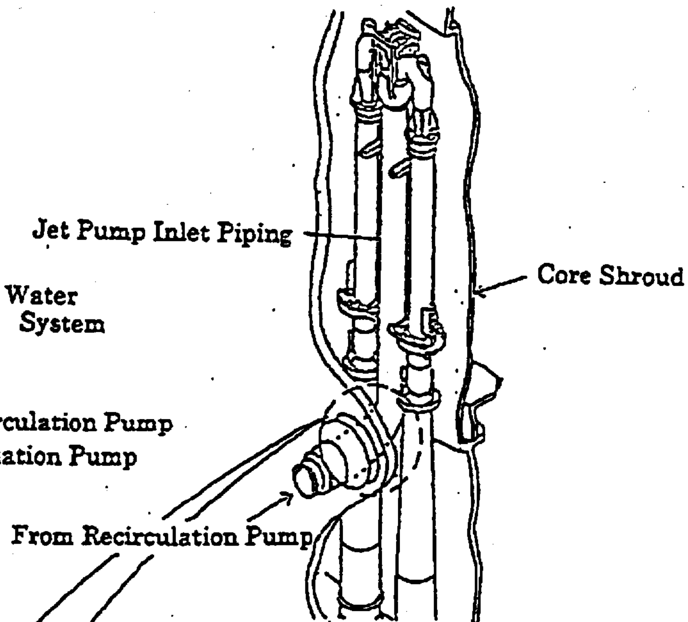
－Results of Tentative Evaluation of INES－

Criterion 1	Criterion 2	Criterion 3	Level
－	－	0－	0－

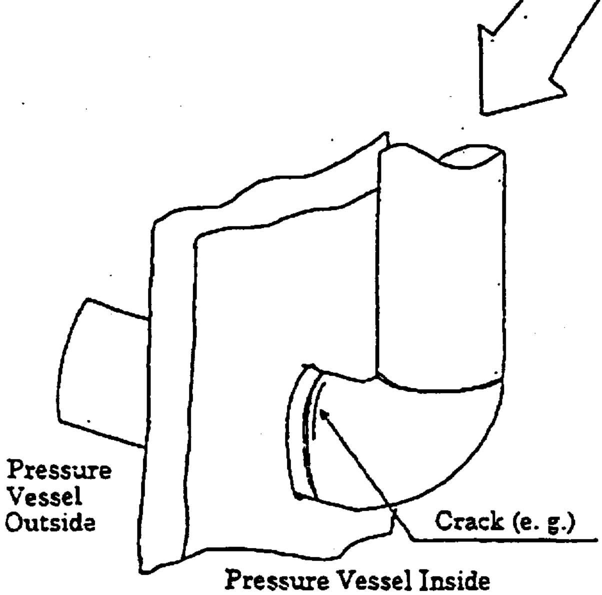
E11

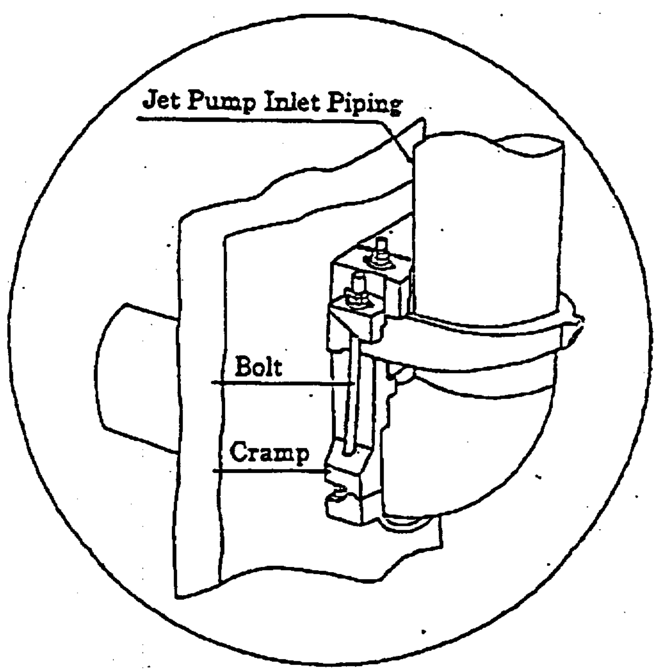
Outline of Reactor Pressure Vessel Inside





Outline of Jet Pump





Enlargement of Jet Pump Inlet Piping

Outline of Cramp Method

Cracks of Inlet Piping of Jet Pumps
at Unit-1 of the Fukushima Daiichi Nuclear Power Station

1. Outline of the Event

Unit-1 has been under the 19th periodic inspection since August 18th. After the work for replacement of a part of the reactor recirculation system piping has been finished, the check on the conditions around the jet pumps in the reactor pressure vessel was performed.

As a result, cracks were recognized in the vicinity of welded part of two of 10 jet pump inlet pipes.

2. Results of Inspection and Survey

(1) Inspection Result

As the results of visual inspection using an underwater camera at welded partof 10 jet pump inlet pipes and its vicinity, one outside peripheral direction crack (about 83 mm length, about 0.2mm width) in the vicinity of jet pump inlet pipe (N2D) and one outside peripheral direction crack in the vicinity of jet pump inlet pipe (N2E) were recognized.

As the results of the detailed inspection of the characteristics of the cracks, the cracks are not linear but are with branches, crimps and irregularity,and have shown the feature of stress corrosion cracking.

As the results of the observation of the relation of the location of the cracks with the welding line of the parts, the cracks are recognized at about 6 mm and 4 mm maximum from the end of the welding bead respectively.

(2) Survey Result

a) As the survey result of the data of water chemistry of the unit, it was recognized to be under the water chemistry conditions able to generate the intergranular stress corrosion cracking.

b) As the result of the welding mock-up test result simulating the plant in the range of grain sensitization due to welding heat, the grain sensitization was recognized around the welding part. It was recognized that the range of the sensitization corresponds to the result of the identified location of the cracks of the plant.

c) As the result of the welding mock-up test, residual tensile stress was seen around the outside of the welding part of the jet pump inlet pipes.

3. Estimated Cause

Considering the above investigation results, the cracks in the vicinity of the welding part of the jet pump inlet pipes are estimated to be the intergranular stress corrosion cracking generated due to superimposed three factors of circumstances, materials and stress.

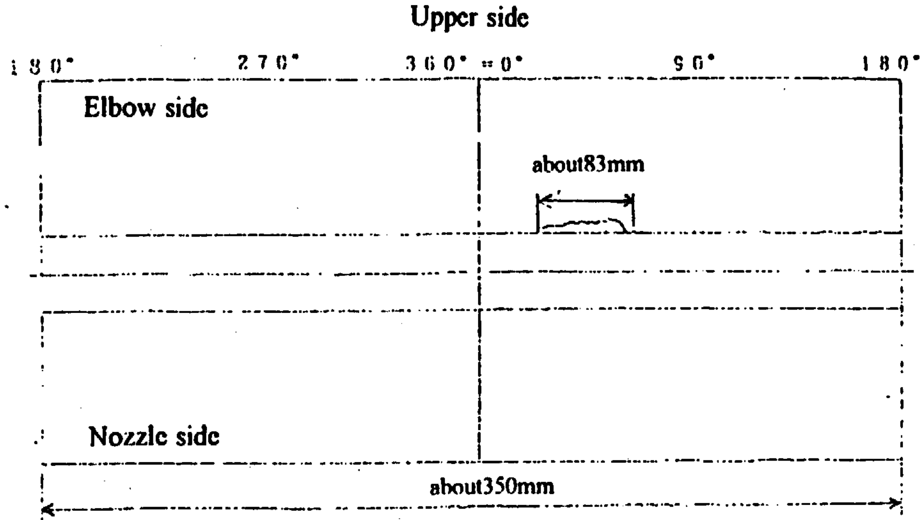
4. Countermeasures

According to the stress evaluation result using fracture mechanics under the conservative assumption that the cracks penetrate the piping wall, it was recognized that the pipe will not break. The progress of the cracks will be suppressed by hydrogen injection system installed during this periodic inspection. Therefore, it was recognized that there is no problem to continue the operation of the plant.

However, it was decided to set clamps for keeping enough strength even in the case that the cracks penetrate all periphery.

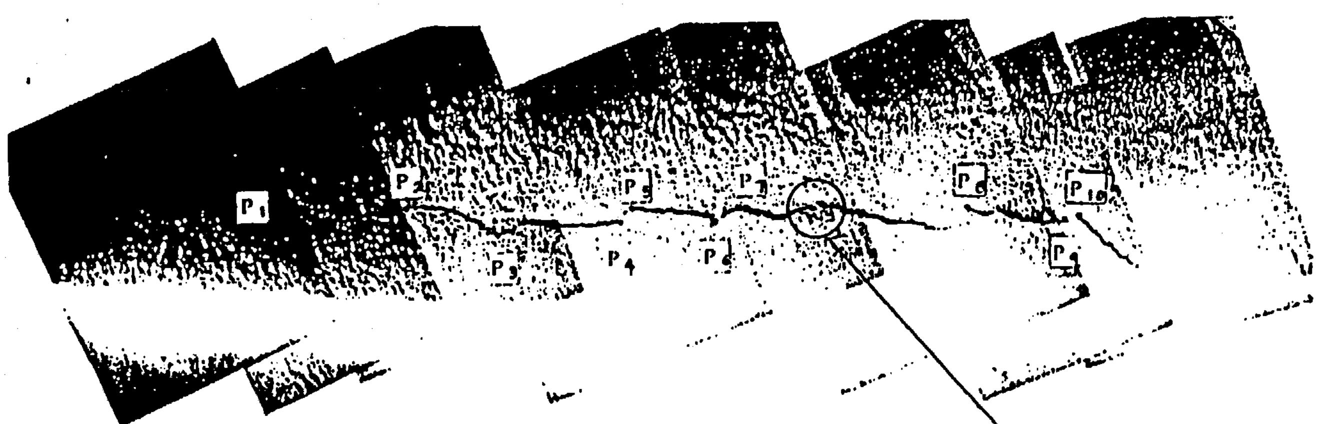
THE MAIN CAUSE EVALUATION TABLE FOR SLIGHT CRACKS IN VICINITY OF WELDED PART OF JET PUMP INLET PIPING

(Phenomenon)	(Cause)		Examination method	Examination result
The cracks in the vicinity of welded part of Jet Pump Inlet Piping	Material flaw		●Confirmation of chemical component, mechanical property and heat treatment condition by Mill sheet.	Satisfied with standard chemical component and mechanical property. And heat treatment have been performed by standard method.
The cracks in the vicinity of welded part of Jet Pump Inlet Piping	Welded flaw	Weld cracking Insufficient weld Blow hole or Wrap slug	●Confirmation of flaw condition by visual inspection. ●Confirmation of weld condition and inspection result after welding by welding record.	Cracks are linear shape, no flaw was found at welding metal. And there is no possibility of welding, because it is confirmed that no problems of welding condition and inspection result according to welding record.
The cracks in the vicinity of welded part of Jet Pump Inlet Piping	Corrosion	General corrosion	●Confirmation of flaw condition by visual inspection.	There is no possibility of general corrosion, because cracks are linear shape, and no general corrosion condition was found at around welding area.
The cracks in the vicinity of welded part of Jet Pump Inlet Piping	Corrosion	Crevice corrosion	●Confirmation of flaw condition and location by visual inspection. ●Confirmation of possibility of crevice on flaw area by drawing and visual inspection.	There is no possibility of crevice corrosion, because cracks are linear shape, no thinning by corrosion and no crevices around area.
The cracks in the vicinity of welded part of Jet Pump Inlet Piping	Corrosion	Pitting	●Confirmation of flaw condition by visual inspection.	There is no pitting around cracking area.
The cracks in the vicinity of welded part of Jet Pump Inlet Piping	Erosion		●Confirmation of flaw condition and location by visual inspection. ●Confirmation of design flow velocity.	There is no possibility of erosion, because no flaw shaped by ditch and bump inside wall is found, and no condition of erosion by flow velocity at around area.
The cracks in the vicinity of welded part of Jet Pump Inlet Piping	Fatigue cracking		●Detail examination of flaw condition by high powered microscope. ●Evaluation of possibility of stress corrosion occurrence and location by stress analysis.	The crack is the fault of fine snaking type which has braches and the stress due to vibration and the under the operating condition are less than the fatigue limit, so there are nopossibility of fatigue cracking.
The cracks in the vicinity of welded part of Jet Pump Inlet Piping	Hydrogen embrittlement		●Detail examination of flaw condition by high powered microscope. ●Confirmation of mechanical property by Mill sheet.	The crack is the fault of fine snaking type which has branches and the mechanical strength of the material is that of the normal austenite stainless steel, so there are no possibility for hydrogen embrittlement.
The cracks in the vicinity of welded part of Jet Pump Inlet Piping	Stress corrosion cracking		●Confirmation of flaw condition and location by visual inspection. ●Examination of water chemistry data. ●Confirmation of chemical component by Mill sheet. ●Detail examination of flaw condition by high powered microscope. ●Confirmation of sensitized condition and residual stress by performing mock-up test simulating material specification and welding condition of actual compornent.	The crack is the fault of fine snaking type which has branches. The resolved oxygen concentration of the reactor coolant was 200ppb. The carbon contents of the material was 0.05~0.07%. The location of the crack appearance is corresponding to the sensitized region defined by the mock-up tests. The welding residual stress of the crack points are estimated to be the tensile stress of about 10~15Kg/mm2.



measured point (P i)	Distance from welding part (X i) (mm	
P_1	X_1	5
P_2	X_2	6
P_3	X_3	5
P_4	X_4	4
P_5	X_5	5
P_6	X_6	5
P_7	X_7	6
P_8	X_8	6
P_9	X_9	3
P_{10}	X_{10}	4 5

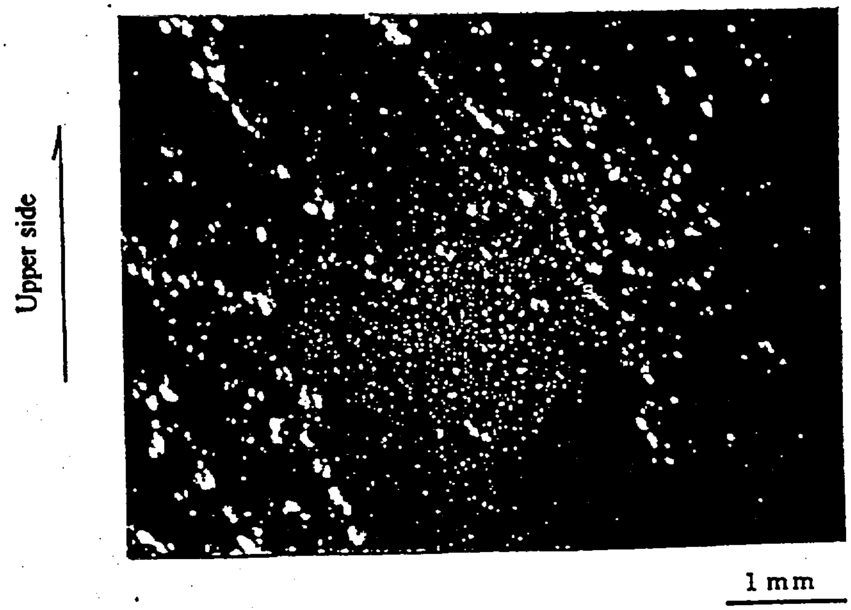
Elbow side

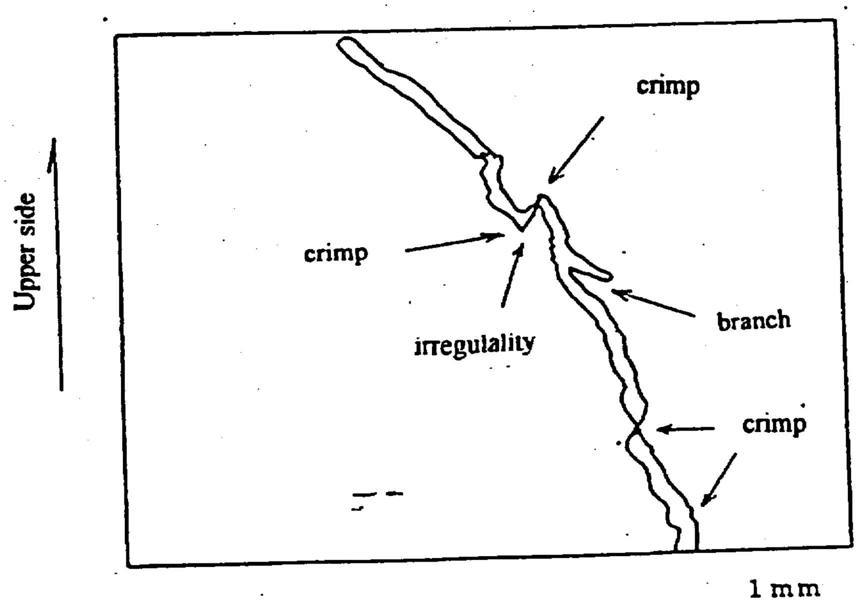


Detail investigation results

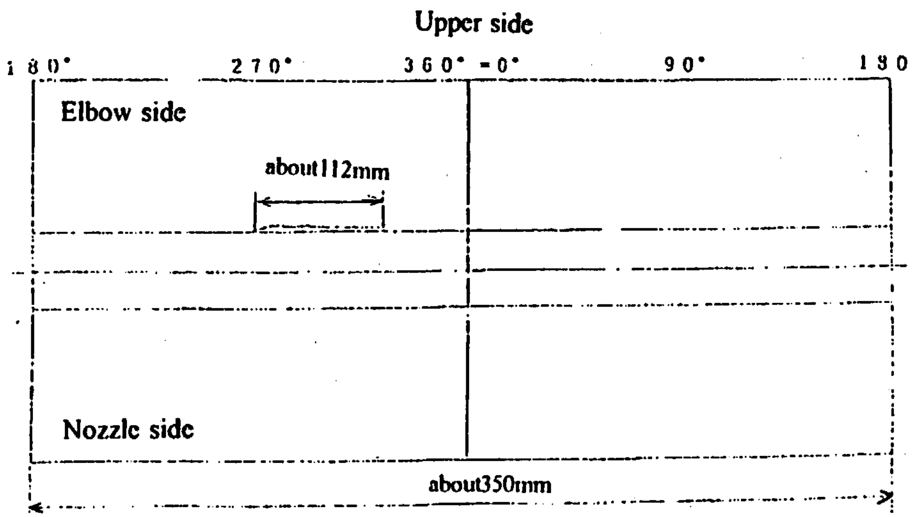
Nozzle side

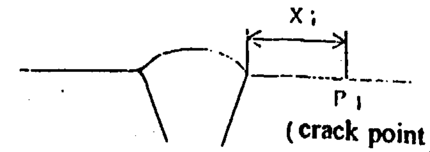
Investigation results of the concerned Jet Pump Inlet Piping part D



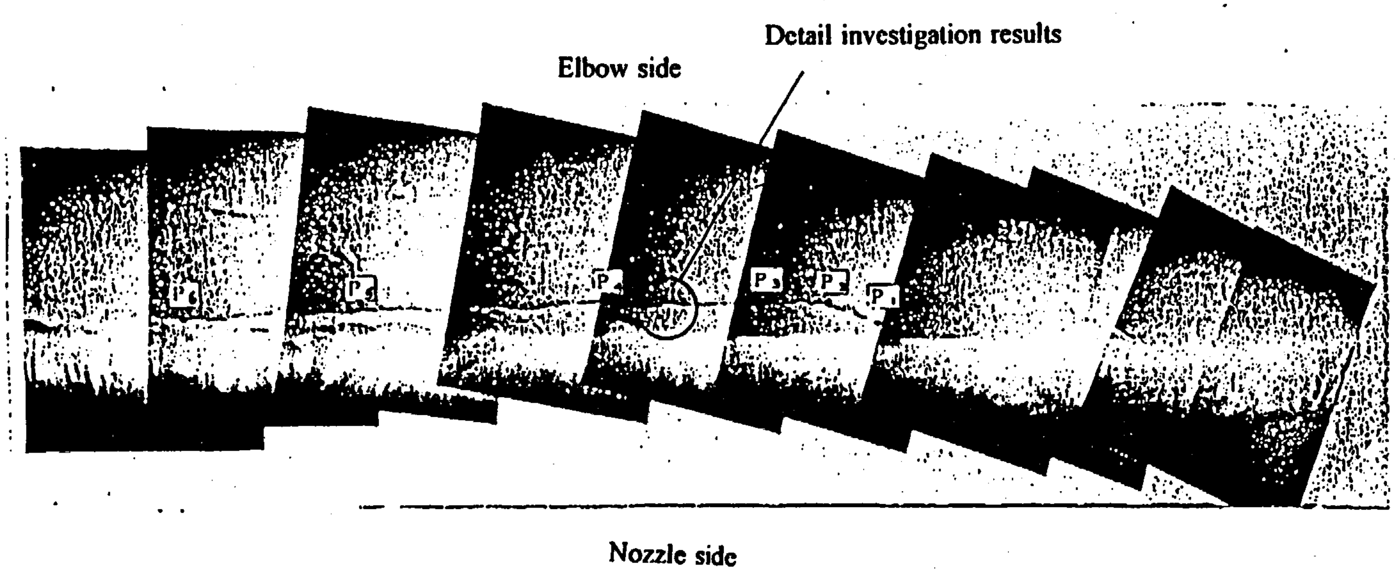


Detail investigation results of Jet Pump Inlet Piping Part D

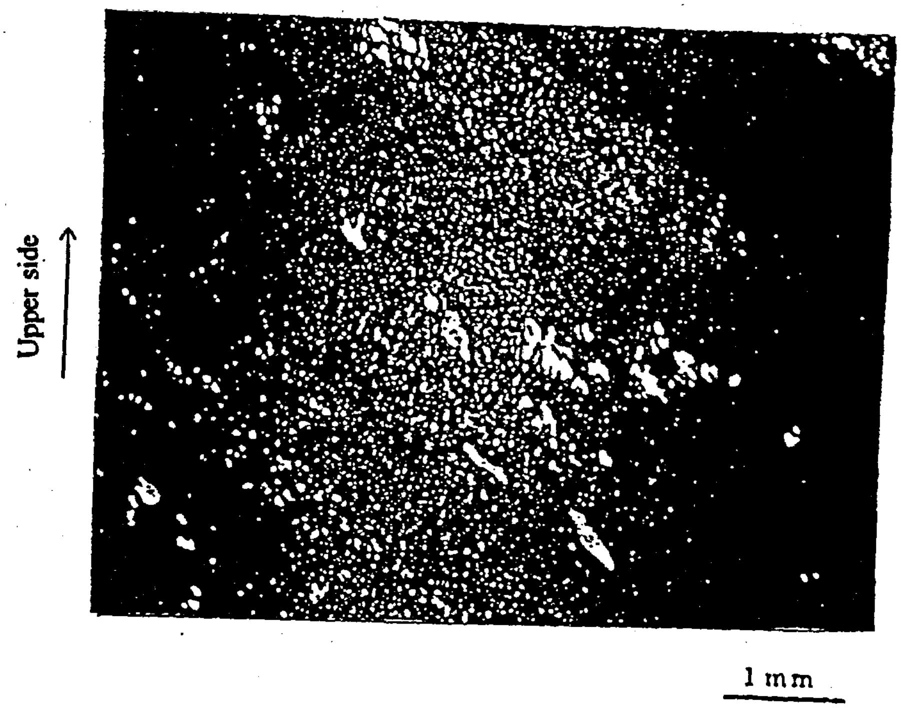


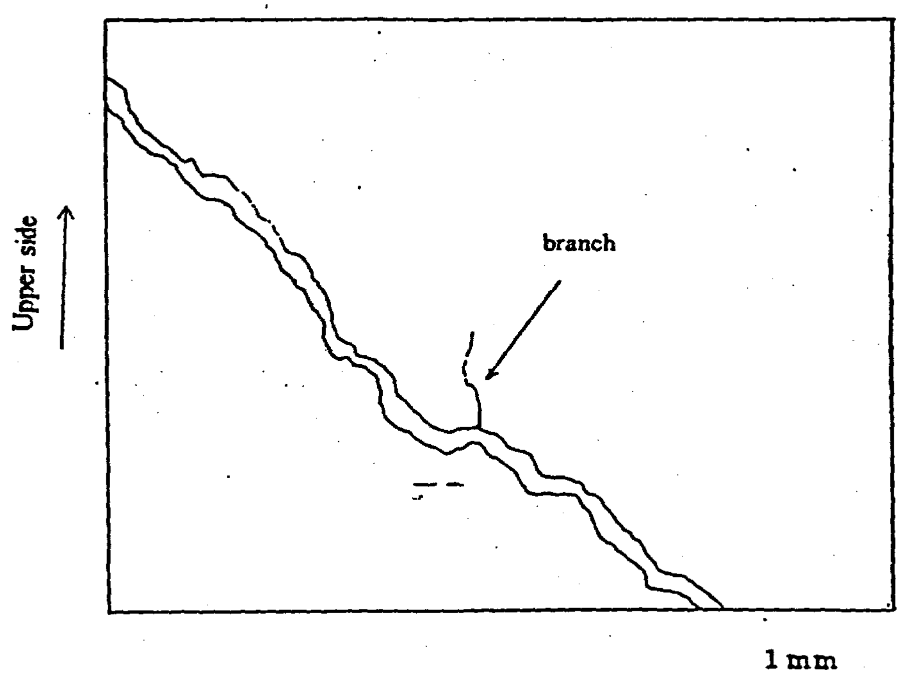


measured point (P i)	Distance from welding part (X i) (mm)	
P_1	X_1	5
P_2	X_2	4
P_3	X_3	6
P_4	X_4	25
P_5	X_5	3
P_6	X_6	15



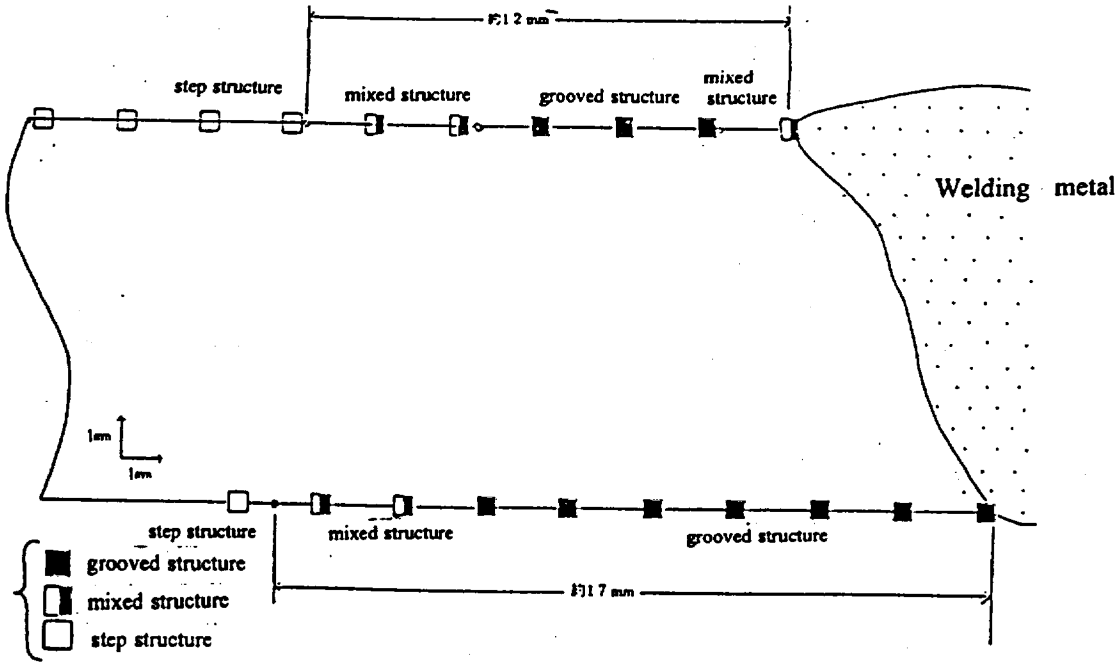
Detail investigation results of Jet Pump Inlet Piping Part E



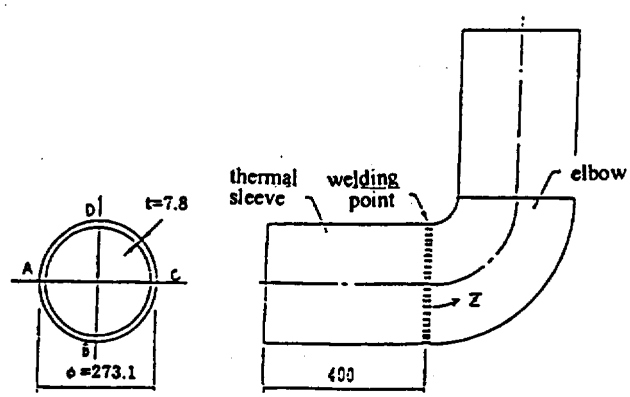


Detail investigation results of Jet Pump Inlet Piping Part E

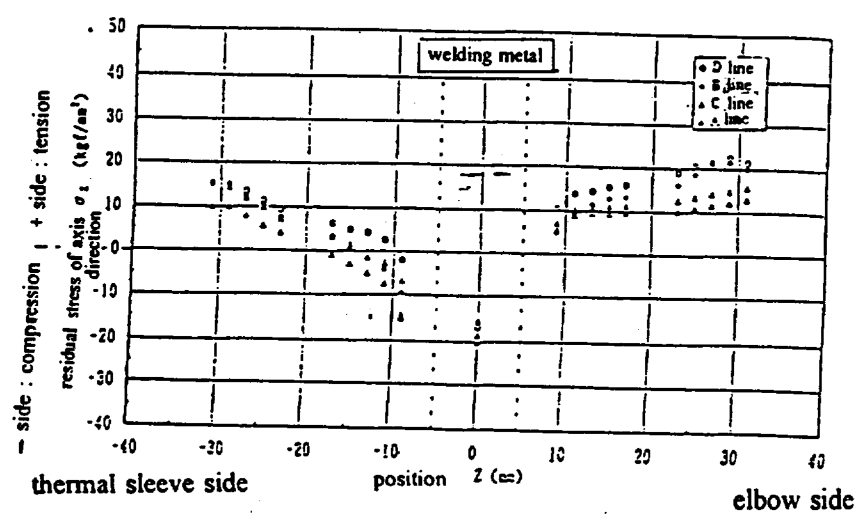
Cross sectional sensitization map of welding joint



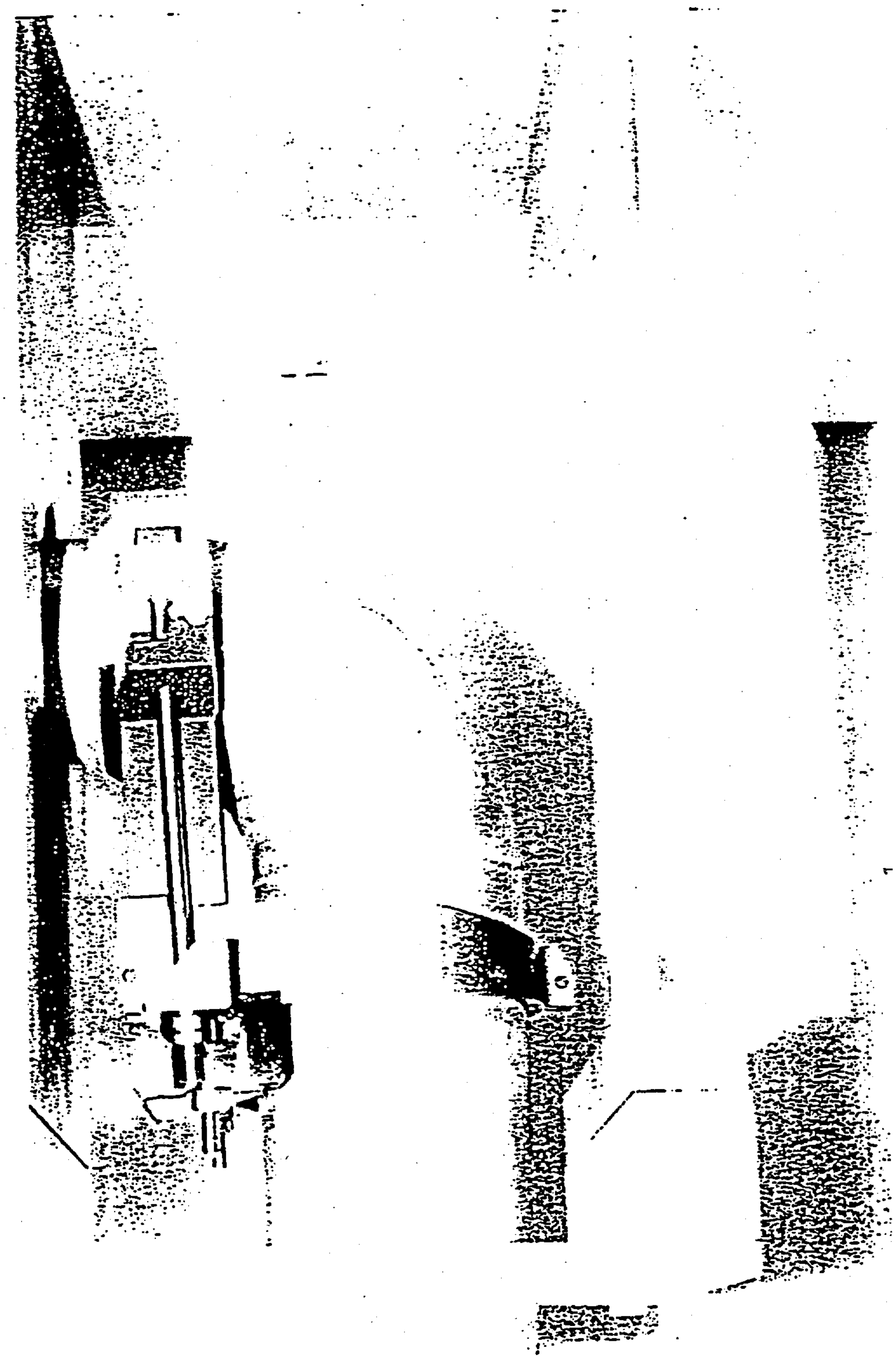
Investigation results of residual stress distribution



Welding point of testing straigt pipe—elbow



Outer surfase stress distribution of axis direction



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97-309

BWRVIP

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

March 25, 1997

Document Control Desk
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11555 Rockville Pike
Rockville, MD 20852-2738

Attention: Mr. C.E. Carpenter

SUBJECT: Transmittal of EPRI Interim Report "Underwater Wet Welding for the Repair of Reactor Pressure Vessel Internals," NP7481, January 23, 1992

Dear Gene,

Enclosed are 10 copies of the EPRI Interim Report "Underwater Wet Welding for the Repair of Reactor Pressure Vessel Internals," NP7481, January 23, 1992. The report is being provided to you for the purpose of supporting generic regulatory improvements related to repair of BWR internal components. The report is not proprietary.

If you have any questions, please call me at 415/855-2340.

Sincerely,

Warren Bilanin

Warren Bilanin
BWRVIP Program Manager

DFX? 1/1 Add Tamir Misra - Paper Copy

Underwater Wet Welding for the Repair of Reactor Pressure Vessel Internals

Interim Report

NP 7481

January 23, 1992

Prepared by

EPRI NDE Center
Repair Applications Division
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Charlotte, NC 28262

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Charlotte, North Carolina
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ABSTRACT

In 1989 the EPRI NDE Center initiated a study of welding repair methods for reactor internals in preparation for potential flaws detected during routine inspections. Research facilities were constructed, studies were made of existing underwater welding capabilities, and performance of underwater wet welding was initiated with the shielded metal arc welding (SMAW) process and the automatic flux-cored arc welding (FCAW) process for remote applications.

Underwater wet welding is not new to the nuclear power industry. In fact, wet welded repairs have been performed on steam dryers, feedwater sparger nozzles, manway covers, and jet pump hold-down beams. All these welds have been performed manually by diver/welders utilizing the SMAW process.

Underwater wet welding studies currently underway at the EPRI NDE Center are aimed at austenitic stainless steels. The objectives of these studies include: understanding the effects of water and pressure on the weld bead; developing waterproof coatings for SMAW electrodes and developing new FCAW welding wires; and eliminating or substantially reducing the problems of porosity, slag entrapment, lack of fusion, and lack of penetration typically associated with underwater wet welding.

Research performed to date with the SMAW process welding a Type 304 stainless steel (ASME SA-182) substrate with a ASME SFA 5.22 E308L filler material has yielded results that are superior to those welds currently acceptable per ANSI/AWS Specification D3.6, Type O. Visual and mechanical testing of multipass groove welds have met the requirements of ASME Section IX for welds performed in the dry. Similar results have also been achieved with the FCAW process utilizing a manipulator to carry the welding torch at a depth of 20 ft.

ACKNOWLEDGMENTS

This report represents the combined efforts of EPRI, vendors, and utilities. The authors would like to thank EPRI for sponsoring the program, Northern States Power for providing water chemistry testing and weld qualification testing, Kemppi Inc. for providing welding equipment and timely support: Airco for welding equipment; Mr. Roger Swain of Welders Supply of Charlotte, N.C. for insight and support; and Dr. Richard Smith of the EPRI NDE Center for his assistance.

In addition, we want to thank all of the NDE Center Repair Applications staff members who were instrumental in the success of the project, especially Darryl Baucom, Steven Gore, and David MacDonald.

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Section 1

BACKGROUND

The Electric Power Research Institute (EPRI), as a result of member utility concerns, initiated a program which deals with inspection and repair of reactor pressure vessel internal components (Figure 1-1). The reactor internals are fabricated primarily of Type 304 stainless steel, a material susceptible to intergranular stress corrosion cracking (IGSCC). The initial work task for this program encompassed a study of the accessibility of all ASME Section XI welds of the General Electric Company boiling water reactor (BWR) models, a survey of the latest inspection technology and systems currently available, and a survey of the latest repair technologies currently available (2). The second task of the EPRI program was to develop full-size mock-ups to evaluate inspection and repair techniques to facilitate the development of new systems and technologies.

Based on the surveys performed in the early part of this program, plant operating conditions, and the fabrication practices used in constructing BWR pressure vessels, it was evident that there is a high probability of the detection of indications or flaws during inspections of reactor internal components. Based on analysis, many of these indications may require repair for continued plant operation. This report will focus on the development of underwater wet welding technology for the repair of stainless steel reactor internals.

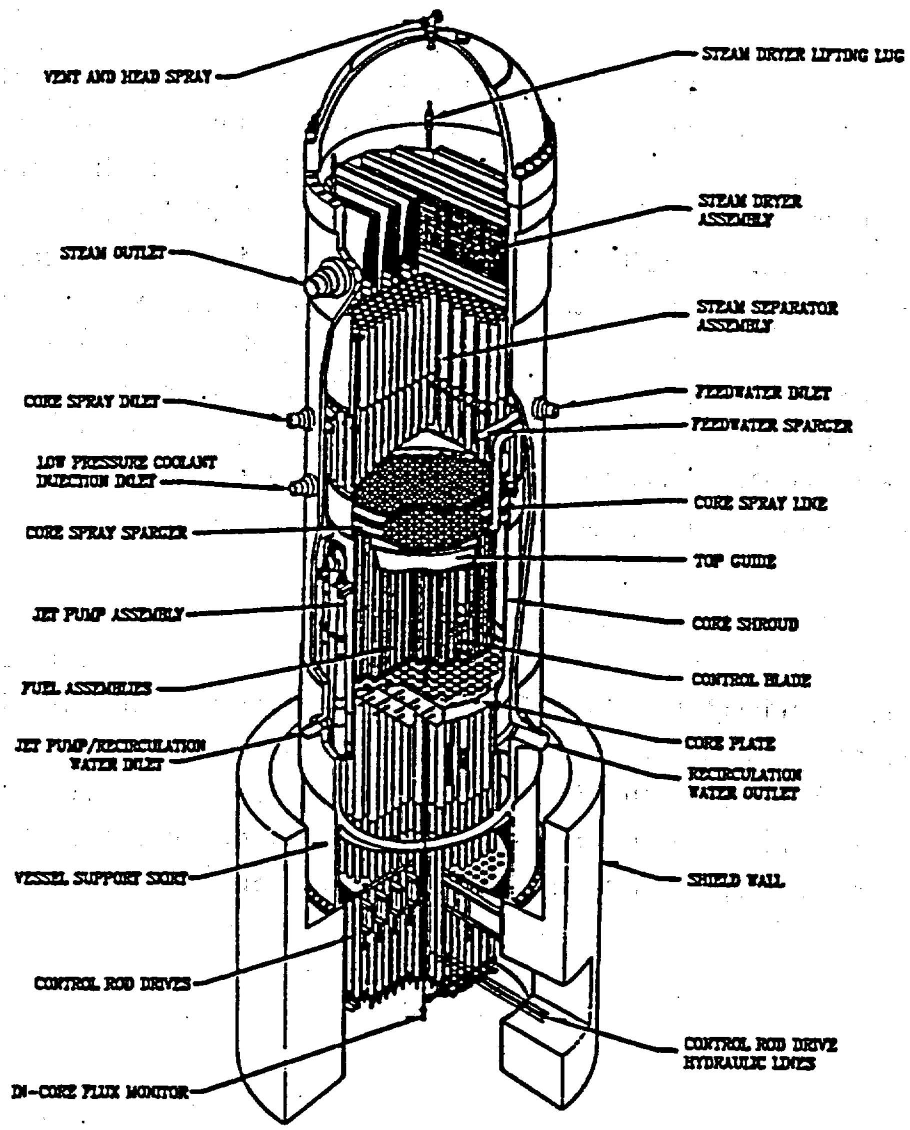

Figure 1-1. Cutaway of a BWR Pressure Vessel (1)

Section 2

INTRODUCTION

Underwater wet welding is not new to the nuclear power industry, in fact there have been numerous wet welding repairs which have been performed on steam dryers, moisture separators, feedwater sparger nozzles, manway covers, and jet pump hold-down beams. Welding underwater is performed not by choice, but as a result of the high radiation fields present in the reactor. While welding underwater does present challenges which require additional preparation, training, and skills, the benefits of reduced radiation exposure to personnel and, in many cases reduced downtime, far outweigh the disadvantages. To date, all in-vessel underwater wet welded repairs have been performed with the shielded metal arc welding process by diver/welders. These repairs have been performed in the upper regions of the reactor pressure vessel where the radiation levels are not very high, with water shielding, and where accessibility for diver/welders is generally unobstructed. While there is accessibility to the components in the upper sections of the reactor vessel, it is quite the opposite in lower portions of the reactor vessel where components generally cannot be removed (Figure 1-1). In this area, a remote underwater wet welding automated process is the only logical choice for performing repairs.

As a result of the limited technology available and the potential needs of the utilities, the following project goals were established for the underwater wet welding program:

- Assemble facilities, equipment, and personnel.
- Improve existing underwater manual welding technology for stainless steel components.
- Adapt an automatic welding process for underwater remote wet welding applications.
- Demonstrate repair welding applications at a depth of 20 ft.

Section 3

ASSEMBLY OF FACILITIES, EQUIPMENT, AND PERSONNEL

Underwater welding test facilities were constructed for the program. The initial test facility was a small 40-gal tank. This tank was used for shielded metal arc welding (SMAW) work in which the 308L stainless steel welding electrode was held underwater but the welder's hands were out of the water. The workpiece was generally six to eight inches below the water surface. Limitations on workpiece size and welder visibility quickly led to a larger and more versatile tank. The second tank was 3 ft by 5 ft and 4 ft in depth with four hand holes and is used for parameter development. Finally, a large 20-ft deep by 20-ft diameter tank (Figure 3-1) was constructed for inspection and repair development with several full scale mock-ups of specific reactor components. Mock-ups included a jet pump annulus section which incorporated a complete jet pump assembly removed from a canceled plant, core support plate with manway opening, core shroud, and several attachment welds to a reactor pressure vessel. A welding platen was also placed in this large tank for welding development testing at 20-ft depths.

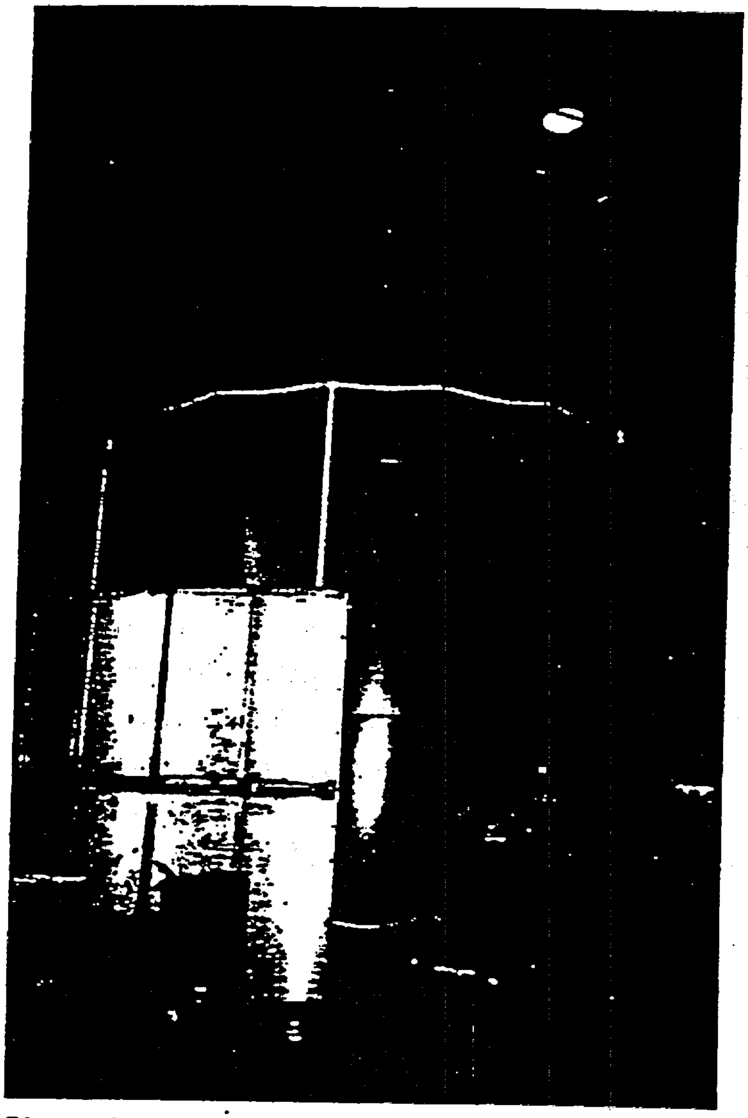

Figure 3-1. 20-ft Deep Underwater Welding Research Facility

Section 4

SMAW PROCESS IMPROVEMENT DEVELOPMENT

A major program objective was to understand and improve existing technology for underwater wet welding, which was largely based on offshore experiences. Manual SMAW is the most common process used for underwater welding applications. However, defects such as porosity and lack-of-fusion, which are typically found with underwater wet welding, can occur due to the fast freezing of the weld puddle. With this in mind the real objective was to develop improved techniques and, if necessary, consumables to address these problems.

Service organizations experienced in nuclear wet welding were contacted and graciously provided information on their current SMAW practices. Several of these organizations also provided their own proprietary welding electrodes as well as sample qualification coupons which were used as comparison samples during welding development activities.

Underwater wet repair welding of nuclear power plant reactor pressure vessel components has been performed, in many cases, in accordance with the requirements of ASME Code Sections IX and XI, and ANSI/AWS D3.6-89, "Specification for Underwater Welding." A review of the 308L stainless steel weld deposits currently acceptable for most types of welds in accordance with ANSI/AWS D3.6 would not be acceptable for ASME Code repairs. To meet future needs, EPRI set out to improve these results and develop weldments that would qualify per ASME Code, Section IX.

Results of SMAW Development Work

Initial weld tests were carried out using various electrodes from industry as well as several developed at the EPRI NDE Center. Testing was also carried out with various welding machines and machine settings. The objective of these tests was to produce a weld deposit that had a smooth and defect-free surface appearance, a slag and porosity-free cross-section, caused minimal water fouling, and exhibited satisfactory mechanical properties.

Evaluations were performed at the 3-ft level and at the 20-ft level, utilizing 308L and 309L stainless steel electrodes on 304 stainless steel base material.

Bead-on-plate and 3/4-in. deep groove welds were made in the flat position. After considerable testing, five electrodes were found to produce very sound welds that met program objectives for appearance and mechanical properties, but only two of these five, both of which were waterproofed by EPRI with a wax-free coating, met the program objectives for their effect on water quality. Figure 4-1 shows shielded metal arc welding being performed at a depth of 20-ft.

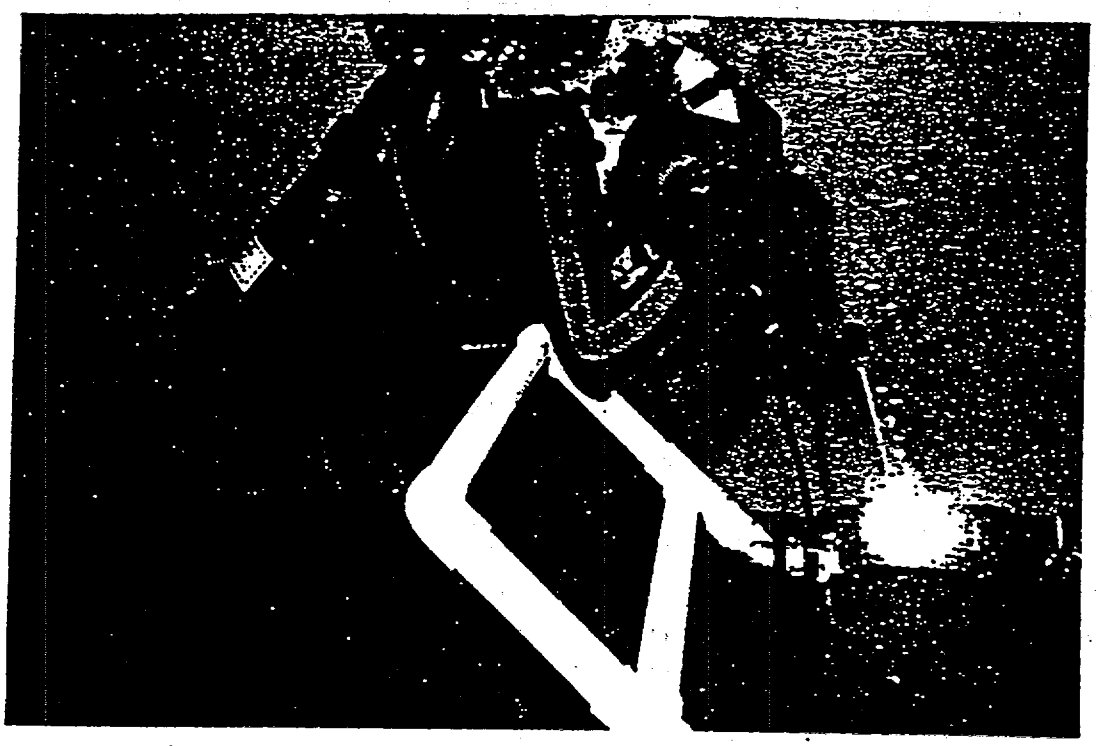

Figure 4-1. Underwater Wet SMAW at a Depth of 20-Ft

While this study did show that there are commercially available welding electrodes that can produce a weld that meets the objectives stated earlier, it required some major changes in the welding process. In order to improve penetration into the base metal and previously deposited layers, pulsed current was utilized. A pulsing, constant current, inverter welding machine (Kemppi Model PSS5000) was provided by Kemppi, Inc. for these tests. By utilizing a pulsed current welding arc it was possible to produce enough heat to penetrate and tie in to the previous layer while maintaining a lower average current and, therefore, a smaller weld puddle. This aided in improved penetration, and reduced the spatter typically found in high current deposits. Without pulsing, slag entrapment and lack of

fusion occurred, particularly in multi-pass welds. This evaluation demonstrated that the existing manual processes can be improved and, in fact, has shown that welding 304 stainless steel materials with the pulsed SMAW process can produce weldments that meet the acceptance criteria of ASME Section IX. Table 4-1 shows the results of the mechanical tests from two 1-in. groove weld tests depositing a commercial grade, Type 312-16 stainless steel electrode (No. 1) and a 309L electrode coated by the EPRI NDE Center (No. 2). The EPRI waterproofed electrode was an Avesta 309L, 1/8-in. diameter with a coating of aluminum paint, and dipped in paraffin.

Table 4-1

TENSILE TEST RESULTS OF SMAW GROOVE WELDS

Electrode	Water Depth	Yield Load (lbs.)	Yield Stress (psi)	Ultimate Load (lbs.)	Ultimate Stress (psi)	Elongation (%)	Reduction of Area
1	1 ft	4,500	46,200	8,640	88,800	15.3	72.0
1A	1 ft	4,100	42,800	8,000	83,600	15.1	74.3
2	1 ft	4,400	45,200	8,400	86,300	23.2	73.0
2A	1 ft	4,500	47,600	8,080	85,400	28.1	73.4

In underwater welding, it is necessary to properly waterproof the flux coating to prevent it from absorbing moisture. Water absorption will cause the flux coating to crack and spall. Additionally, even minor amounts of water absorption will affect the welding quality. Conventional methods of coating electrodes involve the use of a waterproof paint or shellac, and in some commercially available electrodes, a paraffin (wax) coating is also used. Several difficulties were found with these coatings, including contamination of the surrounding water by burnt wax, poor weldability, and poor visibility. Additionally, as welding is performed at deeper depths, the high pressure can cause water to permeate the coatings, causing the flux to absorb moisture.

To eliminate the problems with conventional coating techniques, and provide the welding characteristics required for underwater applications, an alternative method was developed. This is a coating technique that involves submerging the

electrodes in a bath of the waterproofing material (aluminized paint or other suitable material), and placing the submerged electrodes into an autoclave. The autoclave is pressurized using an inert gas to impregnate the electrodes with the waterproof coating. The pressure used is equal or greater than that which will exist at the depth for welding. Pressure impregnation of the electrode flux with the waterproof coating prevents the coating from spalling or separating from the electrode during use, and it provides better coating adhesion.

Further evaluation of the underwater electrodes coated by EPRI was performed at a depth of 20-ft in the horizontal, vertical and overhead positions on both plate and fillet welds. Excellent penetration and bead appearance were obtained with Avesta 308L-PW, 1/8-in. diameter, coated with aluminum paint waterproofing material using a pressure treatment technique, described above. Figure 4-2 shows a sample fillet weld.

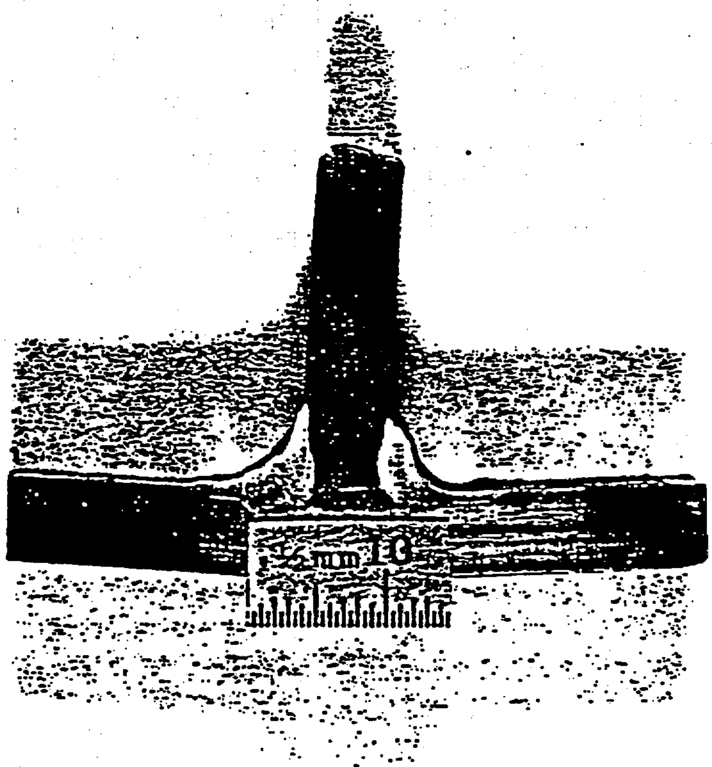

Figure 4-2. SMAW fillet weld performed with Avesta 308L-PW filler material on Type 304 stainless steel. Electrodes were coated with an aluminum enamel.

Test fillet and bead-on-plate welds were performed to evaluate bead shape, weld quality, and penetration using electrodes waterproofed with the pressure treatment coatings. This included:

- Avesta 308L-PW, 1/8 and 3/32-in. diameter
- Thyssen Thermanit JE 308L-15, 1/8-in. diameter
- Arcaloy 308L, 3/32-in. diameter
- Airco 309-16 MR, 1/8-in. diameter
- Thyssen Thermanit Nicro 82, 1/8-in. diameter, nickel-based
- Harris Alloy 1820, 1/8-in. diameter, nickel-based

Different waterproofing materials were used with the pressure treatment technique, including:

- Aluminum Enamel
- Model airplane butyrate dope
- Thompson's Water Seal
- High temperature aluminum paint

The parameters used for these evaluations are listed in Table 4-2. The best results were with the 1/8-in. and 3/32-in. diameter Avesta 308L-PW and the 3/32-in. diameter Arcaloy 308-16, coated with aluminum enamel. This provided sound welds with good penetration, using pulsed current. One key factor in obtaining quality welds with excellent bead appearance and penetration is the use of pulsed current for underwater SMAW welds. Figure 4-3 shows an example of a full-penetration 1-in. thick groove weld, performed with the Avesta 308L-PW electrodes, at a depth of 20-ft.

Initial weld tests have been performed with the nickel based electrodes using similar welding parameters and coatings, as developed for the stainless steel electrodes. The best results to date have been achieved with the Thyssen Thermanit Nicro 82 electrodes, coated with aluminum enamel using the pressure treatment technique. Bead-on-plate welds have been performed with very good penetration and bead appearance. Future testing will include the evaluation of different waterproofing methods, electrodes, multiple layers, and out-of-position welding.

Table 4-2

UNDERWATER SHIELDED METAL ARC PARAMETERS

			Kemppi PSS 5000 Settings						
Electrode	Diameter	Avg. Amps	Volts	Bkgd Amps	Pri. Amps	Pulse Rate (ppm)	Bkgd Time (sec)	Prim Time (sec)	Travel Speed (ipm)
Avesta 308L-PW	1/8 in.	130-150	30-35	122	160	90	.45	.25	6-10
Avesta 308L-PW	3/32 in.	115-125	30-35	105	140	90	.45	.25	6-10
Thyssen Nicro 82	1/8 in.	130-150	33-40	122	160	90	.45	.25	6-10

Note:
Polarity: DCEP
Position: All
Waterproof Coating: Aluminum paint

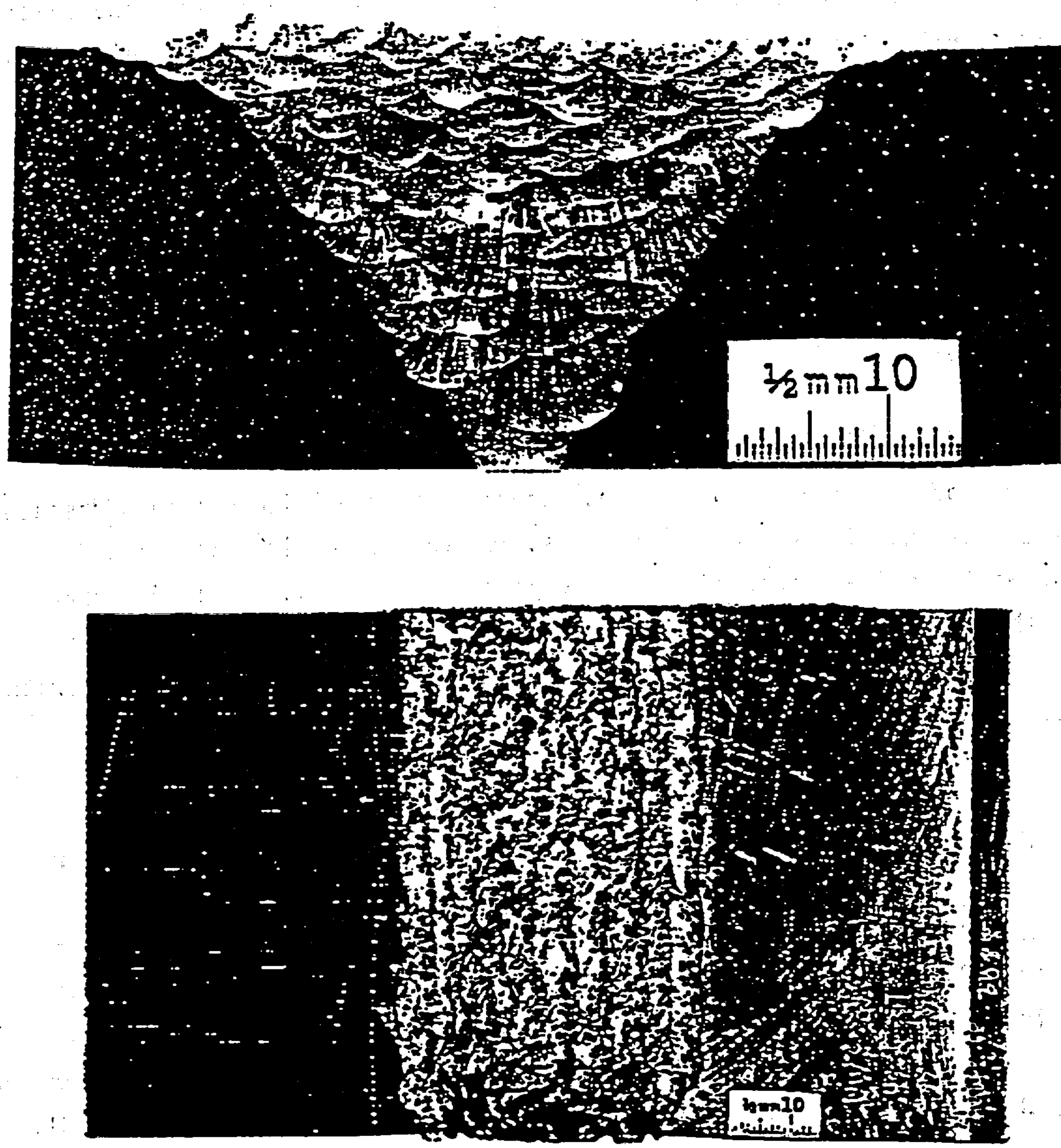

Figure 4-3. Cross-section and surface views of a 1-in. thick Type 304 stainless steel groove weld performed with the SMAW process at a depth of 20 ft. Filler material was Avesta 308L-PW coated with aluminum enamel.

An alternate method was also developed for waterproofing the electrode, using aluminum applied with thermal metal-spray techniques. The aluminum powder provides additional deoxidation for the weld puddle, improving the underwater welding arc characteristics. The coating was applied by the application of a thin layer of aluminum onto the electrode with a thermal metal-spray technique. This metallic layer is dense and provides excellent waterproofing characteristics without the need for wax coatings. Furthermore, the elimination of these coating materials significantly reduces water contamination during welding. This is an important factor for welding applications requiring careful water chemistry control, such as nuclear component repair. Tests were performed with 1/8-in. diameter, Avesta 309L electrodes. These electrodes had good welding characteristics in the flat position. However, they did not perform as well out-of-position as the Avesta 308L-PW, with the pressure treatment coating. Further evaluations will be performed with this technique, using various electrodes and thermal spray parameters.

Flux-Cored Welding for Underwater Applications

Although many of the repair applications for underwater welding can be addressed with the use of manual techniques, more restrictive and/or high radiation locations will require automated welding processes. This would include, for example, the lower two-thirds of the reactor pressure vessel where access is severely restricted and remotely operated equipment would be needed. A literature and vendor survey was performed to determine past uses of underwater remote welding or development programs. Briefly, this review showed that wet underwater welding has been performed mostly with SMAW. The survey revealed that the majority of the automated welding development work had been done with a water-free, dry habitat. In a few cases specialized GMAW techniques utilizing high pressure gas flow or "water curtains" have been employed for automatic wet welding applications (3). Some discussion existed for the use of FCAW, mostly by the Soviets, for low-alloy and mild steel shipbuilding and pipeline applications (4,5). The use of flux-cored austenitic filler materials has also been evaluated by the E. O. Paton Institute and the Dutch Welding Institute for low-alloy steel applications, aimed at reducing the hydrogen content of the weldments (6,7). Since internal components and the reactor pressure vessel cladding are stainless steel, this study focused on the application of ASME SFA 5.22 E308L filler materials.

Although limited, evaluations by researchers of underwater FCAW appeared promising. In addition, it was felt by the NDEC staff that FCAW would be more

versatile than the "gas-shielded only" processes, since it closely resembled SMAW consumables by offering slag protection and could utilize gas shielding, if necessary. Based on the information obtained from the literature and vendor survey, and past experience of EPRI NDE Center Staff, a decision was made to pursue the program objectives with the flux-cored arc welding (FCAW) process. FCAW was chosen due to the beneficial effects of a flux/slag system for protection of the weld puddle and due to the difficulties reported by others in implementing gas shielded processes.

The initial efforts for the development of underwater wet FCAW for stainless steel involved procurement of the appropriate equipment and consumables. Due to the successful results obtained with pulsed current for SMAW, it was decided that evaluations for FCAW should include pulsed current. To support these efforts a pulsed, multi-process power supply was provided by Kemppi, Inc. of Mentor, Ohio, under a long-term loan. This unit is a Model PSS5000, which is an inverter-type power supply and is suitable for GTAW, SMAW, GMAW and FCAW. It provides pulsed DC power, is rated at 500 amps with a 60% duty cycle and can be used as a constant voltage or constant amperage power supply. The unit on loan to the NDEC was equipped with a variable inductance option and a programmable, five-function pendant to allow precise adjustment of the pulse parameters, including primary and background voltage, duration, and wire feed speed. A Model FU30 wire feeder was included which allowed the use of 3/32-in. diameter filler wires (Figure 4-4).

To provide a consistent travel speed, lead angle, and stick out distance for the evaluations, a two-axis manipulator was utilized. This unit is equipped with a waterproofed stepping motor drive, providing precise control of the travel speed. A mounting bracket was fabricated to hold a Tweco TAM-500 machine torch.

Initial testing was performed using a E308L-T3 wire from Weld Mold, Inc., with a product designation of 308L FC-O. This wire is self-shielded and measured 0.045-in. in diameter. The first series of trials involved simple bead-on-plate welds using a semi-automatic torch, immersed six to eight inches underwater. These tests provided several welds that had good bead appearance and were porosity free. Fillet welds were then performed, using parameters developed during the bead-on-plate evaluations. These welds exhibited good bead appearance and were porosity free, however some minor lack-of-fusion (LOF) existed at the weld root. Minor parameter changes improved the penetration of the weld eliminating the LOF condition. Overall, the appearance and quality of these initial fillet welds rivalled some of the best SMAW fillet welds performed at the NDEC and by others.

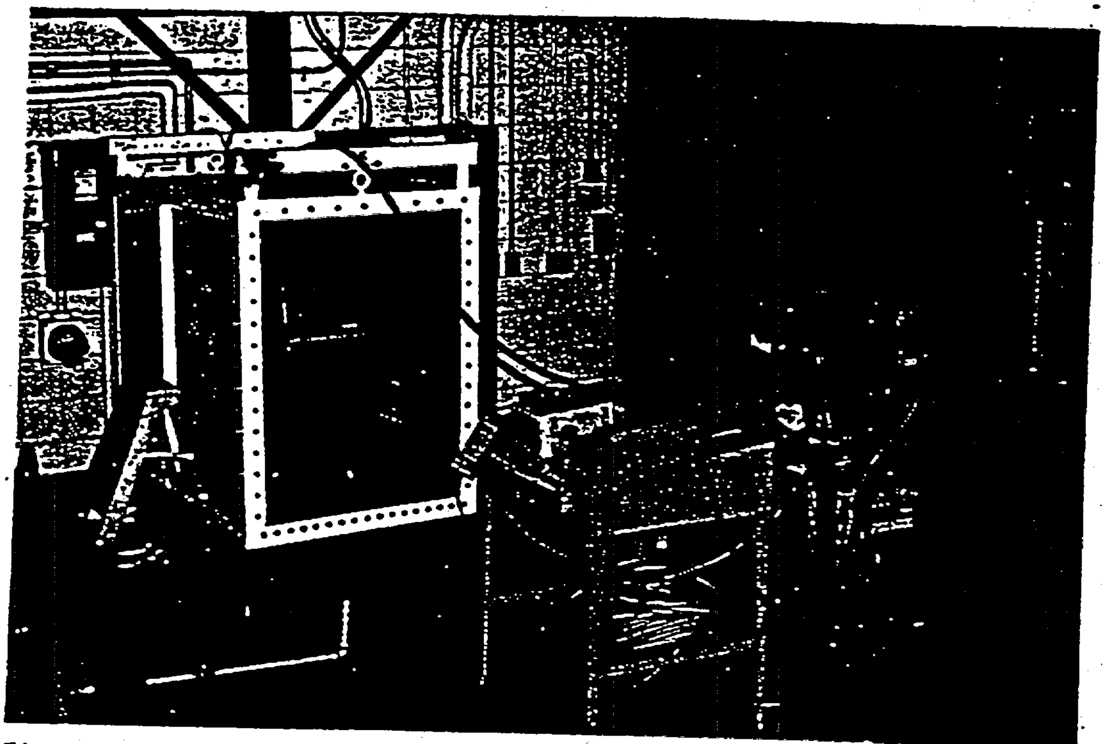

Figure 4-4. Small test tank with Kemppi Model PSS 5000 welding system.

One key area evaluated during the initial development work included the effect of gas shielding. Using the Weld Mold 308L-FC-O (AWS Class E308L-T3) wire, bead-on-plate welds were performed with several shielding gases, including compressed air, argon, helium and CO_2. In each case, the addition of a gas appeared to create turbulence which caused an increase in porosity and poor bead appearance. Several steps were taken to reduce the turbulence, including the use of an extended gas cup, reduced electrode stickout, and varying gas flows. In addition, Stoody AP 308L-T1 wire, designed for use with gas shielding, was also evaluated with similar results. It appears that the gas either did not provide proper shielding or the gas was entrapped in the puddle due to rapid cooling, producing voids in the weld cross-section and an irregular surface appearance. Without the use of specialized torches and equipment it did not seem practical to continue development efforts with the gas shielding.

Further development work was aimed at establishing proper equipment, parameters, and consumable selection for self-shielded flux cored arc welding of stainless steel underwater. A test matrix was developed to evaluate each of these factors. The test welds were performed using the manipulator and machine torch at a depth

of 3-ft. Over 160, 12-in. long bead-on-plate welds and several fillet welds were completed and evaluated as part of the test matrix.

To evaluate the consumables, additional filler materials from several manufacturers were obtained including self-shielded (E308L-T3) and gas-shielded (E308L-T1) wires. The reason for evaluating gas shielded wires for self-shielded underwater welding is that the traditional self-shielded wires, such as AWS Class E308L-T3, are designed for use in a high nitrogen atmosphere (air), which is not a factor underwater. The gas-shielded wires are aimed at operating in an active gas environment, such as CO_2, and it was felt that they may perform well in an aqueous environment. During the evaluations, it was noted that the commercially available stainless steel self-shielded wires (E308L-T3) performed better than the gas-shielded wires. The self-shielded wires created a gas bubble around the molten weld bead and produced a slag covering which can be easily brushed away. This slag covering helped produce a smooth convex bead similar to those made in the dry but with a little more bead height.

It should be noted that all the development FCAW work described in this interim report was completed using commercially available consumables, or slightly modified commercial products. As a result, none of the filler materials were intended for use underwater and the performance discussed for each consumable is strictly related to this application.

During the evaluations, several key parameters were identified for wet flux-cored welding. Voltage, amperage, and travel speed are important, just as they are for other welding applications. In addition, inductance was a key variable, affecting arc stability and bead appearance. Higher levels of inductance, which was adjustable on the Kemppi Stickout, the contact tip-to-workpiece distance, was also important. A short stickout distance increased spatter and a long stickout distance created poor bead appearance. A stickout of 1/2-in. worked well for 1/16-in. wires, and a stickout of 3/4-in. was used for 3/32-in. wires. One additional observation was that the contact tip condition was important to the performance of the weld. Even with a new contact tip the arc characteristics were not always consistent. A Brush Research Company contact tip using a spring loaded ceramic ball to maintain one contact point for the wire assisted in eliminating contact tip problems and provided a consistent stickout. Lead angle is another factor which was found to be critical. After a series of tests, lead angles of 15° to 45° for flat applications and 45° for out-of position welds were selected.

A deviation of more than ±5° was found to have a noticeable effect on the weld appearance.

One of the characteristics of underwater wet welding appears to be that the range of suitable FCAW welding parameters is narrow. Reports by others (5,7) verifies this observation for other wet welding applications. Typically, once a set of parameters was established during the welding trials, only minor adjustments could be made without adversely affecting the weld quality. As an example, it was found that voltage needed to be maintained within 1-2 volts of the selected value to maintain good bead appearance. This may be due to the higher voltages required when welding underwater and/or the arc characteristics of the power supply (8).

Most self-shielded flux-cored consumables operate most effectively when used with reverse (DCEP) current. However, for underwater applications it was observed that straight (DCEN) polarity provided the best bead appearance and penetration with the flux-cored process. Pulsed power was helpful in improving penetration and bead appearance, reducing the "cold" appearance typical of many underwater welds, without the need for a high average current level.

The major variable for obtaining acceptable, repeatable, flux-cored underwater welds is the consumable. For a number of the filler materials evaluated, a suitable set of welding parameters could not be established that would provide a quality weld. After an extensive set of tests, Weld Mold 308L FC-0 wire in 0.045-in. or 1/16-in. diameter for semi-automatic and 1/16-in. or 3/32-in. diameter for automatic applications was selected. This material provided a sound weld with acceptable mechanical properties and bead appearance. The bead appearance is acceptable for single pass welding and, with minimal cleaning (manual wire brushing), multi-pass welds were performed that exhibited no porosity or slag (Figure 4-5). In addition, a sample of Weld Mold 316L FC-0 was tested with similar weld quality and bead appearance. The major drawback for both of these materials is the relatively narrow parameter range for underwater welding. To achieve a good weld each input parameter had to be carefully adjusted within the small band established during the matrix testing.

Some modifications were attempted with the Weld Mold wire to see if the suitable range of parameters and/or bead appearance could be improved. A series of wires were fabricated for EPRI by Weld Mold which were simple modifications to the existing product. The first of these was with a 50% overfill of the flux, to determine if a heavier slag layer would assist bead appearance. Tests showed that

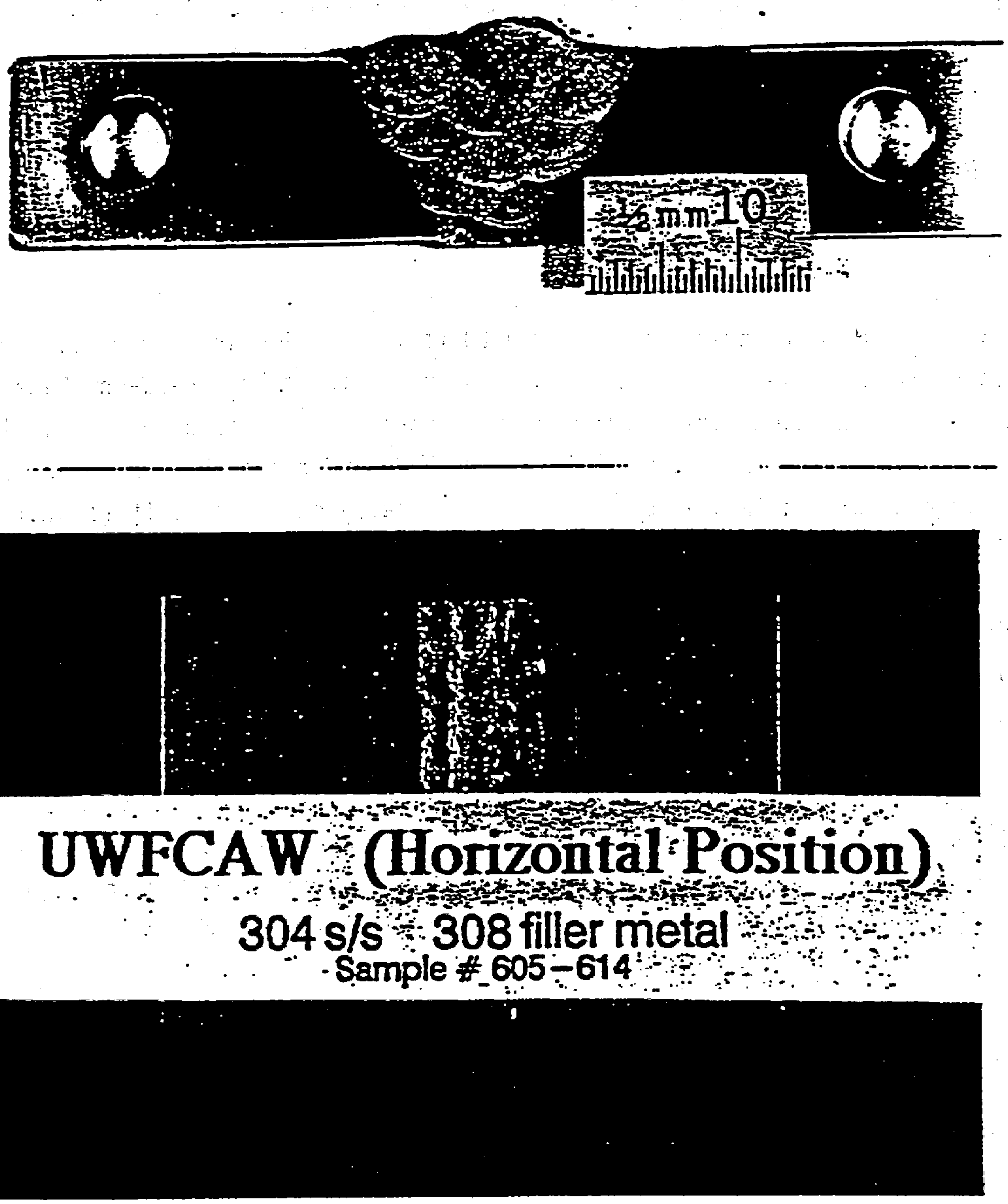

Figure 4-5. Cross-section and surface views of a 308L multi-pass underwater wet groove weld, deposited with the pulsed FCAW process. Backing bar has been removed.

this wire provided a high amount of spatter although the bead appearance was good. A second wire, using the 50% overfill with an increase in silicon, was also prepared. This assisted in eliminating the spatter, but the weld bead had a peaked appearance. Two other wires were fabricated, both containing aluminum additions to provide additional deoxidation. One wire had an aluminum content of 0.5% and the second wire contained 1.0% aluminum. Both wires exhibited poor bead appearance. None of the modified wires provided a substantial improvement over the commercial composition. The commercially available Weld Mold 308L and 316L FC-O wires were selected for further testing.

After obtaining positive results in the 3-ft tank, the manipulator was moved into the 20-ft tank. The program objective for work in the 20-ft tank was to evaluate the effects of pressure on the weld deposits and to look at multi-layer groove welds.

After setting up the test system in the 20-ft tank, all welding variables were set exactly as they were in the 3-ft tank. The resulting single bead-on-plate welds did not show any significant change in bead appearance. Based on the earlier surveys, this was expected. Studies by others have indicated that no observable changes can be seen in the weld arc or bead appearance above 33 ft (1 ATM) (9,10,11,12).

During this development program, an EPRI member utility requested support in evaluating the potential for adding weight to some of their jet pump riser brackets. This particular utility was having trouble with vibrational harmonics that could crack the bracket welds and, based on a computer model, had determined that by adding additional weight, the resonant frequency could be changed, thus minimizing the vibrational problem. It was determined that a three-bead weld metal deposit, 17-in. long, on the top and bottom on each bracket would provide the necessary weight and shape to correct the vibrational problem.

The selected 308L stainless steel filler metal that had been producing the best underwater welds was not originally designed for multi-pass welding in dry applications. With careful control of the parameters, it performed quite well underwater, providing welds that were porosity and slag free. A second requirement from the utility involved the demonstration of a multi-pass groove weld. The groove weld consisted of a 3/16-in. thick plate with a 1/8-in. root gap and a backing bar. The weld was performed with a single root pass with good penetration into the backing bar and good tie-in to the side walls (Figures 4-6 and 4-7). The

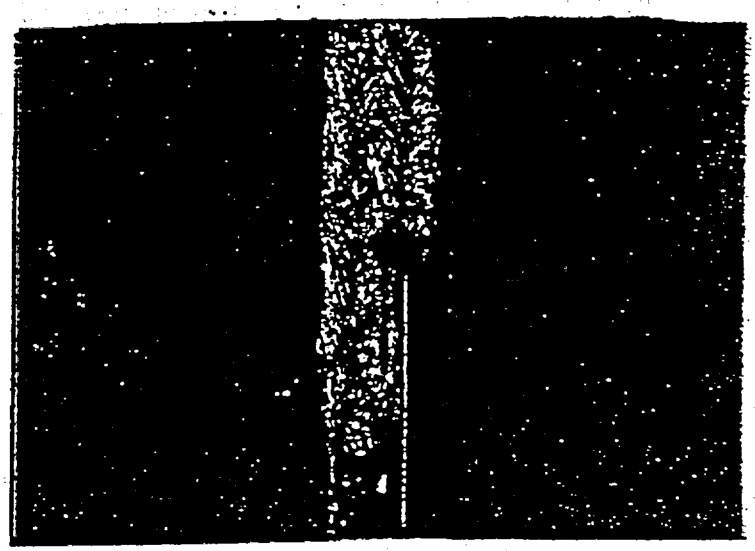

Figure 4-6. Surface view of a 308L multi-bead underwater wet groove weld deposited with pulsed FCAW process.

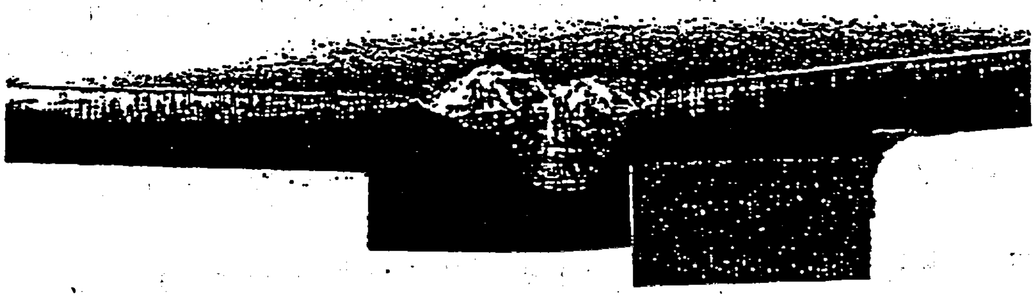

Figure 4-7. Cross-section of a 308L multi-bead underwater wet groove weld deposited with pulsed FCAW process.

groove was then filled with two additional passes and sent for mechanical testing and evaluation. Evaluation of the cross-section showed that the weld was porosity and slag-free, there were no inclusions, and there was no lack of penetration into the side walls or into the preceding beads. Tensile tests exceeded 93,000 psi, with failures occurring in the base metal. Bend tests and radiographs were acceptable in accordance with the requirements of ASME Section IX. Although the utility did not have to make a repair during the planned shutdown, they were ready to utilize this new technology had it been required.

During the course of the wire/weldability evaluations, a second power supply was evaluated. A Miller Pulstar 450, which is a transformer/rectifier-type unit was used with good results. This machine is typical of older power supply designs, using a large transformer and a set of rectifiers to provide welding current. In addition, this type of machine has a characteristically slow response to changing arc conditions as a result of transformer inductance. In comparison, an inverter-type of power supply has less inductance and tends to react quickly to arc conditions. This can be modified through the use of electronic modifications to provide a slower response, which was done with the Kemppi inverter-type unit at the NDEC. The Miller provides a very smooth arc and the weld exhibited a bead appearance and quality similar to those provided by the Kemppi system. The Miller unit offers fewer features, does not offer variable pulse rates (only 60 and 120 cycles/second are available) and is larger than the inverter systems. However, the additional inductance level assisted in providing a smooth, stable arc and, as a result, it was more easily adjusted to obtain a good quality weld. An Airco Mobile-matic wire feeder was used with the Miller power supply. It should be noted that the intent of evaluating these two power supplies was to compare the performance of an inverter-type pulsed unit versus a transformer-type pulsed unit, and should not be considered a proper comparison of the specific brands. Both units provided quality welds for our automatic, underwater FCAW evaluations.

In addition to the studies performed with the fully mechanized welding equipment, a series of tests were performed to determine if semi-automatic underwater flux-cored welding was feasible. Initial tests were done at a shallow depth using both the Kempii and Miller power supplies. These tests demonstrated that semi-automatic FCAW could be performed with good results. Using the transformer-type power supply (Miller), a weld with a smooth bead appearance could be produced. The Miller power supply provided a very stable arc and as a result was much more forgiving for semi-automatic FCAW. Additional tests were then performed at the 20-ft depth, to evaluate underwater semi-automatic FCAW. Fillet welds (Figure 4-

8) and bead-on-plate welds were performed to determine bead appearance and penetration. Excellent results were obtained using the Weld Mold 308L FC-O wire, in 1/16-in. diameter. Evaluations were performed, with promising results, using 0.045-in. diameter wire with the Kemppi PSS5000. Proper evaluation of the 0.045-in. wire was difficult, since the Miller Pulstar has a minimum current setting of 200 amps, which is in excess of the optimum current level for this wire. An Airco Pulse Arc 400 is presently in use for semi-automatic, 0.045-in. welding tests.

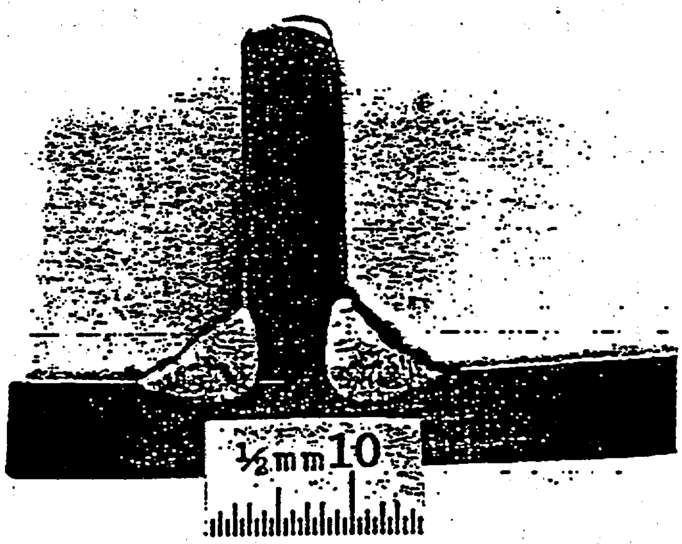

Figure 4-8. Cross-section of a stainless steel fillet weld performed with the semi-automatic underwater wet FCAW process. Weld was completed in the vertical position with 308L FC-O filler material.

The parameters used for wet, underwater, automatic flux-cored arc welding are included in Table 4-3. These parameters are guidelines and differences in equipment, water chemistry, or consumables can affect the welding performance.

Table 4-3

UNDERWATER WET FLUX-CORED ARC WELDING PARAMETERS (17)
(Based on test result at the EPRI NDE Center)

I. Automatic Flux-Cored Arc Welding									
			Kemppi PSS 5000 Settings						
Diameter	Avg. Amps	Avg. Volts	Pri. Time (msec)	Bkgd. Time (msec)	Pri. Volts	Bkgd. Volts	Pulse Rate (pps)	WFS (ipm)	TS (ipm)
1/16-in.	135-155	29-31	4.0	3.25	32	27-28	135	173-196	12
3/32-in.	225-250	27-29	4.0	3.25-4.0	32	28	.125-135	134-160	12

II. Semi-Automatic Flux-Cored Arc Welding							
			Miller Pulstar 450				
Diameter	Avg. Amps	Avg. Volts	Pri. Amps	Bkgrd. Volts	WFS (ipm)	Pulse Rate (pps)	TS (ipm)
1/16-in.	200	30	210	30	190	120	10-15

Note:

Filler Material:	Weld-Mold 308L FC-O
Polarity:	DCEN or DCEP
Lead Angle:	45 degrees
Stickout:	3/8-in. for 1/16-in. diameter wire 1/2-in. for 3/32-in. diameter wire
Base Metal:	304 stainless steel

Water Chemistry

Differences in water chemistry can affect the welding characteristics. The water used for a majority of the tests conducted under the EPRI Reactor Internals Repair Project, was conventional tap water supplied by the City of Charlotte, NC. The water has a chlorine content of 2 ppm. This is the same chlorine content maintained by our filtration system for the large, 20-ft deep tank. The water in the smaller tank was replaced each day from the city water supply, or more frequently depending upon water clarity. Tests were also performed with the mechanized flux-cored welding process, using deionized water to determine if parameter changes would be required for in-vessel repairs. No apparent differences were noted in the welding performance compared to welds performed with the city tap water.

Section 5

CONCLUSIONS

To date, over 30 different FCAW and SMAW welding wires have been evaluated, producing underwater welds in all positions on stainless steel materials. These welds have demonstrated that:

- High quality stainless steel underwater welds can be obtained using shielded metal-arc welding (SMAW) or flux-cored arc welding (FCAW).
- Pulsed current assists in improved penetration and bead appearance for FCAW and SMAW.
- Self-shielded FCAW can be performed remotely, permitting application in limited-access areas.
- Semi-automatic underwater, stainless steel FCAW is feasible and can provide excellent weld quality.
- Underwater stainless steel welds, performed at depths of up to 20 ft, have mechanical properties similar to conventional welds.

Weld qualification tests have been performed at a depth of 20 ft for both the SMAW process and the FCAW process, with 308L filler materials on 304 stainless steel base materials.

Development work is on-going under the EPRI program. Additional work will include the development of new 308L FCAW filler materials to allow the welder/operator more variability in his welding input parameters. Further development work is also planned for nickel-based welding alloys, as these materials are common in the reactor pressure vessel.

Section 6

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10. C. E. Grubbs. Global Divers Inc., New Iberia, LA, Personal Discussion.

11. J. E. O'Sullivan. Pennsylvania Power and Light Co., Berwick, PA, Personal Discussion.

12. S. J. Ibarra. Amoco Corporation, Naperville, IL, Personal Discussion.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

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MEETING WITH BOILING WATER REACTOR VESSEL AND INTERNALS PROJECT AND NRC STAFF

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PUBLIC MEETING

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Nuclear Regulatory Commission
Commission Hearing Room
11555 Rockville Pike
Rockville, Maryland

Monday, May 12, 1997

The Commission met in open session, pursuant to notice, at 3:05 p.m., the Honorable SHIRLEY A. JACKSON, Chairman of the Commission, presiding.

COMMISSIONERS PRESENT:

- SHIRLEY A. JACKSON, Chairman of the Commission
- KENNETH C. ROGERS, Member of the Commission
- GRETA J. DICUS, Member of the Commission
- NILS J. DIAZ, Member of the Commission
- EDGAR McGAFFIGAN, JR., Member of the Commission

STAFF AND PRESENTERS SEATED AT THE COMMISSION TABLE:

- JOHN C. HOYLE, Secretary
- STEVEN BURNS, Deputy General Counsel
- HUGH THOMPSON, Deputy Executive Director for Regulatory Programs
- THOMAS MARTIN, Acting Associate Director for Technical Review, NRR
- MICHAEL MAYFIELD, Chief, Electrical, Materials & Mechanical Engineering Branch, RES
- JACK STROSNIDER, Chief, Materials and Chemical Engineering Branch, NRR
- SAMUEL COLLINS, Director, NRR
- CARL TERRY, VP, Niagara-Mohawk, Chairman, BWRVIP Executive Committee
- ROBIN DYLE, Project Engineer, Southern Nuclear, Chairman, BWRVIP Assessment Committee
- PETE RICCARDELLA, Structural Integrity Associates

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PROCEEDINGS

[3:05 p.m.]

CHAIRMAN JACKSON: Good afternoon, ladies and gentlemen. The purpose of today's meeting between the Commission, representatives of the Boiling Water Reactor Vessel and Internals Project and the NRC Staff is to discuss potential policy issues associated with the NRC staff technical position regarding alternatives for augmented inspection of the reactor vessel.

The NRC staff and representatives of the BWR Vessel and Internals Project have interacted over the past 18 months with regard to a proposed alternative to augmented inspection of the reactor pressure vessel.

In a recent Commission paper, SECY 97-088, the NRC staff stated that no alternative to the expedited reactor pressure vessel inspection requirements would be authorized for boiling water reactor licensees until they have completed at least one examination of essentially 100 percent of their reactor pressure vessel welds and have shown that the examination performed provides an acceptable level of quality and safety.

In its letter dated April 18th, 1997, the BWR Vessel and Internals Project stated that an alternative to the current augmented inspection requirements for all domestic BWRs is warranted. The Boiling Water Reactor Vessel and Internals Project stated that based on a comprehensive study of the reactor pressure vessel design, manufacturing process, in-service inspections to date, operating experience, and extensive probabilistic analyses, only longitudinal shell welds need to be inspected.

Many of these potential policy issues are linked to the staff's determination concerning whether the BWR Vessel and Internals Project's proposed alternative provides an acceptable level of safety.

The Commission looks forward to the discussion with both representatives from the project and the NRC staff. I must say the Commission is interested in understanding what, if any, technical issues relate to policy issues that need to be resolved, and understanding to what extent risk has been considered in the proposed alternative and in the staff's proposed position, and the implications of the staff's position on the industry's time line for performing the augmented inspection.

I understand that copies of the presentations are available at the entrances to the meeting. Unless my fellow Commissioners have any opening comments, Mr. Terry, I guess, you are leading this part of the discussion.

MR. TERRY: Thank you very much. We do appreciate the opportunity to come here.

I'm Carl Terry. I'm vice president of Niagara Mohawk and also chairman of the BWRVIP executive committee. There are a number of other people here from the VIP. In fact, we represent a group of 21 utilities and 36 plants. Eight of those utilities and 15 of those plants are represented here today if we do get into more detailed discussions.

The other thing, up here with me is Mr. Robin Dyle from Southern Nuclear and Dr. Pete Riccardella, who are here to support me in presenting. I do believe that our presentation will come to point as far as the specific questions you raised regarding what is our

proposed alternative, what are the risks associated with that, and the associated benefits with going ahead with this alternative approach.

CHAIRMAN JACKSON: You also should speak to why you feel the Commission should be involved in resolving what many might consider to be technical issues.

MR. TERRY: Okay. Thank you.

First off, we did provide some slides in advance of the meeting. These slides are slightly different, although technical content will not vary really substantially.

MR. DYLE: They are right there in front of you.

MR. TERRY: And they are there in front of you.

As far as the presentation today, going to the agenda slide, we're going to be -- after I make a few remarks, Mr. Robin Dyle will provide additional detail relating to the inspections that have been performed and the details that we're proposing, along with information relating to those in-service inspections.

Dr. Riccardella will go over the basis for our safety assessment of this issue, and then Robin and I will provide some summary remarks.

Going on to the introduction, what we are proposing is an alternative for the BWR RPV shell weld inspections. We believe that that's based upon a very sound and thorough technical evaluation, as well as included in that are deterministic and risk evaluations of these inspections.

The proposed alternative we believe would result in significant savings. These are both savings in exposure, radiological exposure as well as cost, for the industry with no measurable impact as far as safety.

It is important that this issue be resolved. The reason that we asked to be here and talk to the Commission is because we understand there is a disagreement between us and the staff in terms of the recommendation. We believe it has a sound basis and we felt this would be the most expeditious way of addressing the issue. And w are here today to request the Commission's approval of this proposed alternative.

Just very briefly as far as a little bit of backdrop against the rule, on the history slide, of course, the rule was promulgated in September of 1992, and at that time, there were opportunities to provide comments by the industry. However, following the issuance of that rule, we actually formed the BWR Vessel Internals Project. That came out of a consolidated effort to address some issues related to vessel internals specifically, but also, as part of that, we did include inspections and evaluations relating to the reactor pressure vessel.

As a result of that group effort, we did initially meet with the NRC to discuss our technical approach in July or August, rather, of 1995. The reason we did that is -- and we're going to get into this in a little more detail -- the primary issue really related to the fact that we couldn't literally meet the rule without some relief, anyway. So we got into looking at alternatives.

Following that initial meeting, we did submit a detailed report with our proposal in September of 1995. Around the middle of last year, we had a meeting with NRC technical staff and senior management, and as far as we know, while there were some requests for additional information that came out, there really are no unresolved technical issues that we know of as far as our submittal.

As far as the specifics on approach, we looked at a number of options, whether it was an exemption to the rule and other things, and we ultimately determined that an authorization for a technical alternative would be an acceptable legal approach to get this job done.

As far as our proposed alternative, in summary, the current RPV shell weld inspection requirements call for essentially 100 percent inspection of all circumferential and longitudinal welds in the shell weld area.

What we are proposing as an alternative is to inspect essentially 100 percent of the axial welds, i.e., the same as the rule with some minor access clarifications, and zero percent of the circumferential welds. We believe this can be handled as a technical alternative under the current regulations.

With that background, unless there are questions, I would like to move now to Mr. Robin Dyle. Robin is from Southern Nuclear. He's technical chair of our Assessment Subcommittee on the VIP and also he's active in a number of ASME Code committees. Robin.

MR. DYLE: Thank you, Carl.

On the Code and regulatory background slide, just a few key points to make from there. The current Code requirements are to do the 100 percent of the circumferential and longitudinal seam welds, as Carl mentioned; however, we believe that's inappropriate because of a couple of things.

One, the Code treated the BWRs and PWRs as the same when they promulgated the Code changes, and those of us who were there understand that. There's no differences accounted for at a Code level between the experience that BWR would see from fluence, there's no differences in regard to say a PTS event, which is not possible on a BWR but it is on a P.

Secondly, there was no difference in the Code between an axial and circumferential weld, and the stresses are different. So there should be a technical basis for treating those different as far as allowable flaw sizes in, we think, the inspections that would be required.

Secondly, from the regulatory requirement standpoint, when the staff put the rule together in 1992 and invoked the 100 percent requirement from Section 11, they did not consider the differences either in these two points. They treated the Bs and the Ps the same. There was no distinction made between, again, the issues such as PTS and embrittlement related fluence, and there was also no difference in the treatment of how you would evaluate a flaw on the circumferential weld or a longitudinal seam weld.

So those two situations led us to think there were reasons to look at this from a technical standpoint.

Again, as Carl pointed out, most BWRs physically cannot meet the rule, and if you go back and look at the construction history, the construction codes required a lot of things, but one thing that was not in place at that time for most of the BWRs was the Section 11 provisions for inspection, and then as the later plants were being constructed and designed, the rules that were there were not very much in the way of what would be required during in- service inspection. So they were designed and constructed in such a way that there are physical limitations that would prevent us from doing these examinations.

When the rule was put forth, I, representing the owners' group, and some others met with the staff about what was the appropriate way to approach this, because we knew plants couldn't do these examinations, we knew the staff would not want to see 30 or 36 individual relief requests. So we tried to come up with a generic approach, and the next slide really is where we are.

When we went off to do this, we said let's not see how much we ought to reduce the inspections; let's start from ground zero and say what would be the right thing to do for a BWR vessel? What should we inspect? Where should we focus our inspections?

Quite frankly, we were surprised at what we came up with and the recommendation that we have, because we would have thought there would have been more required. But when you go through the technical evaluation again, we think what we're proposing is legitimate.

What we focused on first was safety. We have to operate the plant safely. We have to deal with risk, we have to deal with exposure of personnel.

The second impact or the third impact there would be the cost. There is a cost associated with this, and we looked at that.

Then the last thing, we recognize that the staff has to have defense in depth. That's just a given. They have to know that what we're doing as an industry provides enough assurance that we're operating the plant safely.

So those were the criteria we used and that was the approach that we took going in to try to figure out where we ought to go.

If you would skip two slides over to slide number 9, labelled BWR Fabrication. Just a couple of real quick points I would like to make from a background standpoint. When we went back and looked at this from a fabrication standpoint, here are some of the items that described the vessels. It's shells with rolled plates, you have vertical seam welds and you have circumferential welds.

There are three different welding processes that could be used and there are different cladding steps. One machine clad for most of the shell courses, and then a manual back clad on the field welds.

The bottom line is the seam welds and the cladding all receive a post-weld heat treat, so that's a good thing to have happen.

Also, repair welds, when they were necessary, they were documented and tracked to the same degree that all the vessel welds were done so that you have a high quality repair there, and we know where those repairs are. They're located and we can go find them.

On the next slide, when you get into the fabrication inspections, and I won't go through all the details of those, but there are multiple inspections required. You can see radiography down to penetrant. And then what we've listed there on the presentation are the acceptable flaw sizes that we're concerned with, and construction code.

Typically, if you look at the way these things were put together, the vessel would get an RT and an MT; then you have a PT after cladding to make sure the cladding was put on; and both of these steps ensure that you don't have surface breaking flaws. Then there was a hydrostatic test performed, and then there was another magnetic particle inspection done. All of this to assure that we don't put the vessels in service with large defects, and that's where we are. We think the inspection summary will show that, also.

The next slide on the operations just again is a background to try to point out a little bit of the difference between the BWRs and PWRs.
The BWR, as you're, I'm sure, aware, operates so that the steam region moderates the vessel responses. You have the normal heatups and cooldowns along the steam saturation curve.

One of the key things is the vessel temperatures are normally 100 to 300 degrees above the P-T curve. So we're always in the ductile region, you're always on the upper shelf.

The pressure test after each outage is limiting integrity challenge, and that's done normal operating pressure but at a lower temperature, so it stresses the vessel a little more. But the plant's in cold shutdown and the pressure is carefully controlled and you have the rods in. So if you were going to challenge the vessel, that would be the right place.

The bottom line is, is that if you verify integrity when you've done the pressure test, you're good to go for a cycle. The worst that you could ever postulate happening would be a leak during operation. There would not be any brittle failure. And that's important to know.

CHAIRMAN JACKSON: How does the leak test tell you that?

MR. DYLE: Because if you look at the evaluations we've done, and Dr. Riccardella will get into it in more depth, if you don't fail during the leak test from the structural evaluation, you'll go up and you'll be in the ductile region, so that if you did have anything, the only thing you would have would be a leakage. You wouldn't have a brittle failure of the vessel at operation because your temperature --

DR. RICCARDELLA: The ductility of the material is temperature dependent, and so it tends to be more brittle when it's cold than when it's hot, and we conduct this pressure test or leak test when the vessel is cold.

CHAIRMAN JACKSON: Okay.

MR. DYLE: If you would, turn to Slide Number 13. It's labelled 1997 ISI Summary. And just to give you an update, this is an update from what was originally provided in our report, BWRVIP-05.

We now have responses from 37 domestic units and three international units, and we've got all six designs represented in the results.

Of interest here is back in 1995 when we looked at this, we had over 440 cumulative years of operation. Obviously, we have more in that range. There's some plants that have now operated seven to ten years, and some of them out in the 25- to 30-year range. So we've got a broad perspective.
There are over 16,000 total feet of category B-A weld that could be inspected, and the category B-A comes from Section 11. Of that, over 8,000 feet has undergone a full code examination, and an additional 700 feet has received a partial code examination where you may have had limitations, could only do one side of exam, or limitations due to transducers.

On the next slide --

CHAIRMAN JACKSON: Let me ask you a question. I'm told that in 1990, that inspections of BWR reactor vessel heads at Quad Cities and Fitzpatrick identified surface cracking and sub-surface flaws. Now, can you discuss the implications of those within the context of the conclusions that you reach?

MR. DYLE: Pete?

DR. RICCARDELLA: Yes. That cracking mechanism was specifically addressed in the evaluation, and you'll see how we did address it when we get into the probabilistic fracture mechanics.

MR. DYLE: It was -- surface cracking wasn't associated necessarily with the actual shell welds; it was in the head region.

Back to slide 14, just going through the summary briefly, as I said, there's over 8,000 feet that's been examined, full and partial code examinations. Of that, over 7,000 feet has been examined using techniques which satisfy Regulatory Guide 1.150.

We asked the EPRI NDE Center to evaluate those techniques and their conclusion was, along with ours, that if a procedure was used that satisfied the Regulatory Guide, there was a high degree of probability that we would find the flaws of concern when we did our inservice examinations. So we're confident that those exams are valid and give us good information about the status of the reactor vessel.

To date, out of all the examinations we've done, there's been 17 indications that required evaluation. There's been others that were acceptable to code evaluation criteria. These 17 were all sub-surface. When you do the fracture mechanics that's required by WB 3600, they are found to be acceptable. And of that, the cumulative length of these indications were 31 inches or .03 percent of the weld length that we've examined.

COMMISSIONER ROGERS: How many of those were in circumferential welds?

MR. DYLE: I would have to go back and look for the exact number. The majority of them are in circumferential welds, which -- but again, they were sub- surface, they were manufacturing type defects, and they weren't anything that occurred inservice. So they would have been there all along historically. I could go back in the report and pull the data out and try to get that number for you.

The last item on the page just simply gets to the cost in man-REMs. The average cost when we did the survey is about $3.3 million per interval, which is a ten-year time frame. The interval comes from Section 11. Some units would be less; some would be significantly higher.

Also, the average exposure associated with this was 12.2 man-REM, and that's just to do the inspection. Those numbers would go up for plants that do examinations from the outside diameter. Also, as the plants age, that number could get worse also.

The conclusion of the survey, shown on Slide Number 15, is that the inspections done to date demonstrate the shell seam welds are free from unacceptable fabrication defects which you would expect from the manufacturing processes that were used. We also found no flaws developing during operation.
This evidence supports the conclusion there's on degradation mechanism that's affecting the seam welds and all of these things combined together supports the reduction in inservice inspections that we're proposing.

The next slide is what we propose to do in the future, and that would be that we'll use a demonstrated technique and procedure. We're going to do the right kind of NDE, we'll make sure it can accurately size and detect the flaws of concern, and it will enhance our ability to do that.

Also, as we do these vertical weld examinations, the way they'll be done is in such a way that when you run across a circumferential weld at the intersection, that weld will also be interrogated at the intersection. What this allows us to do is to continue to collect data on the most risk-significant welds and not do the inspections on those that are not risk significant.

CHAIRMAN JACKSON: Let me ask you a question about terminology.

MR. DYLE: Yes, ma'am.

CHAIRMAN JACKSON: What do you mean when you say a risk-significant weld? Aren't all reactor pressure vessel welds essentially risk significant?

MR. DYLE: I think when Dr. Riccardella gets through, you'll see that there are orders of magnitude difference between the vertical seam welds and the circumferential seam welds.

CHAIRMAN JACKSON: That may be the case, but are you telling us that we should believe that circumferential welds are not risk significant? That's your basic position?

DR. RICCARDELLA: I think, first off, understand that certainly a failure of either vertical or circumferential welds is significant, and that's not our point here at all.

What we really want to get to is the risk contribution that's made by doing or not doing inspections of these welds which is coupled to the probability of circumferential welds actually failing. We're certainly not here to tell you that it's unimportant that circumferential welds fail. That would be significant.

MR. DYLE: It's a relative contribution, yes.

That concludes --

CHAIRMAN JACKSON: So you don't mean the risk significance of the weld; you mean the probability of failure of the weld?

MR. TERRY: Right. And we're talking about the risk significance of the decision to inspect or not inspect. That's really the key point here.

DR. RICCARDELLA: The probability of failure is so small as to make the risk insignificant.

MR. TERRY: I think Dr. Riccardella, when we get to his presentation, you'll understand more precisely where we're coming from.

MR. DYLE: That concludes my remarks, unless you've got any questions about that. Dr. Riccardella, who was one of the primary authors and did the fracture mechanics evaluation, is next.

COMMISSIONER ROGERS: Well, I have a question. I don't know where the best place is, but what about the possibility that the weld materials of the circumferential and the vertical welds are not the same? What could be the implications of that possibility?

DR. RICCARDELLA: In our analysis, we've taken into account statistically the possible variability in the properties of both types of welds. We've analyzed the probability of failure considering the variability in the material properties, and as you see, the results come out - - the results that come out are very striking.

COMMISSIONER ROGERS: All right. Why don't you go ahead.

DR. RICCARDELLA: What I will present is an overview of the methodology that we used in conducting this probabilistic fracture mechanics evaluation, some key features of the analysis and conservatisms in the analysis as well as just a quick overview of the results and conclusions.

As has been mentioned, the details of this analysis were presented in this BWRVIP report which was submitted to the staff in September of '95. That was followed by a two sets of requests for additional information which we responded to. I think that the overall volume of paper submitted on this topic was probably about four inches thick worth of response to the RAIs, and our understanding is that all of the technical questions on our analysis methods and conclusions have been answered and that there are no technical issues remaining unresolved on this analysis.

On the next slide, I'll talk a little bit further about the inherent flaw tolerance of BWR and specifically the differences between a PWR and a BWR in this area.

One of the major points is that the BWR vessel is about twice the diameter of a PWR vessel. This creates a much larger annulus of water between the core and the vessel, and the result is lower irradiation fluence in the vessel and, therefore, lower irradiation embrittlement.

The reference temperature, that is the brittle to ductile reference temperature for a BWR varies from -- at end of life varies from 60 to 150 degrees F versus almost twice the value, 300 degrees F, for a PWR. As a result, the material remains ductile. This is for both longitudinal and circumferential welds. The material remains ductile during all normal and transient operating conditions.

This results in an inherent flaw tolerance for longitudinal seam welds for the limiting pressure test condition and the ASME code quarter-inch reference flaw of a safety factor of four against brittle fracture, which is more than twice -- which is twice the code required safety factor of two.

It also leads to the fact that a through-wall crack that's ten times as long as it is deep does not exceed the fracture toughness of the vessel even in the worst irradiated beltline region.

These first two points are made for longitudinal seam welds. Circumferential cracks exhibit even higher safety factors. This is because fundamentally, the pressure stress in a circumferential weld is half the stress in a longitudinal weld.

You've asked about potential service degradation mechanisms. Two that immediately come to mind are fatigue and stress corrosion cracking.

Fatigue is relatively inconsequential in the beltline and in the shell region of a BWR. The vessel system cycling events are very slow and the fatigue usage resulting from these events is very low. There is no rapid cycling or severe thermal fatigue cycling mechanisms that are applicable to the BWR vessel shell region.

Stress corrosion cracking you mentioned the Quad Cities had -- it's definitely a concern in BWRs, both for stress corrosion crack initiation in the cladding as well as the potential for stress corrosion crack growth in the low alloy steel vessel material. The SCC in the cladding has been observed in the field. The SCC growth in the low alloy steel material has been observed only in the laboratory; it hasn't been observed in the field. But both of these mechanisms were specifically addressed in the probabilistic fracture mechanics analysis.

On the next slide, I show an overall schematic of the analytical approach. I think you can read this. Basically it's a Monte Carlo probabilistic fracture mechanics evaluation technique where we select samples from a weld, either from a longitudinal seam weld or from a circ weld. I show here we're sampling an axial or longitudinal weld. A crack is assumed to exist in that sample, and the probability of that crack comes from two sources as shown in the arrows leading to the upper box on the right-hand side, probability of crack size.

We have included both the probability of a manufacturing defect existing in the vessel in accordance with the standard Marshall distribution. This is the distribution that is -- the well known distribution that's been known in PTS evaluations and has been verified with respect to destructive examination of the Midland vessel.

In addition to that, we take into account the potential for cracking to initiate in the cladding, and so we have two potential sources of cracks -- of cracks being distributed in this sample that we selected.

Then, with operating time, we consider the potential for crack propagation, again in a probabilistic manner considering IGSCC crack growth data and the stress distribution both due to normal operating stresses plus potential clad stresses, and then we have the ability to superimpose upon this the inspection or non-inspection.

So we can have certain -- depending on what percentage inspection we assume, we can have certain of these samples that come through the Monte Carlo analysis subjected to inspection and others not inspected, in which case, if we consider inspection, then we superimpose a probability of detection on that inspection and so then we have a remaining probability that this crack will exist, and then we make a comparison of the resulting crack size to the critical crack size, and in doing that, we look at the initial material properties, RTNDT, the possible variation of copper and nickel content in the weld, and the fluence versus time in the weld. So we make a time comparison of K versus KIC.

This is the basic analytical technique that we use to address this problem.

The next two slides, I talk about the key features of the analysis, and I will point out that the starting point for this analytical methodology was the method developed by the NRC to address PWR pressured thermal shock, namely the VISA code which was developed at Battle Northwest -- at Northwest Laboratories.

This includes a probabilistic treatment of the vessel fracture toughness and the radiation embrittlement concerns; the assumed fabrication defects in the vessel, specifically the Marshall distribution with all of the -- all of the defects in the Marshall distribution were artificially moved to the vessel ID surface, which is conservative from the standpoint of a radiation embrittlement, but we did this to be -- and also conservative with respect to stress corrosion crack growth, because that's where the corrosive environment is. We did this to be consistent with the NRC methodology for PTS.

As in the VISA code, it's a multiple random variable, Monte Carlo analytical approach that we used.

We did have to add -- on the next slide -- some features to the methodology to make it specific to analyze BWR vessel ISI, and those include some items I've already mentioned: the treatment of stress corrosion crack initiation in the cladding; the treatment of stress corrosion crack growth in the low alloy steel; the effects of periodic inservice inspection. And because the resulting probabilities are so low, we couldn't just use a brute force Monte Carlo technique. I mean, you'll see in some cases we would have had to take 10 to the 40th iterations. So what we did is we implemented an importance sampling technique out of the literature to speed up and basically to make the calculations feasible.

These are the new features that we added in the analysis. I should mention that we did, for the features that are consistent with the current VISA code methodology, we did benchmark our methodology against the VISA code, show that we got essentially equivalent results, and that benchmarking is documented in the submittals that we made.

On the next slide -- I'm sorry. Previous slide, please.

Another key feature of the analysis is, you know, as you go through these Monte Carlo iterations, a sample either progresses to failure or it doesn't, and the probability of failure is the number of samples out of the total which have progressed to failure.

But what we found was that there were two types of failures that were falling out of the analysis. One is the crack would just grow to the point where we can't analyze it anymore. It got to be 80 or 90 percent through-wall. But we still haven't reached a point where K exceeds K1C. We still haven't predicted a fracture. This is what we would call a leak scenario.

The second type is that somewhere during that crack propagation, due to the combination of a large flaw and a low fracture toughness condition, you would predict K exceeds K1C, and therefore we would predict a brittle fracture.

What we found was the overwhelming majority of cases, even where we did predict failure, were leakage type failures. Something like, you know, 99 out of every 100 failures that we predicted in the analysis were leaks, and only occasionally did we predict a brittle fracture type failure, and when we did, that occurred during the system leak test.

As Robin mentioned earlier, the critical condition from the standpoint of a low pressure stressing of this vessel is the leak test, which is conducted in a cold condition when the reactor is in cold shutdown.

CHAIRMAN JACKSON: So you're arguing that leak before break for the reactor vessel is acceptable?

DR. RICCARDELLA: Absolutely. And it --

CHAIRMAN JACKSON: Why is that acceptable?

MR. TERRY: That's not our argument. I think our argument --

DR. RICCARDELLA: We're doing inspections. We're saying that the analysis demonstrates that if -- in the very unlikely event that we're going to have a problem with this vessel, that that problem would be a leak, not a break. And you will see a little bit further when I present the results exactly how that manifests itself.

Let me just identify some of the conservatisms in the analysis. They are listed here. I have already mentioned the flaws in the Marshall distribution, even though they're generally expected to be distributed through- wall, we've pushed them all to the ID surface.

We have included a conservative treatment of stress corrosion cracking in the cladding. Basically what we said is if our analysis predicts stress corrosion cracking in the cladding, we instantaneously assume that that cladding is through-wall. We take no credit for time for the crack to propagate through the cladding.

We also arbitrarily assume that it lines up with one of these Marshall type manufacturing defects; that is, we haven't assumed that -- as soon as we predict that the cladding is violated, we assume that it's violated over the entire inside surface of the vessel and,

therefore, the Marshall defects will be exposed to the BWR environment and will propagate by stress corrosion.

The rates of stress corrosion cracking in the low alloy steel are based on earlier test data which are shown to be very conservative. More recent test data really shows no stress corrosion crack growth in the low alloy steel, but still we based the analysis on the more conservative data.

As I already mentioned, we have used conservative vessel fracture toughness and radiation embrittlement correlations.

On the next slide, I have a plot, a typical plot of the results of a probabilistic fracture mechanics analysis. There are three curves on this plot. The upper horizontal dash line represents the PTS screening limit; that is, the vessel failure probability that is inherent in the NRC's PTS screening limit.

Then I show two curves. The upper curve designated by triangles is the probability of leakage, and then the lower curve is the probability of actual failure. This is the point that I was alluding to earlier. All of the BWR vessel probabilities are lower than the PTS screening limit, but the probability of a break is much, much lower, it's several orders of magnitude lower versus the PTS -- versus the probability of a leak.

Also, I would address that all of the probabilities shown on this chart are for longitudinal seam welds. We can't even plot the probability of failure or leakage associated with a circumferential weld because it's so many orders of magnitude lower than these.

CHAIRMAN JACKSON: Where is the uncertainty? I mean, these show these as point curves, but whenever you do a probabilistic analysis, you know, there's a certain uncertainty in that analysis, and where would that show up in these curves?

DR. RICCARDELLA: You know, in terms of analytical uncertainties, we have repeated these analyses over and over and we show that they're accurate to within plus or minus a factor of two. I'm not sure if that's what you're asking about, or if you're asking about, you know, potential uncertainties for things that we haven't considered, you know, that we haven't considered in the analysis.

CHAIRMAN JACKSON: I'm asking you about both.

DR. RICCARDELLA: Okay.

CHAIRMAN JACKSON: I mean, there's a certain uncertainty that gets propagated through a probabilistic analysis, and any time you have a probability distribution, --

DR. RICCARDELLA: Yes.

CHAIRMAN JACKSON: -- okay, you're really not talking just simple multiplication or carrying through of point values; you have to recalculate what the distribution looks like.

DR. RICCARDELLA: That's true.

CHAIRMAN JACKSON: And so --

DR. RICCARDELLA: Yes. Those uncertainties are within a factor of plus or minus two on the probability of failure. But, you know, the main point that I would like to make is that these curves are for longitudinal welds, and we're not talking about changing anything for longitudinal seam welds. I would like to make that point with the next slide, which is a table.

In this case, what we've looked at, in this table, the effect on probability -- both probability of failure and probability of leakage of the current requirements, that is the essentially 100 percent of all welds, versus the proposed program, which is essentially 100 percent of seam welds, of longitudinal welds. We have broken this down by the contribution of irradiated longitudinal welds, unirradiated longitudinal welds, and circ welds. And the plot that I showed earlier is what gave the number, for example, irradiated longitudinal seam welds, a probability of failure of 5.68 times 10 to the minus 8. That --

CHAIRMAN JACKSON: With what confidence?

DR. RICCARDELLA: Let's see. I would say within an accuracy of plus or minus a factor of two, but --

CHAIRMAN JACKSON: But with what confidence?

DR. RICCARDELLA: I haven't got a confidence number, confidence interval right at my fingertips.

CHAIRMAN JACKSON: Okay.

DR. RICCARDELLA: But the point is, whatever the confidence, it's exactly the same under the proposed program because we haven't changed anything on longitudinal seam welds when we go from the current requirements to the proposed program. We're talking about the exact same inspection. And likewise, for the unirradiated portion of longitudinal seam welds. We're not proposing any change.

Where we're talking about a change is in welds for which, to the best that we can calculate it -- and here I'm not going to state much confidence in this value other than to state that it's extremely low. We calculated a number of 10 to the minus 40th for the contribution to probability of failure from circumferential welds; many, many orders of magnitude less than that from longitudinal welds. We basically had trouble in any of our Monte Carlo iterations showing a failure, predicting a failure due to a circumferential crack in a circumferential weld.

So what we're saying is that the probability of failure, both failure or leakage, are both already lower than the PTS screening limit and they don't change at all with our proposed program.

So the conclusion slide basically just restates this point. The calculated vessel failure probability is already orders of magnitude lower than the PTS screening limit. This is based on conservative analyses; they could actually be lower if we took some of the conservatisms out of the analysis. The proposed change in inspection scope has an insignificant impact on the already small failure probabilities.

MR. DYLE: Thank you, Pete.

Just a couple of slides and I'll turn it back over to Carl for his closing remarks.

If you look at the slide for impact of implementing the shell weld recommendations, and again, from looking at the probabilistic fracture mechanics, as Pete pointed out, we're not changing anything on the longitudinal seam welds. So comparing apples to apples, there's no change in risk with the program regarding those. But we are talking about removing the circumferential welds, but we don't believe there's any realistic change in the plant safety or risk by not examining those circumferential welds.

Also, we can save at least 200 man-REM in exposure by reducing the number of inspections we do, and that number can go higher for the plants that do OD examinations. As the plants get older and become more contaminated, that number will be greater, also. But that's just from the survey that we've done of what it takes to do the inspections.

There is no consideration in this number for craft support like insulators, scaffold builders and things of that nature. This is just associated with performing the inspections.

CHAIRMAN JACKSON: Do you use similar techniques for doing these inspections as the Japanese use in their reactor pressure vessels?

MR. DYLE: To the best of our knowledge, yes. I know they are working on developing some new tools that we're watching. I believe you may have seen one of them demonstrated at the EPRI NDE Center on one of your visits, and we're eager to see how well that works out. As yet, that has not been done in the field and we're not sure what limitations there will be. But yes, we are eagerly looking for that.

Also, one other thing is we tried to do this in a generic sense in a hope that we could reduce the number of requests for exemptions and relief requests that the staff would have to deal with, because there are so many plants that will not be able to fully meet the rule. They're going to have to deal with exemptions, and this would reduce a number of those.

Finally, there is a significant cost savings to the industry to implement this which would save in excess of $50 million.

CHAIRMAN JACKSON: Commissioner Dicus?

COMMISSIONER DICUS: The 200 man-REM, is that total for all plants?

MR. DYLE: That's total for all plants for one ten-year interval, yes.

The next slide on the current status, where we think we are today with this, we have submitted our technical documentation in the form of the VIP report. We've responded to the staff's RAIs, we provided additional calculations and information on the NDE techniques.

We submitted a request for a technical alternative that's currently pending, and we think we've resolved the technical issues and are awaiting a response to that technical alternative, and that's where we believe we are today.

With that, I'll turn it over to Carl.

MR. TERRY: Thank you, Robin.

In closing, again going back over what we've told you, the BWR vessels were fabricated free of large defects. Robin went over the degree of inspections that were done during the course of that fabrication.

We also talked about the survey results of the ISIs that have been done to date, and they indicate no significant flaws.

In summary, we've looked at about a mile and a half of weld. We found less than three feet of indications, and those were sub-surface indications and are not service- related type flaws.

As far as BWRs, the cold pressure test that we do generally at the end of the outages is the limiting BWR condition. Certainly a failure at any time is not good, but certainly that's -- that's certainly the least risk significant time if a failure were to occur.

ISI of the circumferential welds is really of little value. We see no impact on safety by not doing these inspections, and that's really what's shown by the probabilistic fracture mechanics work that we've done.

As far as the cost savings for reduced inspections, they are substantial with no measurable increase in risk. The inspection recommendations are consistent with what we believe is the right focus, which is to focus the industry and regulatory resources on those issues that really add value from a safety standpoint.

Our alternative specifically is, again, to inspect essentially 100 percent of the axial welds, longitudinal welds, and zero percent of the circumferential welds.

Finally, in closing, by adopting the proposed alternative, the BWR utilities will continue to perform a substantial amount of inspections on the RPV shell welds.

We see no predicted leakage or failure for circumferential welds, and I would point out here that this is something that is unique to the BWRs as far as this condition. The continued inspections of circumferential welds does not add any measurable safety benefit, while it offers substantial savings on the order of 200 man-REM and $50 million for the utilities.

Rapid adoption of this is really critical. We are coming for most plants or a number of plants to the end of this current ten-year interval. This proposal, by the way, is applied for the interval inspections; however, we are coming to the end of the current ten-year interval, making the current review and request for exemption particularly critical and, therefore, we request the Commissioners' approval of this proposed alternative.

CHAIRMAN JACKSON: Commissioner Rogers? Commissioner Dicus?

COMMISSIONER DICUS: One quick question. You're meeting with ASME, I understand, or you have met with them? Could you just very quickly characterize what has come out of those meetings?

MR. DYLE: In our discussions, the item has been discussed at task group and working group and sub-groups responsible for this issue, and the code case, which is based on the

report of doing 50 percent of the longitudinal seam welds and zero of the circumferential, has passed all the way to that point. It is at subcommittee and it is waiting a letter ballot. I'm responsible for writing a white paper to go with that for the members of subcommittee to vote on that.

I have reason to believe there will be a large majority of positive votes there because most of the members also had a chance to vote on this and review the story as it came up through the various committees. And we've deferred writing the white paper so we could roll in any information that might come forward from this meeting so that the code committee is fully aware of everything that's been done.

CHAIRMAN JACKSON: Commissioner Diaz?

COMMISSIONER DIAZ: Just a couple of comments. Obviously, this is a highly technical issue. We certainly appreciate you bringing it up to the attention of the Commission. But I kind of feel inadequate at judging the technical merits of it.

I do believe there is some substantial benefit from addressing the issue again and trying to have the staff, you know, make an additional analysis on your proposal. I certainly don't feel that I can, at this point, address the technical issues on it.

CHAIRMAN JACKSON: Commissioner McGaffigan?

Well, thank you very much.

We will hear from the NRC staff.

MR. TERRY: Thank you.

CHAIRMAN JACKSON: We know who you are.

MR. THOMPSON: I was afraid of that. You know where we live.

CHAIRMAN JACKSON: Mr. Thompson, please.

MR. THOMPSON: Thank you, Chairman Jackson. Good afternoon, Chairman Jackson and Commissioners. Thank you for the opportunity to discuss the staff's position on augmenting examination requirements for boiling water reactor pressure vessels, as we spelled out in our commission paper, SECY 97.88.

At the table with me from NRR is Sam collins, director of NRR; Tim Martin, the acting associate director for technical review; Jack Strosnider, chief of the materials and chemical engineering branch and, from the office of research, Michael Mayfield, chief of the electrical, materials and mechanical engineering branch.

First I would like to thank Mr. Terry, Mr. Dyle and Dr. Riccardella as well as the other members of the BWR vessel and internal projects for their extensive discussion and evaluation that went into the development of their report on BWR reactor pressure vessels shield weld inspection recommendations. Although our judgments differ on how to use the results of their effort, this is an excellent example of their proactive effort in working with the Staff to develop appropriate requirements for inspection and repair of BWR internals,

including the BWR core shrouds, jet pump assemblies, core spray piping as well as a number of other BWR internal components and systems. We believe that these cooperative efforts will resolve safety issues and they benefit everyone.

The staff has carefully reviewed the industry's report and agree that it contains substantial technical arguments for deducing the scope of BWR pressure vessel weld examinations. However, we believe that this reduction should be for inspections following the initial base line inspection that is required by both our regulations and the ESM code.

Our focus today is on the integrity of the reactor vessels, the one component for which there is no redundant safety system. It is vital that its integrity be maintained.

Historically, our ability to predict component degradation has not been perfect. Also, the ASME consensus has evolved over time and currently requires 100 percent examination of the reactor pressure vessel belt line welds every ten years. Today, the staff's presentation by Mr. Strosnider will focus on the need to maintain the defense in depth and to validate the assumptions of the industry's probabilistic model.

I would like to turn the rest of the briefing over to Mr. Strosnider.

MR. STROSNIDER: Thank you. Good afternoon.

First, I would like to indicate that, as Mr. Thompson said, in fact I would like to reemphisize that the industry analysis has provided some substantive arguments for reducing the scope of inspections. So you are not going to hear a general condemnation of their analysis. All right.

But I am going to go through some issues that the Staff considered that led us to conclude that it is appropriate to perform a base line examination before we consider this sort of reduction. Those are the things that I want to focus on.

Specific areas for discussion are listed in the first viewgraph. I want to talk a little bit about the safety significance of the vessel, the rule which you have probably heard enough about now to understand what its intention was, the need for inspections, some discussion about the NRC and ASME inspection philosophies, visions that do exist for relief or alternatives and then our conclusions.

On the next viewgraph talking about safety significance, stated quite simply the assumption is that the reactor pressure vessel failure is an incredible event. Quite frankly, when I got ready to present this particular slide, it was a little difficult for me because we just take that as a given that pressure vessel failure is not something that is credible. The engineered safety features of the plant are not designed to cope with reactor pressure vessel failure. They are not specifically designed for that, either catastrophic failure or leakage. So the consequences of such an event have not really been fully evaluated.

Pressure vessel integrity must be maintained at the highest level of quality and nobody is questioning that statement. An important part of that, Staff's position is that an important part of that is maintaining defense in depth and that is accomplished through inspections and evaluation of inspection results to understand the current condition of the reactor vessel and any potential future degradation modes.

Moving on to the next viewgraph, just a little bit more about the augmented inspection rule. Going back in history to the early to mid-'80s, relief had been granted to the boiling water reactors for performing some of the code required examinations. These were granted under 5055(a) of the regulations. The main reason was the inability to access these locations. The tooling just wasn't available.

However, and the Staff recognized the small amount of inspection that was being performed and, also, at the same time, advances in inspection capability that had occurred, and some of this in particular was overseas where we found that people were doing more examinations, and also recognition of some viable degradation mechanisms that I will talk about later, the decision was to promulgate this rule.

Did require expedited implementation of inspections. This is basically what was required by the ASME, except on a faster schedule because of the concern that time had gone by without any significant inspections. It revoked all the prior reliefs that had been granted and, as I indicated, these were granted largely on the basis that they were just physically unable to do the examinations and it was related to tooling.

Some of the units at that time had inspected less than 10 percent of the shell welds and that is still true today. Even though, as you heard in the earlier presentation, there has been a fairly substantial sample of welds inspected, there are plants out there that have not looked at 10 percent of the shell welds in their plants. I'm sorry, have looked at 10 percent or less.

So the rule was promulgated in '92. The one major comment, public comment that was received on the rule was to provide some flexibility in schedule, specifically for those plants that were near the end of the 10-year inspection interval, that they wanted some flexibility in being able to implement this, do some planning and develop the appropriate tooling. So, in fact, the rule was modified such that plants that were within 40 months of the end of the 10-year interval could go to the next interval, next first period of the next interval. A little bit complicated, but we gave them some extra time to implement the inspections.

Also, it was recognized that even with improvements in some of the tooling and inspection capabilities, that there still may be some areas which are inaccessible and we are talking about where there are lugs or attachments physically inside the vessel such that you just can't get to the weld that you want to examine.

Moving on to the next viewgraph, I want to talk about the need for inspections. First, I would point out the purpose of the reason we perform inspections, just in general. We want to identify problems that we didn't anticipate and, as was noted earlier, prediction of degradation in other components has not always been real reliable. Although in hindsight, some of these degradation modes can be explained, it was really inspections and inspection activities that identified them and examples include stress corrosion cracking in BWR piping.

When this issue first came up, it showed up in some small diameter piping and the thought at the time was it wouldn't happen in large diameter piping. Inspections confirmed eventually that it did.

BWR internals, there have been a number of areas where cracking has been found through inspections and that includes, for example, the access cover holes in the inside of the vessel, the core shroud, which has been getting a lot of attention lately.

So one of the things we want to do is identify things we haven't anticipated. The other thing is that the evaluation of the inspection findings is really a proactive way of looking at the condition of the vessel and, as I said earlier, looking at what potential degradation could possibly occur in the future.

So when indications are found, and it was mentioned in some of the recent examinations indications have been found, they were evaluated, they were found acceptable by the code which is what we would expect, that's what we want. But we also look at those and say, well, what kind of degradation is it? Yes, it is subsurface, it is not exposed to the environment. So, you know, we don't have to be as concerned about that as if it were open to the environment and might therefore see some more aggressive growth.

So those are some of the reasons.we do the research.

CHAIRMAN JACKSON: Let me ask you a question. Is the code meant to be predictive? I mean, is there an established relationship between code-identified degradations and failures?

MR. STROSNIDER: I would say the answer to that is no. There is -- there is work going on now in the risk informed arena which I think is taking into account more looking at what areas as susceptible and what the consequences are. But I think when some of the early code inspection requirements were developed, it was largely just go out and do a sample across the system. For example, look at 25 percent of the reactor cooling system welds, class one welds, pick those and that should be an adequate sample to tell us if there are any problems.

CHAIRMAN JACKSON: Commissioner McGaffigan?

COMMISSIONER McGAFFIGAN: Could I ask, why wouldn't sampling work in this instance, when their probabilities are ten to the minus fortieth, I haven't seen those since I was studying neutrino cross-sections some time ago.

CHAIRMAN JACKSON: Yeah, we know about those.

COMMISSIONER McGAFFIGAN: Which are small.

But why would -- they are proposing no testing of or inspection of the circumferential welds but why -- why wouldn't a sampling technique be adequate?

MR. STROSNIDER: It is a good question. It is one that we have considered. I will get to that, but I will give you a little preview, which is basically that reactor pressure vessels and the reactor pressure vessel welds are not all the same. Okay? You have to realize that there was a discussion about the sort of inspection that was done during fabrication of the vessels. However, that inspection was different, whether it was radiography or surface, in some cases ultrasonic. It changed as the code changed in time. So not all vessels saw the same fabrication inspections.

The welds made in the vessels because of the fabrication process are different. For example, there was a question earlier about are the circumferential welds different than the axial welds. When you look at the process for fabricating these vessels, the ring sections are made up of plates and there is an automatic process once the ring section is laying down the cladding, welding process. Then the rings are welded together and, in most cases, the back cladding as it is called, the cladding over the welds that join the ring sections together, were done manually. So there is a difference.

In the manual welds, what we have seen is that they are not controlled as well, the heat input may be more difficult to control and those may be areas that are more susceptible to degradation. Also, some of the issue that comes up is repair. There have been and it was indicated repairs were made during fabrication.

There are a number of different vendors or shops that were involved in fabricating these vessels. At least four. Some of the vessels actually went through one, two or in one case three of those shops during fabrication. The vessel was partially fabricated, shipped to another vendor for additional fabrication and shipped to another one to be finaled.

So there is a question about whether the welds we are looking at really represent a homogeneous statistical population, to which you could apply sampling. And one of our concerns is that where repair welds may have been made, that those are particular areas that ought to be looked at. And we think the best way to catch that is by doing a one time base line examination.

You know, we have to keep that in perspective. We do not expect that there are significant, huge flaws in these reactor vessels or I would be here taking an even more aggressive decision on this. But we do recognize from the evaluations that have been done that there is the potential that the wrong -- the wrong elements could wind up in the same place. It is a low probability. But we believe that it is appropriate to go confirm the assumptions that are in the analysis and the evaluations to make sure it really is as low as we think it is.

The situation we are talking about, and even in the industry's assessment, they talk about the potential for stress corrosion cracking in the cladding, lining up with some pre -- some fabrication defect that is in the underlying base metal. And perhaps if you go on beyond that and say, well, this was the area of a large repair, was the stress relief, post-repair stress relief effective, what kind of environment are you in in a particular plant? If you add all those up in the wrong place, you might have the potential for a viable degradation mechanism. And a large part of our conclusion is we ought to verify that that doesn't exist out there.

CHAIRMAN JACKSON: Commissioner Diaz?

COMMISSIONER DIAZ: Yes, just in the same vein, wouldn't a 100 percent examination of the longitudinal welds provide you with a very reasonable sample of how the pressure vessel is standing up?

MR. STROSNIDER: What I am suggesting is that the circumferential welds and the axial welds are not necessarily the same population of welds because of differences in fabrication.

COMMISSIONER DIAZ: I know, but that is not the question. The question is, wouldn't a 100 percent examination of longitudinal welds give you a very good program to verify at least, you know, a portion of the industry's analysis?

MR. STROSNIDER: I am sure you could make some statistical inferences from that if you understood how many repair welds were in that sample versus how many repair welds are in the circumferential welds, things of that nature.

CHAIRMAN JACKSON: Are you saying that is not known?

MR. STROSNIDER: I would say, number one, it hasn't been analyzed. It would take a tremendous amount of effort to pull out all those records. We also -- one of the bullets on the next viewgraph talks about the concern for undocumented repairs.

I would point out that what we have also concluded is following an initial base line to verify the condition of the vessels that a sampling program may in fact be appropriate depending upon the results of that base line example.

COMMISSIONER DIAZ: Define a base line.

MR. THOMPSON: Our definition was essentially a 100 percent of accessible. Essentially 100 percent.

MR. STROSNIDER: Let's move on to viewgraph number six and some of this I think I may have already covered in response to questions.

I want to point out that inspections have identified degradation in reactor pressure vessels and these are some of the instances that, in fact, were called out in the backfit analysis that supported promulgation of the rule.

At Hatch One, there was some pre-service ultrasonic testing done. This was actually in the industry report, which identified defects in the recirculation to shell weld nozzles that required repair so they had to be ground out and repaired.

COMMISSIONER DIAZ: I'm sorry, I couldn't hear you.

CHAIRMAN JACKSON: Hatch One.

MR. STROSNIDER: Yes, at Hatch One during fabrication inspections, ultrasonic testing did identify defects in the recirculation nozzle to shell weld that exceeded -- from what I can read it exceeded the code acceptance criteria and required repair. So there were defects in some of these vessels during fabrication. There were repairs made. And there were varying degrees of inspection.

COMMISSIONER DIAZ: But a nozzle is always a high stress point so it is not the same as the rest of the vessel.

MR. STROSNIDER: True, but this was not service induced. This was fabrication. And it may be a more difficult spot to weld, that's true.

The state of the art inspection methods have identified indications requiring code evaluation. I have heard mention of Brown's Ferry did inspections in 1993. They were using state of the art inspection methods. Fifteen indications required evaluation by code. They would not have been evaluated under the old inspection procedures but they were under the new, detected and evaluated under the new procedures. They were found acceptable; they were subsurface.

In 1995, Pilgrim also performed a state of the art inspection. They found no indications requiring flaw evaluation and this is the information we have available. I wanted to point that one out because in terms of the reactor vessels being similar and there are differences, these were in fact made by different vendors, different results from the inspections.

With regard to viable degradation mechanisms existing, first, it is a given that the BWR environment is an aggressive environment. It can support crack growth. Certainly in stainless steel, we have seen this in piping and internals. Ferritic, as was indicated, some of the early data show that stress corrosion could be supported in some of the ferritic base metal. Some of the more recent data says no, there is some mixed results on that.

With regard to actual experience, there was a mention of the Quad Cities Unit Two, indications that were found in 1990. These were not in a shell weld, they were in the flange, the head weld. There were 34 surface flaws found during that inspection. The longest one was 30 inches long. It penetrated, at its deepest point, through the cladding and about two-tenths of an inch into the heat effective zone in the base metal. So about seven-tenths of an inch deep.

CHAIRMAN JACKSON: Is there a difference between the, you know, are there sufficient differences between the construction of the reactor vessel head and the reactor pressure vessel itself to make the head more susceptible to these degradation mechanisms?

MR. STROSNIDER: Using the same welding processes, there may be some difference, perhaps, in how easy the fit- up is and I can't say there is anything particularly or -- I don't know, staff is shaking their head no difference.

I can't really add anything beyond that.

COMMISSIONER DIAZ: The environment is not the same.

MR. STROSNIDER: No.

COMMISSIONER DIAZ: There is a different environment in the head.

MR. STROSNIDER: There is a different environment. That is certainly true, in that you are in a steam environment in the head.

I just comment, we got into looking at differences in environments on the core shroud where we thought all the cracking was going to be up high because of the more aggressive environment and it didn't turn out that way.

What you have to remember is you have a lot of competing parameters in developing and sustaining cracking and it includes the environment, it includes the stresses, it includes the

material properties and it -- you have to be careful in trying to assume you know how all those are going to come together.

So that was the experience at Quad Cities. It was evaluated that that flaw was found that it was acceptable as it was found. There was some grinding done on it to smooth it out and then it was found acceptable for continued service. But the grinding, of course, reduces the stresses there and makes it less susceptible to any continued growth.

The backfit package that went along with the rule in 1992 referenced some experience with stress corrosion cracking in feedwater nozzles siphons where again cracking was initiated in stainless steel but grew into the ferritic material. It occurred at Brunswick and also at a Chinese plant.

Finally, this one was interesting, Fitzpatrick, this was also I believe in 1990. They found a surface crack in the reactor vessel head. This was higher in the head than at the flange weld. Interesting. This was an unclad head. There was no stainless steel cladding on this vessel.

When they went back and took a close look at this, it turned out that the surface indication that was there was some sort of fabrication scratch or defect. It happened to be in the area of some subsurface slag inclusions that were about 12 inches in length. The maximum depth at that location was about two inches.

Those appear to have been fabrication, not service induced defects but one of the things that we heard and that we have been considering is what's the likelihood that the wrong situations could add up at the same time. This is in a location where, in all likelihood, had it been clad it would have been done manually and those are areas where we know there is a greater susceptibility to stress corrosion cracking of the clad and if that sort of crack joined up with this sort of preexisting defect, it might be a concern.

As you heard, the analysis does make an assumption, okay, that in fact if you grow through the clad, you sample from a distribution and have that match up with some fabrication defects. One thing I point out here to recognize is a lot of the Monte Carlo analysis is often assumes independence of all these different parameters. In this case, they have tried to address that but I think the point is there may not be independence because some areas are just more susceptible to having these adverse conditions.

COMMISSIONER DIAZ: May I make a comment?

CHAIRMAN JACKSON: Please.

COMMISSIONER DIAZ: You know, this is not my area. I am here, you know, apples and oranges. You are mixing flanges and heads that are carbon steel that are not, you know, stainless steel with defects from manufacture and putting all that together in the context of the reactor pressure vessel. And I don't think they are the same thing, you know, from the little of what I know. I think they are completely different issues.

I mean, we know that there is a stress corrosion cracking issue with boiling water reactors and we have always known that. They have taken care of that.

Now, the question is, have we ever found a deficiency or degradation in a reactor pressure vessel, in a boiling water sufficient to say, hey, this is not acceptable and you have to do something about it? Have we ever found one?

MR. STROSNIDER: I am describing what has been found and the inspections that were performed.

COMMISSIONER DIAZ: No, you have not said that there is one that has actually been significant to the point that it is not acceptable to the staff or, at least, that is what I heard.

MR. STROSNIDER: That's correct.

COMMISSIONER DIAZ: So all of them have been acceptable to the staff so the staff concluded that they did not really degrade to the point that it posed a safety question; is that correct?

MR. STROSNIDER: That is absolutely true and as I indicated earlier, that is our expectation. I hope that we never find and I don't think we will find flaws in a reactor vessel that compromise its integrity.

COMMISSIONER DIAZ: The if is not the issue. The question is, have you found one and I guess your answer is no.

MR. STROSNIDER: No, we have not found one.

COMMISSIONER DIAZ: Thank you.

MR. COLLINS: Commissioner, I guess it is important to know that I think part of what Jack is trying to stress is because we have not done the 100 percent examinations we have not established a base line which would indicate what the potential is for that to occur other than an in-process issue, which would be a leak. And, of course, that has been avoided.

COMMISSIONER DIAZ: I understand the difference.

COMMISSIONER ROGERS: Just before you leave that, though, it does seem to me that you have -- you do have a total disagreement with the industry on this question of whether there is a viable degradation mechanism for welds. I mean, you have cited a number of examples of degradations that you have found but I didn't hear you mention any in a weld.

Their statement, their concluding statement was, an absence of degradation mechanisms substantiates vessel integrity, dot, dot, dot. And you are saying there is a viable degradation mechanism and so it seems to me there is a total conflict on that issue.

MR. STROSNIDER: Yes, and the real issue here, first of all, there is a degradation mechanism which everyone acknowledges in the stainless steel cladding. There are cracks that have been found, service induced in the cladding. The question is, will it grow into the ferritic base metal, all right? And as I indicated, and I think as was indicated in their presentation, some of the early data indicate that you could grow cracks if you have a high enough driving force. Some of the more recent data says, no, you wouldn't expect that.

All right.

We have not seen an example where it has really been given a chance. Probably the closest was quad cities. That was found early in the inspection and the defect was corrected. The analysis that the industry did did suggest that if you had cladding flaws growing into significant fabrication defects where you get a high enough driving force, something like 30 KSI root inch applied stress intensity factor that there could be a mechanism.

So, as I indicated, the data are not all that clear, all right? And given that uncertainty, our conclusion is that we should go take a look.

The last thing on this viewgraph I wanted to talk about was the potential for undocumented repairs. I am not sure how much difference it makes whether they are documented or not in terms of the potential for degradation although, as was said, there was a lot of work done, a lot of procedures in place to document this sort of thing.

However, the research office says the reactor vessel down at Oak Ridge National Laboratory which we have been doing examinations on, looking at welds, looking at density of defects and that sort of thing. And one of the things they found in that reactor vessel was a significant repair to one of the shell welds which was not documented. It was not in the documentation that we acquired with the vessel. I don't know if Mike wants to expand on that at all but --

MR. MAYFIELD: Just that it turned out to be a quite large defect or repair, in some cases according to the laboratory running as much as three-quarters of the way through the wall thickness. It spanned several feet. The only indication in any of the documentation is that there were -- there was a repair based on high-low mismatch that you get when you line up the two rings but there was certainly no suggestion of the extent of this repair in any of the documentation that we acquired.

COMMISSIONER ROGERS: Would that have been done at the time of fabrication?

MR. MAYFIELD: Yes, sir.

COMMISSIONER DIAZ: Of course, repairs are part of the industrial process.

MR. MAYFIELD: Yes. And, in and of itself, we weren't bothered by it. It is just that it is one more bit of information that feeds into this puzzle.

CHAIRMAN JACKSON: Okay. Let's move on.

MR. STROSNIDER: Moving on to viewgraph number seven, again, the need for inspections, the conclusion that we reached here is that we think again a base line inspection, which I will define as essentially all the welds they can get access to and take a look at is appropriate in order to verify the low probabilities that we are seeing.

As I said earlier, you are not going to hear a condemnation, general condemnation of the analysis that was done by the industry. We think it had a lot of insights and that there is a lot of merit to it but we do think there are enough questions, looking back at the history, that it is appropriate to go do that sort of base line examination.

What we are looking for is what we consider a very low probability event. But we are talking about the reactor pressure vessel and we feel that the safety significance of the vessel warrants doing that sort of inspection.

Having done that, we do think that the analysis that has been presented, after we look at the results of that base line, provide perhaps good basis for going through a sampling inspection and that could mean significant impact on the resources expended in subsequent intervals.

Going on to slide number six, just a discussion on the NRC and the ASME code inspection philosophy. You heard some of this. Basically, the code has evolved over time. It currently does require 100 percent inspection, essentially 100 percent inspection, which means 90 percent recognizing some of the limitations. Anything less than 90 percent requires actually some granting of relief or alternative by the NRC under 5055(a).

I should point out that some of the NRC certainly was a proponent in some of these code changes that went to larger examination percentages. But our position has been consistent with the ASME code for some time which, actually, since 1975 has required at least 100 percent base line examination. Essentially 100 percent.

You heard that the industry is pursuing with the ASME codes some changes in these requirements. In fact, we encourage that in one of our letters, particularly with regard to those inspections that might be performed subsequent to a base line.

CHAIRMAN JACKSON: Is that to say then that if the code is changed, the staff will change its position?

MR. STROSNIDER: No.

But we will certainly assess the changes in the code and see through our rulemaking process if that is the appropriate answer. And, as I said, we have encouraged after a base line examination the notion that the evaluations performed support a sampling sort of inspection.

CHAIRMAN JACKSON: Where in the process is the BWR owner's group in its request to change the code? I mean, how far along?

MR. STROSNIDER: As Mr. Dyle indicated, it has been through several committees. I am not sure I can give you all the way up through the subcommittees.

CHAIRMAN JACKSON: I mean, how much longer do you think this is going to take? Is it hard to predict?

MR. STROSNIDER: I don't know. Is there someone who was at the code meetings from the staff that can address that?

Gil Millman?

MR. MILLMAN: Pardon my laryngitis; I have been at code meetings for the last week.

This particular code case did come up to the Subcommittee on Nuclear In-Service Inspection last December. At that time, Mr. Dyle withdrew it and on the basis that it would go forward only when there was a technical basis document supporting it and so it waits at the subcommittee for that action.

CHAIRMAN JACKSON: I see.

Commissioner Diaz?

COMMISSIONER DIAZ: I don't know whether the question is valid any more but you said no to whether this type of change in the position, you know, regarding the ASME. Does that mean the staff's position is independent of the ASME?

MR. STROSNIDER: Well, in general.

COMMISSIONER DIAZ: In total?

MR. STROSNIDER: In general, the process that we go by is the Code of Federal Regulations endorse industry codes and standards. Sometimes we endorse those with some exceptions or with some additions and my comment is basically that we will not only observe but we have people who will participate in the code activities and make sure that our concerns are identified early.

When the code reaches conclusion, either in a code case or in a change to the code, we will assess that as part of the rulemaking process and see how it would be endorsed in the regulations.

But we don't -- it is not a given that we just take it the way it's --

CHAIRMAN JACKSON: Have there been cases where the staff -- the staff's position has not been consistent with the code and the staff has come out with a more conservative position?

MR. STROSNIDER: Yes.

MR. COLLINS: Yes.

CHAIRMAN JACKSON: Okay.

MR. STROSNIDER: One other comment is we did -- we went out last week basically a poll looking to see what the positions are internationally with regard to this type of inspection. We have three responses so far, one from MITI, the Ministry of Industry and Trade in Japan. They require 100 percent each 10 years, every 10-year interval --

CHAIRMAN JACKSON: Of vertical --

MR. STROSNIDER: Of the shell welds.

CHAIRMAN JACKSON: Of all of them?

MR. STROSNIDER: Yes, longitudinal and circumferential.

COMMISSIONER ROGERS: BWRs as well as PWRs?

MR. STROSNIDER: Yes.

We do understand also that there is some discussion with their industry about possibly changing that at some point.

The Spanish do 100 percent of axial and circumferential each 10 years and also in Sweden they do 100 percent.

I would point out that a lot of this is driven by what is in the ASME code and that is an international code so there are other countries who follow that and in fact do follow pretty much what the NRC is doing.

I would also point out, though, that Sweden has been leading, perhaps, in the area of risk-informed in- service inspection and they still do this sort of inspection.

Viewgraph nine, talking about granting relief and I think the main point I wanted to make here is that we recognize that certainly with the current tooling there are some limitations as to what can be inspected.

In the industry submittal, they talk about, however, some of the improvements that have been made and they talk about an inspection in 1983 at a BWR 3 facility where they were able to get 41 percent of the circumferential welds and 52 percent of the longitudinal. In a more recent 1993 examination, this was at a BWR 4 so there might be some slight differences, but they achieved 78 percent of the circumferential welds and 91 percent of longitudinal. So there has been progress in terms of the tooling and the technology.

You also heard mention the device that has been demonstrated at the EPRI NDE center that was developed by the Japanese. You understand there is at least one U.S. company looking at commercializing that in the U.S. and it is basically a submersible device which is, as I understand, self-propelled and can move around. It is very thin. The word we got is it could get probably 90 percent of the welds in most of the vessels out there. I don't know how far that is from actual implementation. We know there have been demonstrations at the NDE center and they are ongoing in Japan.

I think the point here is that progress can be made in terms of improving the inspection technology. And some of this, again, we haven't seen all the details but it sounds like it would have reduced setup time and even personnel exposure as opposed to putting big manipulators on top of the vessel, being able to put in some submersible which you can operate from some distance.

COMMISSIONER DIAZ: That is not commercially available in this country. Will it be in the next five years?

MR. STROSNIDER: Not right now, no. And I don't know. Like I said, the industry is following that. As Mr. Dyle indicated, they are aware of it.

COMMISSIONER DIAZ: In other words, it is a long term thing. It is not something that is going to be available next year?

MR. STROSNIDER: I don't know what the schedule is. As I said, it has been demonstrated and is -- there are some in-vessel demonstrations going on in Japan.

CHAIRMAN JACKSON: I've seen it. EPRI is working on it.

MR. STROSNIDER: So with regard to granting reliefs and, as I pointed out, the rule does -- and it specifically included, and I am looking at slide number 10 now --

CHAIRMAN JACKSON: Let me go back to slide nine. You say the industry proposal is for NRC to grant a large number of reliefs from requirements based largely on probabilistic assessments and I note that in your paper, the Staff stated that it had concluded that rejection of the project's probabilistic arguments to support authorization of inspection alternatives, et cetera, is consistent with the Commission policy on the use of probabilistic risk assessment.

Can you explain, you know, the basis of that statement and is the staff's current position risk informed and can you relate that to ongoing efforts with respect to a risk-informed ISI and IST, okay?

MR. STROSNIDER: A statement that was in the Commission policy, let me see if I can actually get the -- well, this I can just read. This was a quote from the Commission policy statement that use of probabilistic risk assessment methods, the staff used the safety goals in making regulatory decisions regarding backfitting new generic requirements but not to make specific licensing decisions including granting relief from unnecessary requirements.

That is a quote from the policy statement.

MR. COLLINS: It is on page 4.

MR. STROSNIDER: August 19, 1995. I was looking for the policy statement but it is in the paper.

But I guess I would also point out that, to try to keep this in context, the evaluation that was submitted by the industry is really not full-blown risk assessment. It doesn't go out to the consequence stage administration that sort of thing. It doesn't assess what happens if you have a leak, for example, and it does include some deterministic arguments with regard to fabrication and that sort of thing. So it is sort of a mix.

But we thought that was an issue that we at least questioned when we looked at it and said, well, is this an appropriate basis for granting release and it would be release for essentially all the BWR plants. Does it maintain defense in depth as we think is appropriate?

CHAIRMAN JACKSON: Commissioner McGaffigan?

COMMISSIONER McGAFFIGAN: The difference is 30 orders of magnitude between longitudinal welds and circumferential welds and in their analysis. You have gone through a long explanation as to why there might be something there that no one has foreseen and therefore you want to inspect them all but 30 orders of magnitude, have you looked at that difference and that analysis and found a flaws in it?

MR. STROSNIDER: There are no specific problems that we have identified in the way the analysis -- in the modeling itself. It has to do with looking at assumptions, input parameters and, quite frankly, our experience in trying to predict what may or may not happen. I refer back to some probabilistic assessments on piping and that sort of thing where people failed to take into account loadings and they found degradation. They weren't in the model.

So one of the reasons you do inspections is to find out what you are not smart enough to put in your model.

As I said, you are not going to hear a condemnation of the analysis that they have done and it does show a significant difference.

MR. THOMPSON: Commissioner, to get to your point, as Jack explained, we are dealing primarily with the up- front assumptions that you predicate that risk questions on and the uncertainties that are involved as articulated by the staff here with the fabrications and the records and the history and the repairs and the lack of a base line. Lacing that base line, the staff really is missing a key piece of information to predicate the change under 5055(a) which is allowable if you are able to meet the statement of an acceptable level of quality and equivalent acceptable level of quality and safety. That is essentially where we are.

MR. STROSNIDER: On viewgraph number 10, I just briefly indicate that, as I said earlier, that the rule acknowledged when it was promulgated that there could be some areas that are difficult to access and in fact the wording in the augmented inspection rules where people are unable to do inspections, they may propose alternatives.

Quite frankly, it takes a little bit of thinking but it is our assessment of the industry's proposal, we think, that proposal can be used to justify some of these areas where you just can't access them. But we also think that in terms of defense in depth that you should do the scope of the inspection that you can do.

So, the conclusions are that we -- again, we think the industry's analysis has merit. It has added a lot of insights to pressure vessel integrity issues. We have concluded for the reasons we just discussed that a base line examination of those welds that can be accessed should be performed. That the report and the work they've done can be used to support relief where, in fact, they just can't access some of these welds and that future modifications to the inspection requirements may be appropriate after completion of the base line.

It would be our plans to complete that in a safety evaluation that we could issue in probably about six weeks or so.

CHAIRMAN JACKSON: Let me ask you three questions quickly. If uncertainty isn't in part influencing the staff's position, are there alternatives such as pilots or targeted implementation or some other strategy to provide some additional information to support the staff's position?

MR. STROSNIDER: Well, I think the question came up. One obvious thought that comes up there is could you deal with this on a sampling basis and draw inferences from the sampling basis. And --

CHAIRMAN JACKSON: That is one example. But one could take a -- and I guess this is a different -- it depends on what you mean by sample. You could take all the plants and have a sample of areas. You can take a subset of plants and do 100 percent. That's a sample. Et cetera, et cetera.

Have you given some thought to these kinds of alternatives?

MR. STROSNIDER: Well, we thought about that and, again, the conclusion we reached was do as much as you can at this point and then look at a sampling basis because after you have gone through and looked at all the welds and confirmed the -- you know, really given confirmation of the quality that was there when they were originally fabricated and, as we pointed out, there have been improvements in inspection techniques, we can see things today we couldn't see then, you have confirmed that in fact you don't have all the wrong conditions at the same location, you have confirmed that there is something you didn't anticipate, then you basically we think can go to a sampling method where you are monitoring for any sort of degradation that might show up.

CHAIRMAN JACKSON: So basically you want a database which you believe you don't have at this stage of the game, is that the point?

MR. STROSNIDER: Yes.

CHAIRMAN JACKSON: Have you discussed this at all with the ACRS?

MR. STROSNIDER: We have not had any recent discussions. The ACRS was involved in the original promulgation of the rule back in '92. They looked at that and supported it, as I understand it.

CHAIRMAN JACKSON: Do you intend to document the technical basis for your rejection of the industry group's proposal then in a safety evaluation report?

MR. STROSNIDER: Right. We would document the discussion basically that I just gave you and a safety evaluation which I would expect to complete in about six weeks.

CHAIRMAN JACKSON: And what kind of time line are we operating under?

MR. STROSNIDER: Well for, as I say, issuing the safety evaluation, I would put a target of about six weeks.

It is important, and I think the industry pointed out, when you look at the rule and where the plants are in their inspection intervals, that many of these examinations would need to be performed in the next year or two and the planning has to be done, equipment has to be available. So we recognize that a decision of position needs to be made sooner rather than later.

CHAIRMAN JACKSON: Okay, is that it?

MR. THOMPSON: That concludes our presentation. We would be prepared to answer any questions.

CHAIRMAN JACKSON: Commissioner Dicus, questions, Commissioner Diaz?

COMMISSIONER DIAZ: Yes, I just have a quick comment. Knowing the difference between these reactors and the difference between circumferential and longitudinal welds, I actually don't see, although you might have it in six weeks, a basis for denial of the industry request. It seems to me like 100 percent longitudinal inspection program with some beef behind it, I mean, to get it done in a very, you know, reasonable period of time will provide a good base line. And from there, during that period of time, we might be able to develop a program that will provide some basis for the circumferential welds.

I actually see no technical information that has been presented that says this is, you know, unreasonable or is not adequate protection of health and safety. Because most of the things that have been presented are peripheral to the main issue of how the pressure vessel is attacked and how are the -- you know, the differences in stresses between circumferential and longitudinal welds.

So unless I see something different, I don't see why a program that actually addresses 100 percent longitudinal wells as soon as possible, will not be a good base line to consider, you know, than the circumferential welds.

CHAIRMAN JACKSON: I would like to thank the representatives of the BWR Vessel and Internals Project and the NRC staff for briefing the Commission regarding the issues associated with the staff's technical position regarding alternatives for augmenting inspection of the reactor vessel. As I mentioned in my opening remarks, you know, the Commission is not a commission of technical experts and so, I don't believe in an hour and a half we can sit here and sort through all of that. It is important for the Commission to understand aspects of the technical basis for the staff's position so that if there are any policy issues involved, the Commission can make informed decisions.

It is also important for the public and the industry and as well, as the discussion today has revealed, the international regulatory community to understand the staff's positions. So given the recognition of the important role that the reactor pressure vessel does play in implementing the Commission's defense in depth philosophy but given that you have even said yourself that the project has proposed some technically sound discussions for implementing a reduced scope augmented inspection, the staff should complete, on an expedited basis, the development of the safety evaluation report on the Boiling Water Reactor Vessel and Internal Project proposed alternative and to consider whether there is a tiered approach to getting at the issue. And if it is not technically possible, you should tell us that.

This safety evaluation report would then serve as the staff's documented and defensible basis for resolution of the issues and any -- document any open issues and would facilitate any commission decisions if they are appropriate on any of the related policy issues.

So unless there are any further comments, we are adjourned.

[Whereupon, at 4:50 p.m., the meeting was concluded.]

NRC STAFF'S POSITION ON AUGMENTED EXAMINATION REQUIREMENTS FOR BOILING WATER REACTOR PRESSURE VESSELS PURSUANT TO 10 CFR 50.55a(g)(6)(ii)(A)

May 12, 1997

E/4

Jack R. Strosnider, NRR

SUBJECTS FOR DISCUSSION

- SAFETY SIGNIFICANCE
- AUGMENTED INSPECTION RULE
- NEED FOR INSPECTIONS
- NRC/ASME INSPECTION PHILOSOPHIES
- RELIEF PROVISIONS
- CONCLUSIONS

SAFETY SIGNIFICANCE

- RPV Integrity Must be Maintained to the Highest Level of Quality
- Engineered Safety Features are Not Designed to Cope with RPV Failure
- Defense-in-Depth Maintained by Inspections and Evaluation

10 CFR 50.55a AUGMENTED INSPECTION RULE

- Required Expedited Implementation of Augmented Inspections of RPV Shell Welds

- Revoked Industry-Requested Reliefs from Inspection Requirements Previously Granted

- Incorporated 1989 Edition of ASME Code, which Required Inspection of All RPV Shell Welds

NEED FOR INSPECTIONS

- To Ensure Quality of Components by Monitoring for Unanticipated Degradation and Assessing Significance of Defects

- Prediction of Degradation in Other Components has Not Been Highly Reliable

  - BWR Piping and Internals

McC. Question about why sample is insufficient

NEED FOR INSPECTIONS (Con't.)

- **Inspections Have Identified Degradation in RPVs**

Hatch 1 Reconnaissance

  - **State-of-the-Art Inspection Methods Have Identified Indications Requiring Code Evaluation**

Browns Ferry 3 Pilgrim

  - **Viable Degradation Mechanism for RPV Welds Does Exist**

Quad Cities, [illegible]

  - **Potential Exists for Undocumented Repairs**

NEED FOR INSPECTIONS (Con't.)

- Inspections Support Analytic Assumptions and Identify Potential Unexpected Degradation Phenomena
  - Validate Flaw Distribution Assumptions
  - Validate Assumptions Regarding Degradation Mechanisms

NRC / ASME CODE INSPECTION PHILOSOPHY

- ASME Code has Required *at Least One "Essentially 100 Percent"* RPV Weld Inspection Since 1975
- 1989 Edition of ASME Code Requires Essentially 100% Examination of Beltline Welds Every Inspection Interval
- NRC Staff's Position Consistent with ASME Code
- BWROG Pursuing Changes to ASME Code
  - ASME has Not Approved Changes

GRANTING RELIEFS FROM THE REGULATIONS

- 10 CFR 50.55a Incorporates Mechanisms for Granting Reliefs from Rule Where Alternatives Can be Shown to Provide an Acceptable Level of Quality and Safety

- Industry Proposal is for NRC to Grant Large Number of Reliefs from Requirements of 50.55a Based Largely on Probabilistic Assessments

GRANTING RELIEFS FROM THE REGULATIONS (Con't.)

- **Staff Position**

  - Technical Bases Insufficient to Support Eliminating Baseline Examinations

  - Limited Reliefs May be Necessary Where Licensees are Unable to Perform Examinations at Inaccessible Locations

  - Changes in Inspection Scope May be Appropriate After Base Line Examinations Completed

NRC STAFF'S CONCLUSIONS

- BWR Licensees Should Complete at Least One Examination of All RPV Beltline Welds Capable of Being Inspected to Validate Analysis Assumptions

- BWRVIP-05 Report Can Be Basis for Granting Limited Alternatives Under 10 CFR 50.55a(g)(6)(ii)(A)(5)

- Future Modifications to Inspection Requirements May Be Appropriate After Completion of Baseline Examinations

Structural Integrity Associates, Inc.

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May 21, 1997
SST-97-001

Mr. Gene Carpenter
U. S. Nuclear Regulatory Commission
11555 Rockville Commission
Rockville, MD 20852-2738

Subject: Transmittal of VIPER source code

Dear Gene:

Enclosed diskette contains the source code for the VIPER software.

Note on the software:

Program:	VIPER
Version:	1.0
Date:	June 1996
Hardware Plateform:	Intel X86 CPU or compatible
Operating System:	MS_DOS Version 6.XX
Language:	C
Compiler:	Microsoft C Compiler Version 6.0

A makefile to compile the program is included in the diskette. To compile the program, type
nmake viperv1.exe

The program uses file 'viper.inp' as the input file. The output is under the filename 'viper.out'. To run the program, type 'viperv1'.

If you have any questions, please call me at 408-978-8200 (e-mail stang@structint.com).

Very truly yours,

Stan S. Tang, P. E.
Consultant

gsv
cc: P. C. Riccardella
EPRI-88-102

E15

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F/6

97-477

BWRVIP

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

June 3, 1997

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

Attention: C. E. Carpenter

Subject: BWRVIP Vessel Inspection Program Evaluation for Reliability (VIPER) Software and Users Manual

Enclosed is one copy of the BWRVIP Vessel Inspection Program Evaluation for Reliability (VIPER) Software and Users Manual. The enclosed software and users manual is being submitted to the NRC for information only and as a means of exchanging information with the NRC for the purpose of supporting generic regulatory improvements related to inspections of BWR reactor pressure vessel shell welds.

The enclosed software and users manual is for a computer code developed for analysis of reactor pressure vessel fracture mechanics as documented in "BWR Vessel and Internals Project, BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations (BWRVIP-05)."

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

If you have any questions on this subject please call Warren Bilanin of EPRI at (415) 855-2340.

Sincerely,

CTerry

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

DF03 Add: Paper Copy
1/1 Timir - Misra

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

07

Attachment 1

EPRI
Electric Power
Research Institute _Powering Progress through Innovative Solutions_

May 30, 1997

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

Subject: Request for Withholding of Proprietary Software; 10CFR2.790(a)(4) "BWRVIP Vessel Inspection Program Evaluation for Reliability (VIPER) Software and Users Manual"

Dear Sir/Madam:

This a request under 10CFR2.790(a)(4) that the NRC withhold from public disclosure the proprietary software identified above (the "Software"). One copy of the Software and the affidavit in support of this request are enclosed.

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

The Software is for the NRC's internal use and may be used only for the purpose for which it is disclosed by EPRI. The Software 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 Software should be directed to Warren Bilanin of EPRI at (415) 855-2340.

Sincerely,

Art Kenny

Arthur Kenny
Intellectual Property Attorney
Intellectual Property Department

Enclosures

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

RE: BWRVIP Vessel Inspection Program Evaluation for Reliability (VIPER) Software and Users Manual

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 software and users manual listed above that is sought under this affidavit to be withheld (the "Software") 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 Software consists of trade secrets of EPRI and that the NRC will receive the Software from EPRI under privilege and in confidence.

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

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

(ii) The Software is of a type customarily held in confidence by EPRI and there is a rational basis therefor. The Software is trade secrets and is held in confidence by EPRI because to disclose it would prevent EPRI from licensing the Software 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 Software, 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 Software. The rational basis that EPRI has for classifying the Software 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 Software will be transmitted to the NRC in confidence.

(iv) The Software is not available in public sources. EPRI developed the Software only after making a determination that the Software was not available from public sources. It required a large expenditure of dollars for EPRI to develop the Software. 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 Software, show that the Software is highly valuable to EPRI. Finally, the Software was developed only after a long period of effort of at least several months.

(v) A public disclosure of the Software would cause substantial harm to EPRI's competitive position and the ability of EPRI to license the Software both domestically and internationally. The Software 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:

May 30, 1997

Arthur Kenny, Esq.

Subscribed and sworn before me this day: May 30, 1997

Tamsen Helen Gagnon Notary Public

Tamsen H. Gagnon, Notary Public

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

STAFF EVALUATION OF INDUSTRY PROPOSAL TO ELIMINATE INSPECTION OF BWR RPV CIRCUMFERENTIAL WELDS

M. Mayfield
E. Hackett
L. Shao

August 5, 1997

STAFF EVALUATION OF BWRVIP-05 REPORT

<u>Background</u>

- 10 CFR 50.55a modified in 8/92 to require inspection of RPV welds in accordance with ASME Section XI on expedites schedule -- revoked all previous reliefs

- BWR Owners' Group sponsors the BWR Vessel and Internals Provject (BWRVIP) technical committee

  - BWRVIP prepared a report (BWRVIP-05) using probabilistic methods to demonstrate that inspection of the circumferential welds was not warranted -- probability of failure too low for concern

- Staff concluded that no alternative to the required inspections should be granted until at least one inspection of beltline welds was completed

  - Some BWR RPV's have never been inspected during service
  - Preservice inspection techniques used at time of vessel fabrication are now known to have low Probability of Detection (POD)

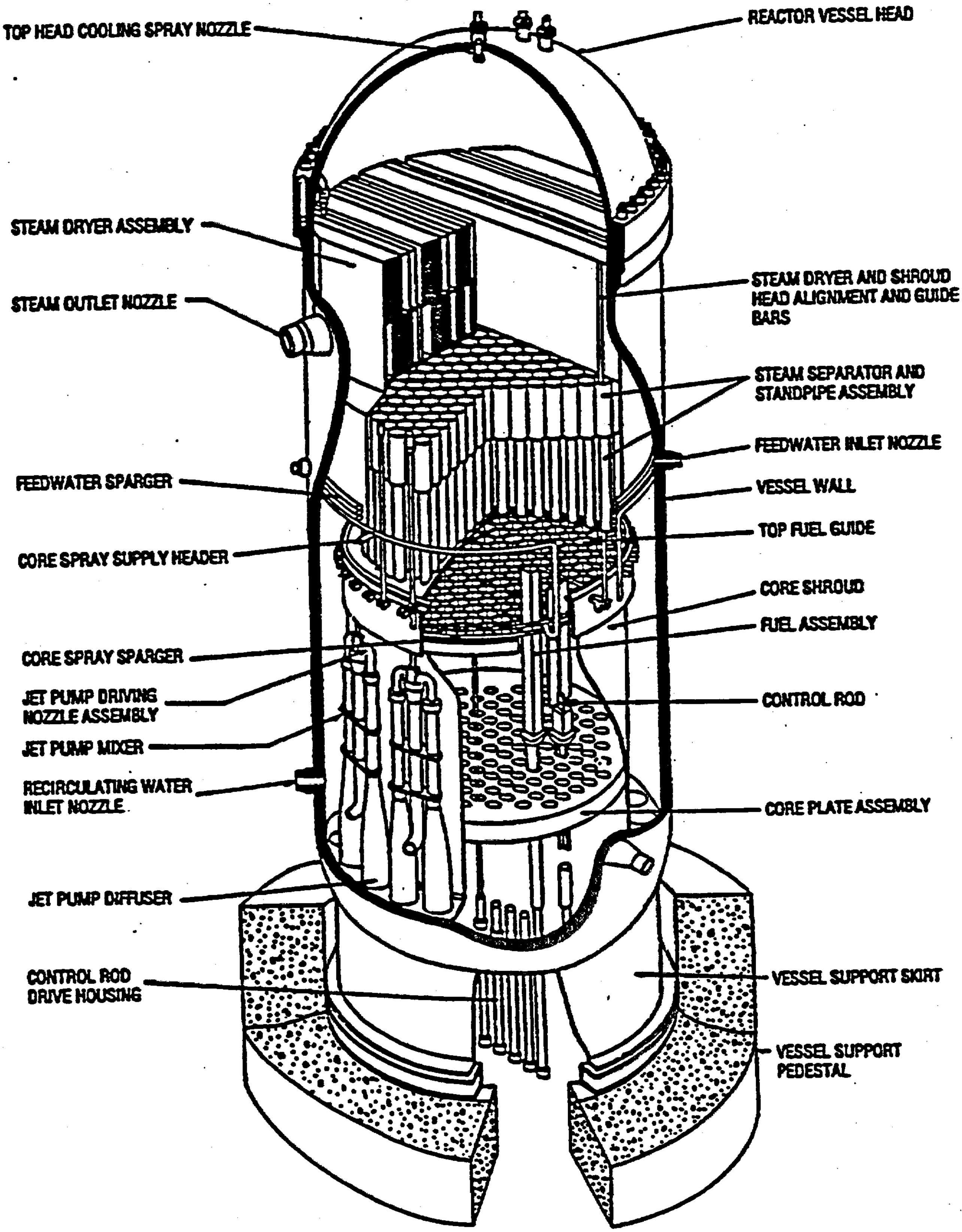


Figure 2.2 General Arrangement of a Boiling Water Reactor Vessel and Internal Components

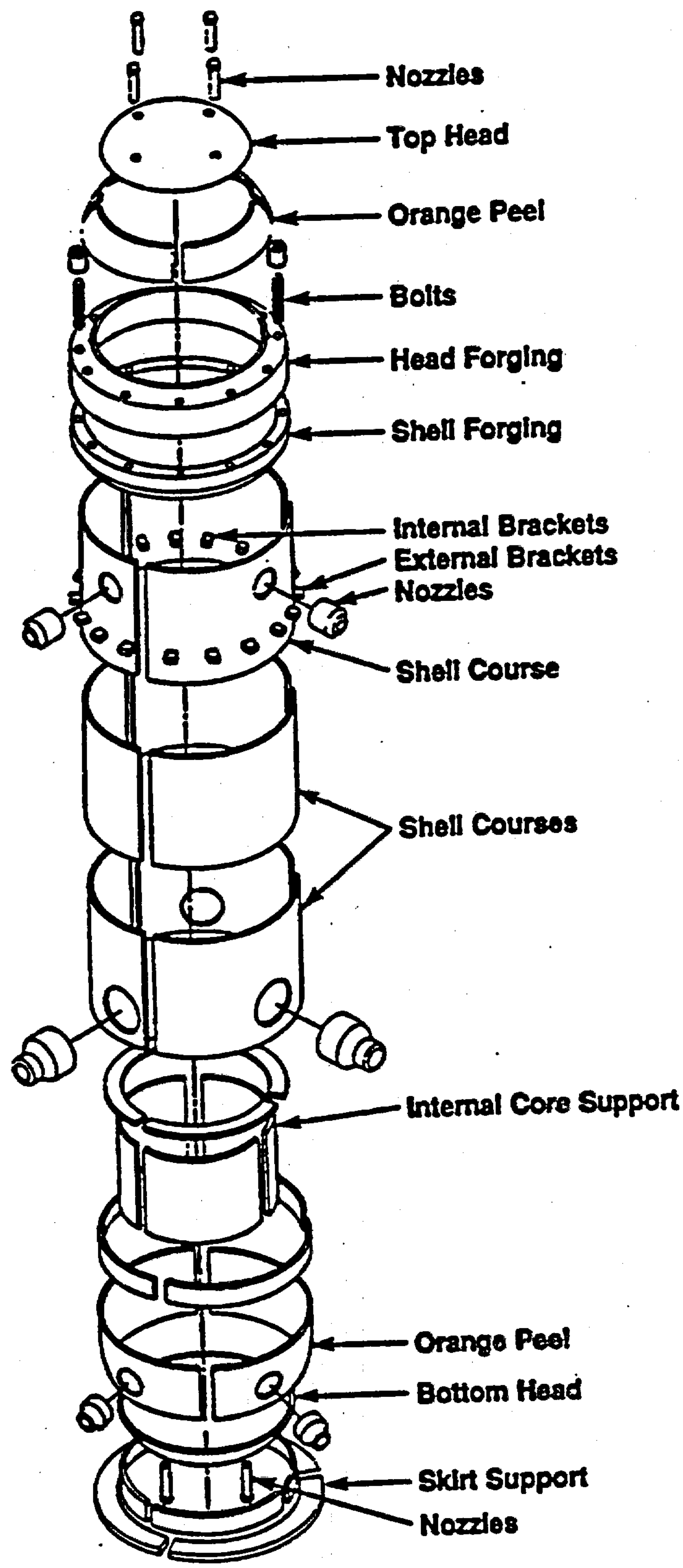


Figure 2.3 Exploded Schematic View of a Typical Boiling Water Reactor Pressure Vessel

STAFF EVALUATION OF BWRVIP-05 REPORT

Background (cont.)

- BWRVIP requested meeting with Commission to contest staff's position

- Meeting held on May 12, 1997 --

  - BWRVIP presented their technical analysis
  - Staff presented their qualitative basis for requiring inspection -- PSI not effective; defense in depth; credible service-related degradation mechanism; no past ISI for many vessels; RPV failure not tolerable

- Commission directed staff to complete the technical evaluation of the BWRVIP-05 report on expedited basis

  - Consider a tiered approach in gathering additional baseline information and/or implementing rule
  - SER to provide "comprehensive evaluation of the probabilistic analysis contained in the BWRVIP proposed alternative"
  - Review by ACRS

STAFF EVALUATION OF BWRVIP-05 REPORT

<u>Industry Analysis Results</u>

- BWRVIP performed probabilistic fracture mechanics analysis using design basis transients

- Generally a credible analysis framework -- staff has identified some specific problems in methodology, but not major shortcomings

  - Major problem lies in selection of transient -- design basis versus considering beyond design basis transients in probabilisitc analysis

- BWRVIP analysis reports <u>very</u> low probability of failure values

  - Apx. 5×10^{-8} per 40 years for axial welds
  - Apx. 5×10^{-40} per 40 years for circumferential welds

- Given the design basis transient considered, the staff would not compute significantly different values

STAFF EVALUATION OF BWRVIP-05 REPORT

Staff Analysis Results

- Analysis framework builds on Pressurized Thermal Shock analysis computer code -- probabilistic fracture mechanics analysis

    - Treats materials, fluence, and flaws as stochastic variables
    - Pressure-Temperature taken as point values for this analysis (typically stochastic in nature that vary in time)

- Significant effort to update flaw size distribution -- will be important for PTS but turned out to not have a significant impact on BWR RPV analysis

STAFF EVALUATION OF BWRVIP-05 REPORT

Staff Analysis Results

- Fabrication differences between axial and circumferential welds do not permit use of axial weld inspection results to infer condition of circumferential welds -- "tiered" approach is statistically invalid

- Most significant difference between staff and BWRVIP analyses is the transient

  - Staff identified a cold overpressure event at a foreign BWR -- similar enough to U.S. plants to be applicable to this analysis
  - PRA staff in RES and NRR considered potential for similar cold overpressure transients -- found 35 event precursors in U.S. database since 1980
  - Event frequency is on the order of 10^{-3} per year -- determined both by the operational event and by event tree analysis

STAFF EVALUATION OF BWRVIP-05 REPORT

Staff Analysis Results (cont.)

- Staff analysis predicts probability of failure on the order of

  - Apx. 10^{-5} per year for axial welds
  - Apx. 10^{-8} per year for circumferential welds
  - Both values are essentially 95% confidence bounds

Next Actions

- Public meeting with industry on August 8 to discuss staff analysis results -- SER (draft) to be made available next week

- NRR likely to request additional calculations from RES to support further requlatory actions for axial welds

- Regulatory position on inspection of circumferential welds still evolving

DISCUSSION OF INDEPENDENT ASSESSMENT OF BWRVIP-05

CONDITIONAL FAILURE PROBABILITY

E/8

CONDITIONAL PROBABILITY OF VESSEL FAILURE

- Performed Probabilistic Fracture Mechanics (PFM) Analyses for 3 General Classes of BWR RPVs (CE, B&W and CB&I)

- General Analysis Framework Similar to PTS Probabilistic Analysis

  → Material Properties, Fluence, Flaws Treated as Random Variables
  → Analysis is LEFM-Based
  → Cladding and Weld Residual Stresses Included in Analysis
  → Pressure and Temperature taken as Point Values (1150 psi, 88°F) for Cold Overpressure Event

- Reference cases developed for each RPV type

  → PFM Analysis Performed by Sampling On ± 3 Standard Deviation (σ) "within a vessel"
  → Sensitivity Studies Performed by "shifting the mean" Between Vessel by $+2\sigma$

ANALYSIS FRAMEWORK COMPARISON

- **Many Similarities to Analysis Framework in BWRVIP-05**

#	Key Variable	BWRVIP-05	NRC Staff Analysis
1	flaw aspect-ratios (surface length/depth)	∞ for circumferential flaws ∞ and 10 for axial flaws	2, 6, 10 (equally likely) for axial and circumferential flaws
2	stress intensity factors	Vessel Ri/t = 10	Vessel Ri/t = 20 (from ABAQUS weight functions)
3	weld residual stresses	Yes (cosine thru thickness)	Yes (from PWR vessel data)
4	cladding effects	Yes (some differences)	Yes
5	fracture toughness: K_{Ic}, K_{Ia}	mean curves from SECY 82-465, <u>no</u> statistical sampling	ASME Code curves as "mean-2σ", <u>with</u> sampling
6	crack initiation checked at:	crack depth and surface points	checked at 9 to 10 points around crack periphery
7	flaw size and density distribution	Marshall distributions, density = 30 flaws/meter^3	from PVRUF vessel, similar to Marshall, 994 flaws/m^3
6	stress corrosion cracking flaw initiation and growth	Yes	not explicitly in analysis (assumed through-clad cracks)

MAJOR DIFFERENCES IN FRAMEWORK

- Treatment of Fluence

  → BWRVIP-05 Considered Moderate Levels of Fluence (Point Values) in Tables 8-7, 8-8, and 8-9
  → Staff Found EOL Fluence Predictions to be Higher in Some Vessels

- Flaw Depth and Density Distributions

  → BWRVIP-05 Used Marshall Depth Distribution With a Density of 30 flaws/meter^3
  → Staff Used SAFT-UT Inspection Data on a PWR Vessel (PVRUF) Weld to Determine Flaw Depth Distribution and Flaw Density
  → Flaw Size Distribution Similar to Marshall
  → Flaw Density of 994 flaws/meter^3, With Significant Number of Flaws of Size $\leq$ 2 mm depth

SUMMARY OF RESULTS

Vessel Fabricator	Weld Orientation	Reference Case P(F|E)	95% Conf. Confidence Bound Between Vessels	Sensitivity to Flaw Size and Density[C]
Babcock & Wilcox (B&W)	Axial	9.5 x E-3	4.6 x E-2	1.6
	Circumferential	1.0 x E-6 [A]	> 1.4 x E-5 [A]	---[A]
Combustion Engineering (CE)	Axial	1.5 x E-3	2.9 x E-2	2.1
	Circumferential	< 1.0 x E-7 [A]	< 1.0 x E-7 [A]	---[A]
Chicago Bridge & Iron (CB&I)	Axial	4.3 x E-6	> 8.9 x E-5 [B]	1.2
	Circumferential	< 1.0 x E-7 [A]	< 1.0 x E-7 [A]	---[A]

Notes:

A Insufficient or no failures to accurately determine reference case failure probability -- up to 10 million simulations

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.

C Observed ratio was less than 1. Insufficient failures to accurately determine ratio.

ISI EFFECTIVENESS

- Effect of "One ISI Early in Plant Life" on Flaw Depth Distributions Was Investigated

- POD Curve From PISC-II Report (1993) Considered to Determine Effect on Flaw-Depth Distribution

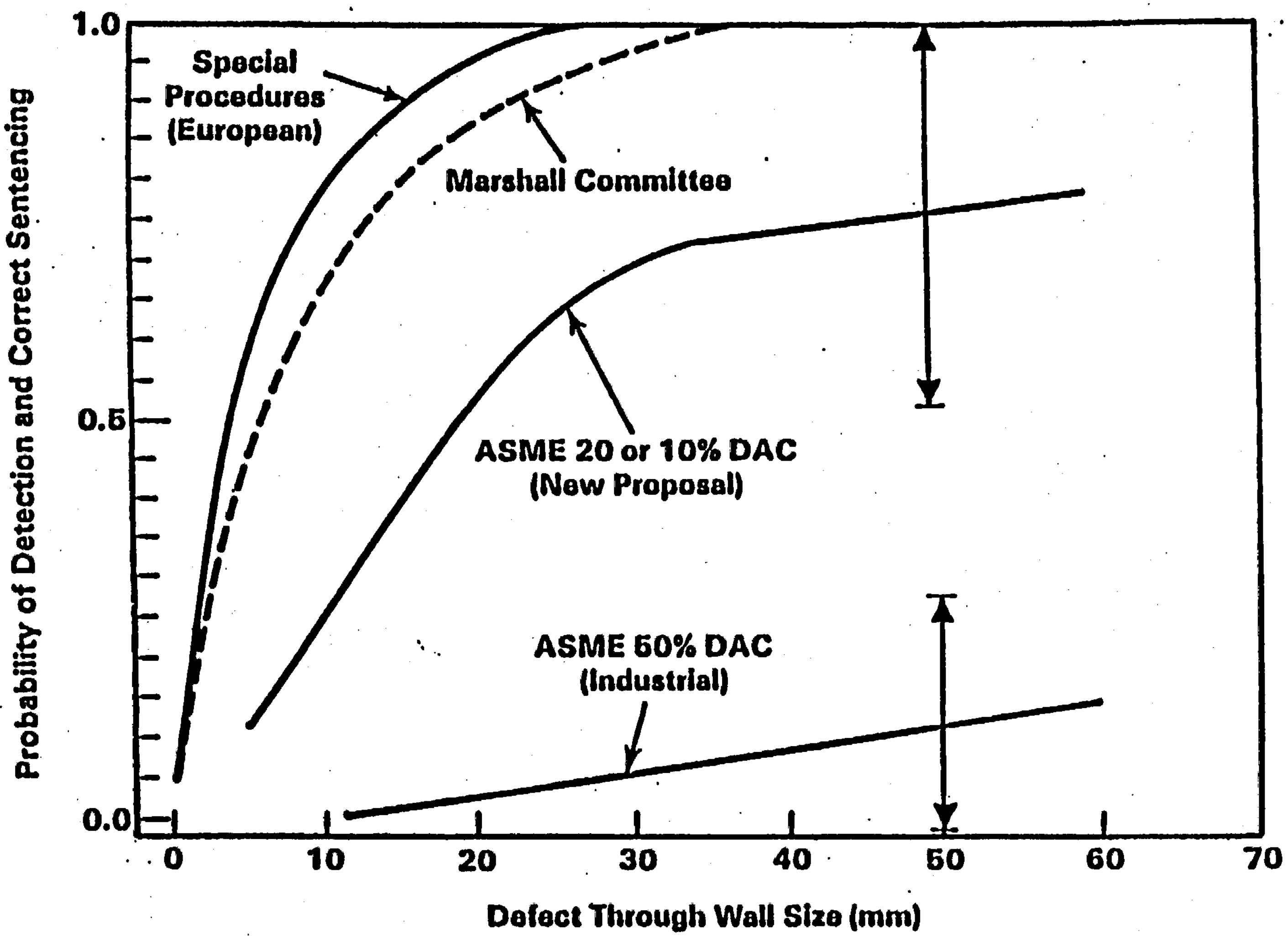
  → POD Achievable Using Modern Technology

- BWRVIP-05 Method-C POD Curve Also Considered

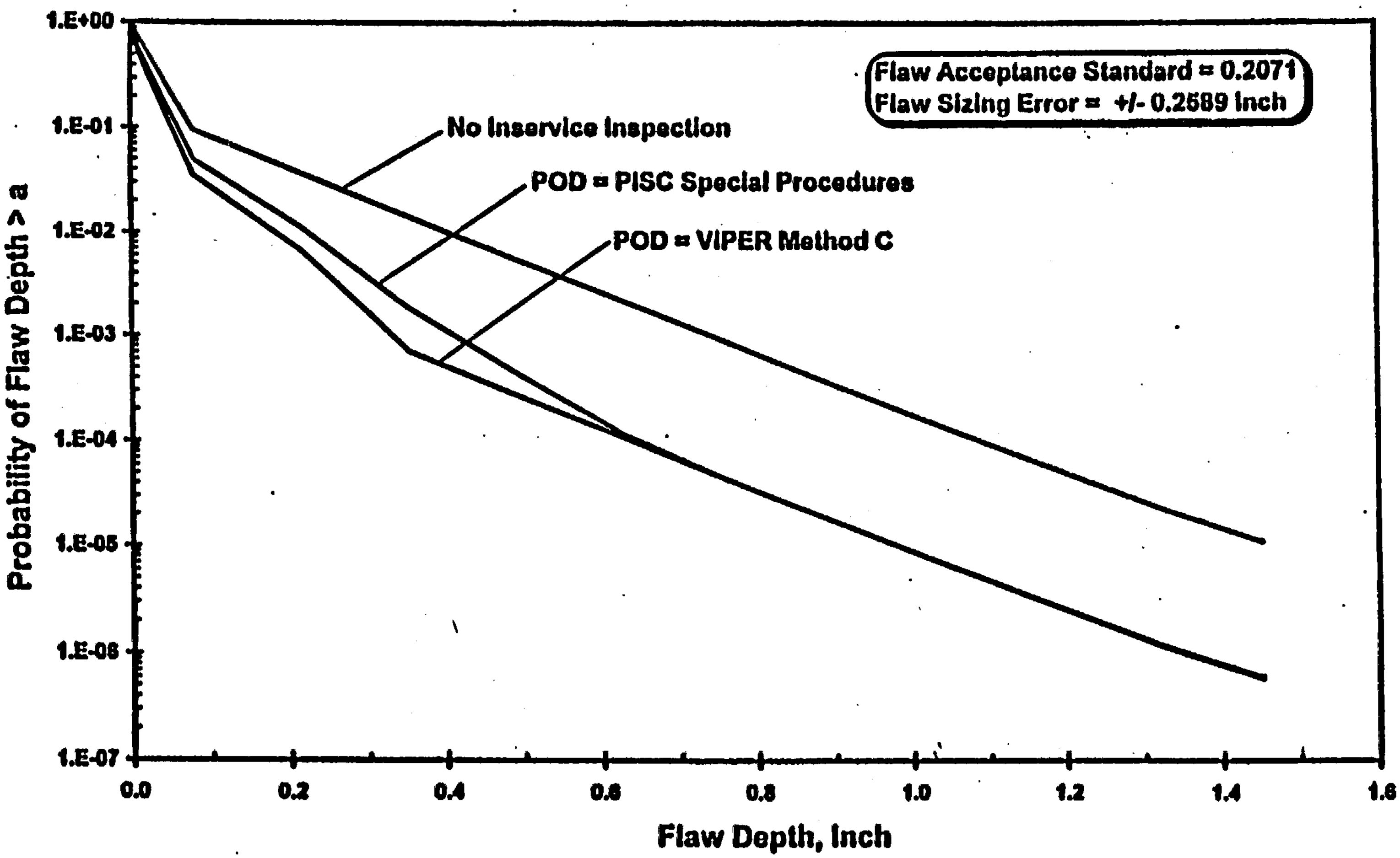
  → Curve Very Optimistic for Current Inspection Technology

- PNNL Computed Revised Flaw-Depth Distribution Using Staff Determined Best-Estimate Flaw Depth Distribution and --
  → PISC-II POD curve
  → VIPER Method-C POD

POD Curve from PISC-II Special Procedures (Nichols, Crutzen, 1993)



Effect of ISI with Differing POD Curves on Flaw Depth Distribution



RESULTS OF ISI EFFECTIVENESS

Vessel Fabricator	Weld Orientation	$P(F \mid E)_{without\ ISI}/P(F \mid E)_{with\ ISI}$ (for PISC-II POD)	$P(F \mid E)_{without\ ISI}/P(F \mid E)_{with\ ISI}$ (for VIPER Method C POD)
B&W	Axial	3.1	3.9
CE	Axial	3.4	5.0
CB&I	Axial	8.6	---[A]

Notes:

A No failures in 12 million simulations for the case with ISI

DISCUSSION AND COMMENTS

- Best-Estimate Maximum P(F|E) for Axial Cracks is on Order of 1.0×10^{-2} (B&W fabricated vessels)
- Best-Estimate Maximum P(F|E) for Circumferential Cracks is on Order of 1.0×10^{-6} (B&W vessels)
- Staff PFM Sensitivity Analyses Show that "between vessels" Variability in Fluence to have Highest Effect on P(F|E) When Fluence "Mean" Shifted to "mean + 2σ(between vessels)"
- 95% Confidence Bounds on P(F|E) (for B&W Fabricated Vessels) are:
  → 4.6×10^{-2} for Axial Cracks
  → 1.5×10^{-5} for Circumferential Cracks

APPENDIX

TABLE 1: VALUES OF MATERIAL PROPERTY PARAMETERS USED IN BWR RPV PFM ANALYSIS

VARIABLES	COMBUSTION ENGINEERING RPVs			BABCOCK & WILCOX RPVs			CHICAGO BRIDGE & IRON RPVs		
	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 ($X10^{19}$ n/cm2) (E>1MeV)	0.126	0.024	0.10	0.053	0.010	0.036	0.191	0.036	0.23
IRT_{NDT}, INITIAL REFERENCE TEMPER-ATURE (°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 2: THREE REFERENCE CASES FOR BWR VESSELS PFM ANALYSIS

Random Variable ($\mu \equiv$Mean), ($\sigma \equiv$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 xE19)	0.126 n/cm^2	0.053 n/cm^2	0.19 n/cm^2
σ (Fluence xE19)	0.024 n/cm^2	0.01 n/cm^2	0.036 n/cm^2
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/cm^2	0.125 n/cm^2	0.651 n/cm^2
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
Effect of ISI on Flaw Size Distribution	Revised Flaw Size Distribution	Revised Flaw Size Distribution	Revised Flaw Size Distribution

Thermo-mechanical Properties & Vessel Geometry

- Thermal conductivity (BTU/hr.ft.°F) = 24 (base-metal), 10 (clad)
  Specific heat (BTU/lb.°F) = 0.12 (base-metal, clad)
  Density (lb/ft^3) = 489 (base-metal, clad)
  Elastic Modulus (ksi) = 28000 (base-metal), 22000 (clad, from PWR test data)
  Thermal expansion coefficient (/°F) = 6.9E-6 (base-metal), 9E-6 (clad)
  Poisson's ratio = 0.3 (base-metal, clad)
  Cladding stress-free reference temperature (°F) = 515
  Heat-transfer coefficient ($BTU/hr.ft^2.°F$) = 1000

- Vessel internal radius, Ri = 112.5"; base-metal thickness = 5.25"; clad thickness = 0.2"; total thickness, t = 5.45"

- Cold overpressure transient: 1150 psi internal pressure at 88°F temp.

- Total axial and Hoop stresses computed due to
  - Internal pressure
  - Clad base-metal differential thermal expansion
  - Weld residual stress thru thickness -- from an actual PWR vessel axial weld (6.5 ksi maximum, which is lower than the 8 ksi max in BWRVIP-05)

Conditional P(F|E) for Reference Cases and Sensitivity Runs

Conditional Probability of Failure P(F|E) for Babcock & Wilcox Fabricated BWR Vessels

#	Case Description	Inner Surface $(\mu+2\sigma)$ RT_{NDT}°F	P(F|E) Circumferential Flaws	P(F|E) Axial Flaws
1	Reference Case 1	114.5	1.0 x E-6	9.5 x E-3
2	Mean Fluence = 0.125 x E19 n/cm²	145.1	2.5 x E-5	5.6 x E-2
3	Mean Copper = 0.317 wt %	135.1	1.0 x E-6	1.1 x E-2
4	Mean Nickel = 0.608 wt %	114.9	< 1.0 x E-6	9.7 x E-3
5	Mean Initial RT_{NDT} = 58.3°F	120.5	< 1.0 x E-6	9.8 x E-3
6	$(\mu+2\sigma)$ Flaw Density & Flaw Depth Distribution from PVRUF Vessel Data	114.5	4.0 x E-6	1.5 x E-2
7	Mean K_{Ic} = 0.94 x ASME Mean K_{Ic}	114.5	6.0 x E-6	2.0 x E-2
8	ISI with best-estimate PISC POD from Table 7-11 (column 2)	114.5	< 1.0 x E-7	3.1 x E-3

Reference Case definition:

Mean Fluence = 0.053 x E19 n/cm²
Mean Copper = 0.287 wt %
Mean Nickel = 0.6 wt %
Mean Initial RT_{NDT} = -5°F

Conditional P(F|E) for Reference Cases and Sensitivity Runs

Conditional Probability of Failure P(F|E) for Combustion Engineering Fabricated BWR Vessels

#	Case Description	Inner Surface $(\mu+2\sigma)$ RT_{NDT} °F	P(F|E) Circumferential Flaws	P(F|E) Axial Flaws
1	Reference Case 1	93.2	< 1.0 x E-7	1.5 x E-3
2	Mean Fluence = 0.326 x E19 n/cm²	137.5	2.1 x E-5	3.7 x E-2
3	Mean Copper = 0.256 wt %	97.4	< 1.0 x E-7	2.9 x E-3
4	Mean Nickel = 0.776 wt %	94.4	< 1.0 x E-7	2.5 x E-3
5	Mean Initial RT_{NDT} = -50°F	99.2	< 1.0 x E-7	3.8 x E-3
6	$(\mu+2\sigma)$ Flaw Density & Flaw Depth Distribution from PVRUF Vessel Data	93.2	6.0 x E-7	3.1 x E-3
7	Mean K_{Ic} = 0.94 x ASME Mean K_{Ic}	93.2	1.0 x E-6	4.3 x E-2
8	ISI with best-estimate PISC POD from Table 7-11 (column 2)	93.2	< 1.0 x E-7	4.4 x E-4

Reference Case definition:

Mean Fluence = 0.126 x E19 n/cm²
Mean Copper = 0.226 wt %
Mean Nickel = 0.76 wt %
Mean Initial RT_{NDT} = -56°F

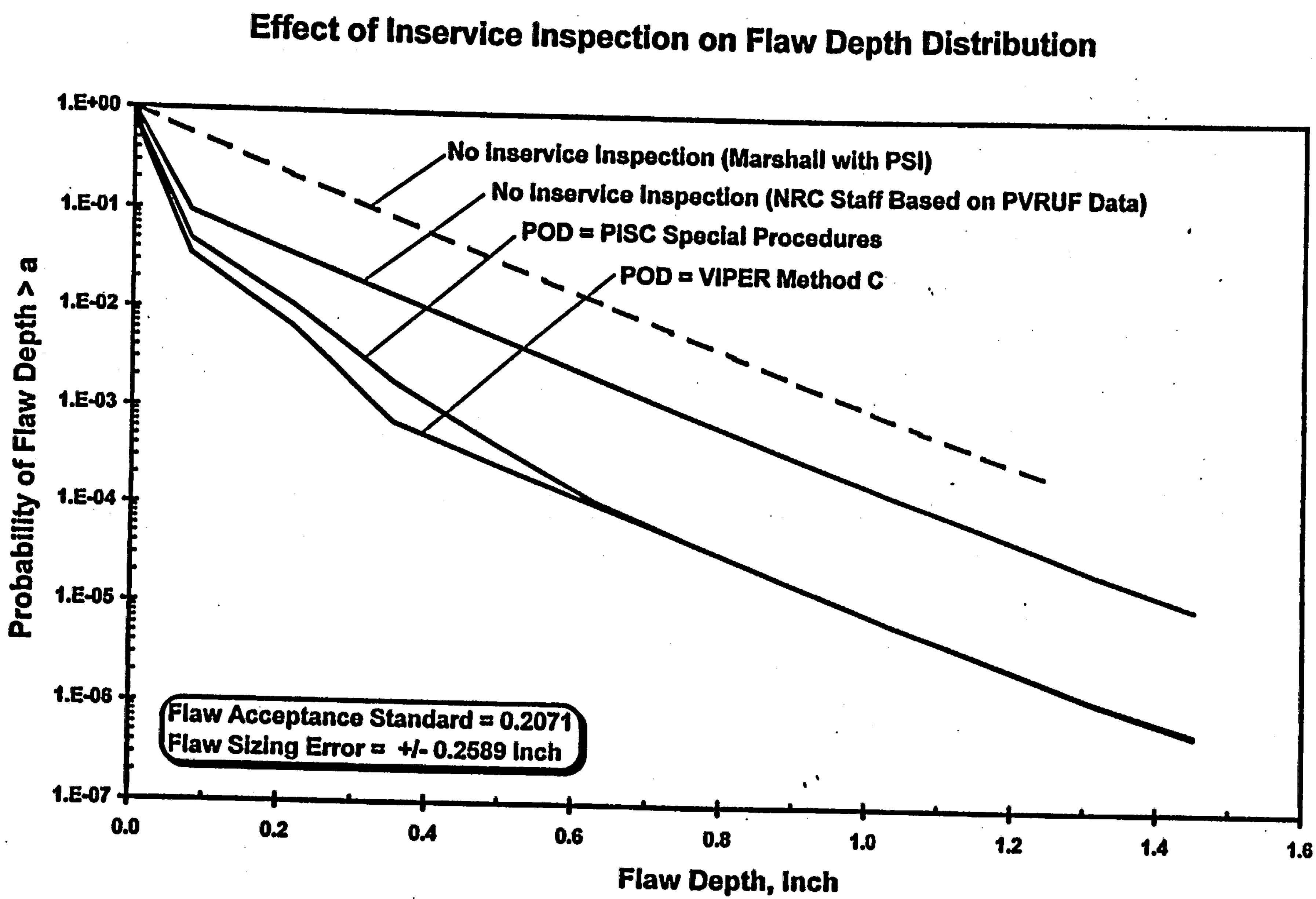
Conditional P(F|E) for Reference Cases and Sensitivity Runs

Conditional Probability of Failure P(F|E) for Chicago Bridge & Iron Fabricated BWR Vessels

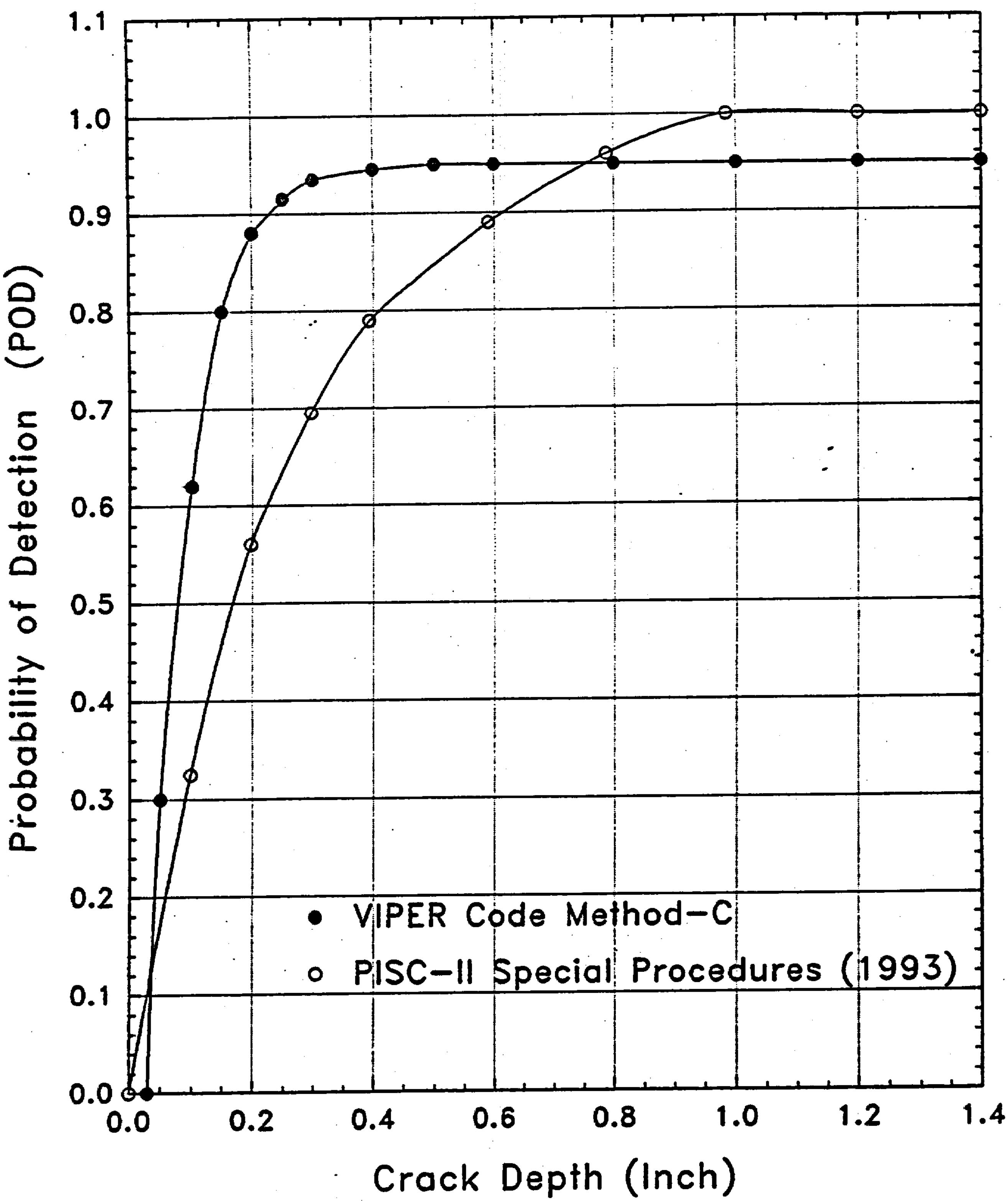
#	Case Description	Inner Surface $(\mu+2\sigma)$ RT_{NDT} °F	P(F|E) Circumferential Flaws	P(F|E) Axial Flaws
1	Reference Case 1	32.7	N/A	4.3 x E-6
2	Mean Fluence = 0.651 x E19 n/cm²	50.0	N/A	1.7 x E-4
3	Mean Copper = 0.05 wt %	40.5	N/A	---
4	Mean Nickel = 0.96 wt %	32.7	N/A	---
5	Mean Initial RT_{NDT} = -50°F	38.7	N/A	---
6	$(\mu+2\sigma)$ Flaw Density & Flaw Depth Distribution from PVRUF Vessel Data	32.7	N/A	5.0 x E-6
7	Mean K_{Ic} = 0.94 x ASME Mean K_{Ic}	32.7	N/A	---
8	ISI with best-estimate PISC POD from Table 7-11 (column 2)	32.7	N/A	5.0 x E-7

Reference Case definition:

Mean Fluence = 0.191 x E19 n/cm²
Mean Copper = 0.04 wt %
Mean Nickel = 0.93 wt %
Mean Initial RT_{NDT} = -56°F


Effect of Inservice Inspection on Flaw Depth Distribution
No Inservice Inspection (Marshall with PSI)
No Inservice Inspection (NRC Staff Based on PVRUF Data)
POD = PISC Special Procedures
POD = VIPER Method C
Flaw Acceptance Standard = 0.2071
Flaw Sizing Error = +/- 0.2589 Inch
Probability of Flaw Depth > a
Flaw Depth, Inch
1.E+00
1.E-01
1.E-02
1.E-03
1.E-04
1.E-05
1.E-06
1.E-07
0.0
0.2
0.4
0.6
0.8
1.0
1.2
1.4
1.6

A Comparison of POD Curves



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

October 10, 1997

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

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION REGARDING BWRVIP-05 (TAC NO. M93925)

Dear Mr. Terry,

By letter dated September 10, 1997, Robert L. Seale, Chairman of the Advisory Committee on Reactor Safeguards (ACRS), provided recommendations on several areas that the ACRS determined needed additional review before the NRC staff issues its final safety evaluation report (SER) on the BWR Vessel and Internals Project's (BWRVIP) proprietary report, "BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations (BWRVIP-05)," dated September 28, 1995, as supplemented by letters dated June 24 and October 29, 1996, and May 16, June 4, and June 13, 1997. The ACRS subcommittees on Materials and Metallurgy and on Severe Accidents met with the NRC staff and representatives of the BWRVIP on August 26, 1997, and the full ACRS committee reviewed the BWRVIP-05 report and the associated staff independent safety assessment, dated August 14, 1997, during the 444th meeting on September 4, 1997.

The ACRS made several recommendations in both the above referenced letter and meetings regarding the staff's review of the BWRVIP-05 report. Enclosed is a request for additional information related to these recommendations. We request that you expedite your response to this request in order that the staff may complete the final BWRVIP-05 SER, and have the ACRS review this SER, prior to the Spring 1998 outage season.

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

Enclosure: as stated
cc: See next page

E/19

REQUEST FOR ADDITIONAL INFORMATION REGARDING REPORT EPRI TR-105697 "BWR VESSEL AND INTERNALS PROJECT - BWR REACTOR PRESSURE VESSEL SHELL WELD INSPECTION RECOMMENDATIONS (BWRVIP-05)," SEPTEMBER 1995

1) Provide a technical justification regarding the adequacy of the Marshall flaw size distribution used in probabilistic fracture mechanics calculations. This distribution predicts that the fraction of flaws larger than a specific size decreases exponentially with flaw size. It is this exponential decrease that is chiefly responsible for the very large differences in failure probability between axial and circumferential welds. It is not clear that the experts who formulated this distribution as a bound on the frequency of large flaws intended for the distribution to be used to compare the relative frequencies of the approximately 2 cm flaws that lead to failure in axial welds and the approximately 4 cm flaws that lead to failure in circumferential welds.

    a) Evaluate all available inspection data (PWRs, BWRs, and research), including those obtained in past inspections of the welds, to determine if the Marshall flaw size distribution produces realistic flaw distributions.

    b) Address the flaw size distribution uncertainties associated with the BWRVIP-05 analyses which show that flaw size distributions inputed to the models are justifiable and consistent with available data.

2) Provide an uncertainty analysis to address conservatisms inherent in existing calculations, including the assumptions that weld flaws penetrate through the cladding and that all flaws are located on the inner surface of the vessel. The uncertainty analysis should also consider accident sequences that pose a threat to the welds, and operator actions that affect the probability of challenges to the integrity of BWR vessel welds.

3) Discuss the relative value of partial inspections of the welds, including limitations in both accessibility of welds and the capability to detect flaws by inspections.

4) Discuss whether the industry is developing and maintaining a database on vessel embrittlement specific to BWRs. International fabrication experience, to the extent available, should be considered in developing this database.

5) The NRC staff requests that the BWRVIP provide an analysis for beyond the current license limit. In these analyses, the contributions of base metal failure should be included.

6) Considering that plate sections can have a higher number of total flaws than the weld sections, determine the probability of vessel failure from flaws in the plate sections and its impact on the BWRVIP-05 report conclusions.

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

Robert L. Seale, Chairman
Advisory Committee on Reactor Safeguards
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Dr. Seale:

SUBJECT: BOILING WATER REACTOR PRESSURE VESSEL SHELL WELD INSPECTION RECOMMENDATIONS (BWRVIP-05)

By letter dated September 10, 1997, you provided a summary of the Advisory Committee on Reactor Safeguards' (ACRS) review of the NRC staff's independent safety assessment, dated August 14, 1997, of the Boiling Water Reactor Vessel and Internals Project's (BWRVIP) proprietary report, "BWR Vessel and Internals Project, BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations (BWRVIP-05)." This report, dated September 28, 1995, as supplemented by letters dated June 24 and October 29, 1996, and May 16, June 4, and June 13, 1997, 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 essentially 100 percent of the axial welds and essentially zero percent of the circumferential welds[1]. The ACRS subcommittees on Materials and Metallurgy and on Severe Accidents met with the staff and representatives of the BWRVIP on August 26, 1997, and the full ACRS committee reviewed the BWRVIP-05 report and the associated staff independent safety assessment during the 444th meeting on September 4, 1997.

In your letter, the ACRS made several recommendations regarding the staff's review of the BWRVIP-05 report. Following is our understanding of your September 10, 1997, recommendations:

a) The industry proposal to inspect less than 100 percent of the vessel welds is essentially a request for a change in the current licensing basis for BWRs as a class. This request should be handled using the risk-informed process now being developed by the NRC staff.

b) Some additional efforts are needed to address uncertainties associated with the BWRVIP-05 analyses, such as showing that flaw size distribution input to the models are justifiable and consistent with available data, including those obtained in past inspections of the welds. Data from inspections of welds in PWR vessels may also be of use in the definition of realistic flaw distributions.

G:\BWRVIP\BWRVIP05\SEALE.LTR

CONTACT: C. Carpenter, DE/NRR
415-2169

1 The intersections of the axial and circumferential welds would have been included under this proposal such that approximately 2 - 3 percent of the circumferential welds would be inspected.

c) The adequacy of the Marshall flaw size distribution, which is used by both the staff and industry in probabilistic fracture mechanics calculations, needs careful consideration. This distribution predicts that the fraction of flaws larger than a specific size decreases exponentially with flaw size. It is this exponential decrease that is chiefly responsible for the very large differences in failure probability between axial and circumferential welds. It is not clear that the experts who formulated this distribution as a bound on the frequency of large flaws intended for the distribution to be used to compare the relative frequencies of the approximately 2 cm flaws that lead to failure in axial welds and the approximately 4 cm flaws that lead to failure in circumferential welds.

d) The uncertainty in the nature of the flaw distribution is the most critical factor in determining the relative probability of failure between circumferential and axial welds. Additional uncertainties need to be addressed to more accurately assess the actual failure probabilities of BWR vessel welds. To address such uncertainties, a comprehensive analysis is needed of accident sequences that pose threats to the welds. Careful consideration should be given to operator actions that affect the probability of challenges to the integrity of BWR vessel welds. The uncertainty analysis should address conservatism inherent in existing calculations, including the assumption that weld flaws penetrate through the cladding and that all flaws are located on the inner surface of the vessel.

e) The staff should consider the relative value of partial inspections of the welds. In truth, 100 percent inspection of welds will seldom be practicable because equipment configurations in BWRs will limit access to some welds. Limitations in the capability to detect flaws by inspections need to be recognized as well.

f) The ACRS encourages the staff to continue development of a database on vessel embrittlement specific to BWRs.

g) The apparent rapid increase of vessel failure probability at 40 years, however, suggests that it is important to also analyze failure probabilities for plant life extensions beyond the current license limit. In these analyses, the contributions of base metal failure should be included.

Additionally, during the August 26, 1997, briefing of the ACRS subcommittees on Materials and Metallurgy and on Severe Accidents, the following recommendations were made:

h) International fabrication experience, to the extent available, should be considered in developing the database discussed in (f) above.

i) Considering that plate sections can have a higher number of total flaws than the weld sections, determine the probability of vessel failure from flaws in the plate sections.

The staff is working on these issues, and will address your recommendations in the final safety evaluation report (SER). The staff plans to complete the final SER, and have the ACRS review it, by January 1998. However, this presumes that industry provides a timely and adequate response to the staff's initial request for additional information (RAI), dated August 14, 1997, and the subsequent RAI which was developed from the above ACRS recommendations. Further, it should be noted that if the staff agrees in its final SER that relaxed inspection requirements for BWR circumferential shell welds are warranted, further actions will be required to modify existing NRC regulations.

Sincerely,

L. Joseph Callan
Executive Director
for Operations

cc: Chairman Jackson
Commissioner Dicus
Commissioner Diaz
Commissioner McGaffigan
SECY

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, BWR STANDBY LIQUID CONTROL SYSTEM/CORE PLATE ΔP INSPECTION AND FLAW EVALUATION GUIDELINES (BWRVIP-27)" (TAC NO. M98708)

Dear Mr. Terry:

By your application dated April 25, 1997, you submitted for NRC staff review the Electric Power Research Institute (EPRI) proprietary report, "BWR Vessel and Internals Project, BWR Standby Liquid Control System/Core Plate ΔP Inspection and Flaw Evaluation Guidelines (BWRVIP-27)," EPRI Report TR-107286. The BWRVIP-27 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-27 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

cc: See next page
Enclosure: As stated

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UNITED STATES
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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, BWR STANDBY LIQUID CONTROL SYSTEM/CORE PLATE ΔP INSPECTION AND FLAW EVALUATION GUIDELINES (BWRVIP-27)" (TAC NO. M98708)

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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

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

March 27, 1998

Mr. Carl Terry
BWRVIP Chairman
Niagara Mohawk Power Company
P.O. Box 63
Lycoming, NY 13093

Subject: CLOSEOUT FOR BWR VESSEL AND INTERNALS PROJECT (BWRVIP), UPDATE OF BOUNDING ASSESSMENT OF BWR/2-6 REACTOR PRESSURE VESSEL INTEGRITY ISSUES (BWRVIP-46)

Dear Mr. Terry:

On May 19, 1995, the NRC issued Generic Letter 92-01, Revision 1, Supplement 1 (GL 92-01, Rev. 1, Supp. 1), "Reactor Vessel Structural Integrity." In GL 92-01, Rev. 1, Supp. 1, the NRC requested that nuclear licensees perform a review of their reactor pressure vessel structural integrity assessments in order "to identify, collect, and report any new data pertinent to [the] analysis of [the] structural integrity of their reactor pressure vessels (RPVs)." The Supplement also requested that licensees assess the impact of any new data relative to requirements which encompass pressurized thermal shock (PTS) and upper shelf energy (USE) evaluations; and any potential impact on low temperature overpressure (LTOP) limits or pressure-temperature (P-T) limits.

In November 1995, the BWRVIP submitted the report "BWR Vessel and Internals Project, Bounding Assessment of BWR/2-6 Reactor Pressure Vessel Integrity Issues (BWRVIP-08NP)." At that time, no new data had been considered for the bounding assessments. A January 7, 1997, letter from K. R. Wichman (NRC) to R. G. Carter (EPRI) served as follow-up to a December 13, 1996, conference call. Specifically, the letter reiterated one of the requests of GL 92-01, Rev. 1, Supp.1 which was to provide the NRC with a description of actions taken or planned to locate all data relevant to the determination of RPV integrity. The letter also restated the staff's position that although the use of bounding values for licensing actions would be acceptable, it is necessary to collect available data in order to demonstrate that the values used are, in fact, bounding.

By letter dated December 23, 1997, the BWR Vessel and Internals Project (BWRVIP) submitted "BWR Vessel and Internals Project, Update of Bounding Assessment of BWR/2-6 Reactor Pressure Vessel Integrity Issues (BWRVIP-46)," as an updated generic BWR response to GL 92-01, Rev. 1, Supp. 1. BWRVIP-46 considered additional data retrieved by Owners Groups, nuclear steam supply system vendors, and the BWRVIP for determination of new bounding chemistry values for some of the weld wire heats. In particular, sources of the new data were: 1) the Combustion Engineering Owners Group (CEOG) database released in July 1997 which contains all known data for CE fabricated welds in PWR and BWR vessels; 2) Framatome Technologies Incorporated (FTI) analyses of Linde 80 welds which are documented in NRC Inspection Report 99901300/97-01 dated January 28, 1998; 3) FTI's

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analysis of electro-slag welds which was referenced in a Dresden and Quad Cities pressure-temperature (P-T) limits submittal dated September 20, 1996; and 4) Chicago Bridge and Iron quality assurance records.

The BWRVIP-46 report evaluated all new data, where applicable, for all BWRs in order to determine new bounding chemistries. Only those BWRs with increases in copper (Cu) greater than 0.05% were evaluated for impact on P-T curves (for BWRs with less than 0.05% increase in Cu, the staff compared the chemistry value ranges to the values in the reactor vessel integrity database (RVID) for completeness). Bounding chemistry values were assumed for each weld heat to project embrittlement levels. These values were compared to the limiting adjusted reference temperature (ART) and previously calculated values in the BWRVIP-08NP report. The staff verified that the chemistry values used in the BWRVIP-46 report did, in fact, bound values from the CEOG report, the FTI inspection of weld fabrication records, and the RVID.

The staff concluded that all but one of the BWR vessel have P-T curves that would be conservative when assuming the new bounding chemistries, where applicable. One vessel, Cooper, has an increase in ART of 3°F (102°F to 105°F) when the new bounding chemistry is assumed. This change is not expected to have a significant impact on the Cooper RPV integrity assessments. In addition, the new data in all cases do not affect the previous conclusion that the equivalent margin analysis for all US BWR/2-6 vessels is bounding for upper shelf energy in beltline welds.

The BWRVIP-46 report does not represent a commitment for any utility. It is the staff's expectation that BWR licensees will review the information in the report in order to respond to forthcoming plant-specific requests for additional information (RAIs). Issuance of plant-specific RAIs was also discussed at a NRC/NEI meeting on November 12, 1997, and at the NRC/NEI Reactor Pressure Vessel Integrity Workshop on February 12-13, 1998.

This letter serves as close-out for TAC No. M99897 which was opened for review of the original response to GL 92-01, Rev. 1, Supp. 1 (report BWRVIP-08NP).

Thank you for your cooperation.

Sincerely,

JRS/for

Gus C. Lainas, Acting Director
Division of Engineering
Office of Nuclear Reactor Regulation

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Division of Engineering
Office of Nuclear Reactor Regulation

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98-277

BWRVIP

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

MEMORANDUM

June 17, 1998

TO: BWRVIP Repair Committee
Louise Lund
Gene Carpenter
Lew Willertz
Robin Dyle

FROM: Bruce McLeod/Bob Thomas KJW for

SUBJECT: Minutes of Meeting with NRC Regarding Weldability of Irradiated Materials

A meeting with the NRC was held on June 4, 1998 to discuss a possible BWRVIP/NRC joint project which would have the objective of improving our ability to weld irradiated stainless steel. The meeting was attended by: Louise Lund and Gene Carpenter of the NRC; Bruce McLeod and Lew Willertz from the BWRVIP; Robin Dyle of Inservice Engineering; and Bob Thomas and Ken Wolfe of EPRI.

When stainless steel is irradiated to relatively high levels, the boron and nickel contained in the steel transmute to helium. When welding is attempted on irradiated materials with a high helium content, the helium coalesces and forms small bubbles which may result in cracks in the weld. The BWRVIP has a specific need to be able to perform a weld repair on certain irradiated locations for which mechanical repairs are not available. The most obvious of these is the weld which attached the jet pump riser brace to the vessel wall.

Both the NRC and the BWRVIP Repair Committee have requested funding in their 1999 budgets for investigations related to the weldability if irradiated materials. It would appear to be in both parties best interests to conduct a joint program. The objective of the meeting was to discuss how such a joint project might be conducted and what its scope might be. A decision on whether to proceed with the project will depend on the outcome of the funding requests.

The discussion fell into the three basic areas listed below.

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1. Memorandum of Understanding

A Memorandum of Understanding (MUO) is in place between the NRC and EPRI which describes the conditions under which a joint project may be undertaken, what it's scope may include and how it shall be conducted. In short, the subject project falls within the guidelines of the MOU as long as it results primarily in the generation of a body of data which both NRC and EPRI can use to their own ends. An attachment to the MOU which describes the scope of the project needs to be developed and approved by EPRI and NRC management.

2. Project Scope

All parties at the meeting agreed that, due to the limited amount of funding available, the project would have to focus on a practical solution to solving to the problem rather than embarking on an academic study to attempt to fully understand all aspects of the phenomenon. It was agreed that the workscope should not attempt to address such things as improved methods for calculating fluence or boron content at ex-core locations. Rather, it should address the following:

- Determination of the local helium content directly by measurement

  Approximate threshold values for the helium level at which welding becomes problematic are known. A search should be conducted to determine if methods for in-situ measurement of helium content are available. In addition, it should be determined if there are ways to "pre-treat" an irradiated material to lower it's helium concentration or otherwise improve its weldability. These techniques should be benchmarked.

- Establish a "Never Mind" Boundary

  The results of EPRI reportBWRVIP-45 and other research indicates that there are regions in the reactor where the fluence remains low for the life of the plant and, consequently, weldability is not an issue. The project should attempt to define this region.

- Short/Long-term materials issues

  When welding is performed on an irradiated material with a high helium content, it appears that the cracking develops immediately if it is going to develop at all. Thus, it does not represent a safety issue since the cracking would be found in a post weld inspection and dealt with. It does, however, represent an economic concern to a utility since, if cracking is observed, an alternate method of repair may need to be implemented, presumably prior to restart. The project should verify that the cracking does occur immediately and establish acceptance criteria for any cracking observed. Long term effects of the cracking on material properties should be established.

- Perform tests of welding techniques

  The project should evaluate the best techniques for performing welds on irradiated material and perform tests to verify their acceptability. Such tests will be expensive and must be selected judiciously. Irradiated material must be located on which to perform the tests. It was noted that it may be cost effective to perform the tests at the location where the material is currently stored rather than shipping the material to a lab.

3. Project Organization

The project will be managed jointly by the NRC (Louise Lund) and EPRI (Bob Thomas). Details will be decided at a later time.

A number of action items were recorded which will provide additional information required for project planning. The action item list is attached.

The group agreed that a conference call should be held on July 15 to assess the status of the project and define further actions.

The meeting was adjourned.

C: BWRVIP Technical Chairmen
EPRI Task Managers

Action Items
6/4/98 Meeting on Weldability of Irradiated Materials

1. Contact TEPCO. Attempt to obtain information/data on weldability issues encountered during shroud replacement activities. (EPRI)

2. Contact GE to determine what they are doing in the area and determine if a partnering agreement is possible (EPRI- 6/30). If so, attempt to schedule a meeting with GE to discuss (EPRI- set date for meeting by 7/15).

3. Invite TEPCO to 10/98 Water Reactor Safety meeting. (NRC).

4. Ask MITI/TEPCO to make presentation to NRC. (NRC)

5. Talk to German contacts (VGB, Siemens) to determine what useful information they may have. (EPRI)

6. Contact organizations that may have techniques for insitu determination of helium content. These may include: Battelle, NW Labs, SWRI, Savannah River (Lauden), ORNL, Westinghouse, Japanese patents, etc. (Willertz/Wolfe determine what's available by 6/30)

7. Develop list of available irradiated materials for use in the project. Query BWRVIP members (including Swedish and Japanese utilities) as well as others (e.g., Portland, Vallecitos, etc.). (EPRI – letter out by 6/30; request response by 7/15).

8. Develop strawman test plan for evaluating welding techniques (material properties, irradiation level, heat input, etc.(Dyle)

9. Set up conference call for 7/15 to discuss project status. (Lund/Thomas)

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Agenda

Process to Revise NUREG-0313, Rev. 2
March 16, 1999

USNRC Offices
One White Flint North.
Room 4-B-8
Rockville, MD

8:30 AM	Introductions	G. Carpenter – NRC
8:45 AM	BWRVIP Presentation on NUREG-0313 • Historical Perspective • Scope and Approach • Schedule and Resources	S. Lewis – BWRVIP
9:30 AM	NRC Remarks	
9:45 AM	Technical Discussions	All
12:00 PM	Lunch	
1:00 PM	Technical Discussions	All
3:00 PM	Conclusions	All
4:00 PM	Adjourn	

E/13

Meeting with BWRVIP
March 16, 1999

NAME	TITLE	ORGANIZATION	TELEPHONE	E-MAIL
C. E. Carpenter	Lead Project Manager	NRC/NRR/DE/EMCB	301-415-2169	cec@nrc.gov
Bill Koo	Sr. Matls Engr.	NRC/NRR/DE/EMCB	301-415-2706	WHK@NRC.gov
Robin Dyle	Engineer	Inservice Engineering	205.497.0497	RDyle@MSN.com
Frank Ammirato	NDE Section MGR	EPRI	704-547-6129	fammirat@epri.com
Timir Misra	Engineer	NRR/DE/EMCB	301-415-2467	txm2@nrc.gov
Michael McNeil	Materials Engineer	NRC/RES/DET	301-415-6794	mbm@nrc.gov
Joe Muscara	Senior Metallurgical Eng.	NRC/RES/DET	301-415-5844	JXM8@NRC.GOV
John Lewis	ISI Program MGR.	Enertech Servus	770 271-8456	jtlewis2@aol.com
L. J. Victory Jr.	Vice President	Enertech Servus	770-271-8449	LJVJR@aol.com
Pat O'Regan	Proj Mgr RI-ISI	EPRI	508 497 5045	poregan@epri.com
Glenn Warren	BWROG Chairman	BWROG	205-992-5940	wgwarren@southernco.com
Bob Penny	MGR-BWROG	BWROG/VIP	914-681-6288	Robert.Penny@NYPA.GOV
Jim White	Vice President	Curtiss Wright Flow Control	516-293-3800 x486	jwhite@tre curtisswright.com
Bob Carter	Prog. Mgr.	EPRI	704-547-6019	bcarter@epri.com
Bob Hermann	Sr. Level Advisor for Mats Sci	NRC	301-415-2768	RAH@NRC.GOV
Tim Abney	Licensing & Industry Affairs Mgr	TVA/Browns Ferry	256 729-2636	teabney@tva.gov
Steve Lewis	BWRVIP Assess. Comm Tech. Chair	Entergy	601-437-6194	SLEWIS1@ENTERGY.com

ATTACHMENT 1

Revision of NUREG-0313, Rev. 2

March 16, 99 Mtg

Steve Lewis - BWRVIP

Bob Carter - EPRI

NRC/ Industry Mtg

Outline

- Purpose of Meeting
- Goals of Program
- Background
- Scope
- Approach
- Schedule
- Resources
- Conclusions

Purpose of Meeting

- Begin discussions with NRC staff regarding a revision to NUREG-0313
- Express BWRVIP and BWROG commitment to the program
- Review and discuss basic scope and approach
- Examine schedule and resources
- Explore details of a generic approach

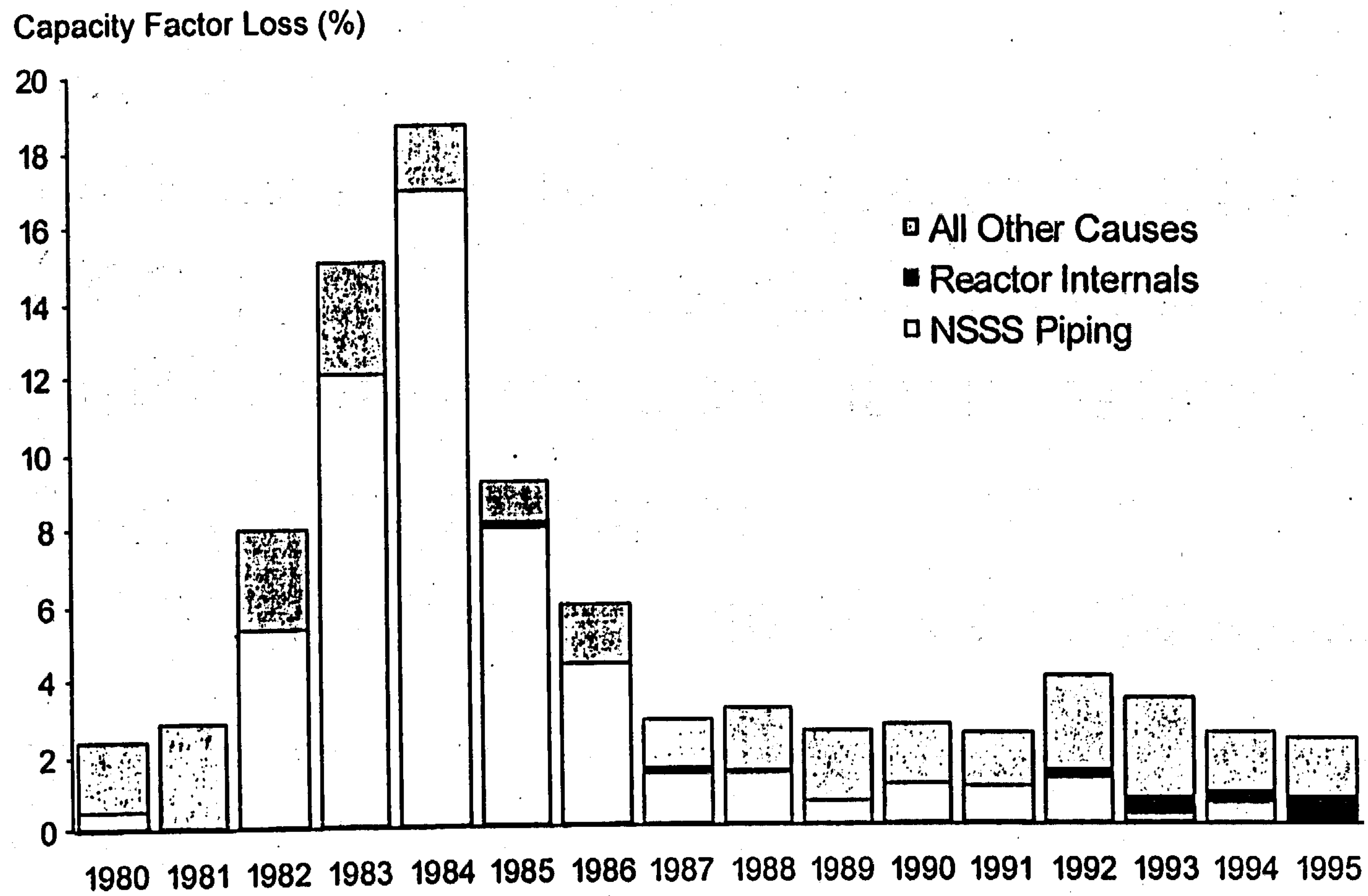
Goals

- Develop a technical basis to revise NUREG-0313
  - Incorporate previous and current IGSSC work of the BWRVIP and BWROG
- Product must be beneficial to industry and acceptable to NRC
- Submit to NRC by end of 1999

Background

- Austenitic stainless steels are widely used in boiling water reactors
- Intergranular stress corrosion cracking (IGSCC) has occurred in piping welds and more recently in various reactor internal components
- Industry performed extensive inspections, repairs and replacements of piping during 1980 to 1988

Reduction in BWR capacity factor due to corrosion-related degradation



BWR materials

- Exposed materials include
  - Type 304 and 316 stainless steels
  - Type 308 weld filler and cladding
  - Low-carbon grades 304L, 316L, 316NG
  - Nickel-based alloys 600, 82, 182

IGSCC in BWR piping: stainless steels

◆ 1960s	◆ Scattered incidents of IGSCC
◆ mid-1970s	◆ Small diameter piping
	◆ Association with weld residual stress
◆ late 1970s	◆ Large diameter piping
◆ mid-80s	◆ Crevice-induced in 304L and 316L
	◆ Association with surface cold work

Industry response

- Collaboration on remedy development
  - BWR Owners Group for IGSCC Research
    BWROG I 1979-1983; BWROG II 1984-1988.
  - New developments and adopted innovations
- Plant-specific decisions on remedy selection varied
  - Full or partial piping system replacements
  - Local repair and augmented inspection
  - Local mitigation and augmented inspection
- Regulatory guidance on remedy implementation
  NUREG-0313 Rev 2, 1988

NUREG-0313, Rev. 2 categories

Category	Weld Description	Inspection Frequency
A	Resistant materials	25% sample every 10 years (Same as Code)
B	Non-resistant materials stress improved within 1st 2 years of operation	50% every 10 years (at least 25% in 6 years)
C	Non-resistant materials stress improved after 2 years of operation	Once within 2 cycles of stress improvement then once per every 10 years
D	Non-resistant materials, no stress improvement	100% every 2 refueling cycles
E	Cracked - reinforced by weld overlay or mitigated by stress improvement	Every 2 refueling cycles
F	Cracked – Inadequate or no repair	Every refueling outage
G	Non-resistant, not inspected	Next outage

IGSCC control strategies implemented

- Detect IGSCC before damage compromises system integrity
- Remove found defects before continued growth compromises system integrity
- Prevent initiation by introducing a resistant material
- Maintain structural integrity and prevent unacceptable growth by reinforcing with a resistant material)
- Prevent initiation by modifying the residual stress distribution
- Prevent further growth by modifying the residual stress distribution
- Slow initiation and growth using improved water chemistry

Inspection of piping welds

- An industry campaign began in 1983 to qualify procedures and examiners for inspection of BWR piping welds
- Subsequent developments in ultrasonic examination techniques, equipment and procedures led to satisfactory results by 1988
- Evolved into Performance Demonstration Initiative (PDI) and ASME Appendix VIII

Summary: IGSCC countermeasures

- Replacement
- Weld overlay repair
- Residual stress modification
  - On new weldments
  - On shallow existing IGSCC
- Corrosion-resistant cladding
  - Internal, prior to welding
- Water chemistry improvements

Reasons to revise NUREG-0313

- Since 1984, losses in capacity factor have been dramatically reduced (see slide 6)
- IGSCC countermeasures are effective
  - Inspections are confirming little or no new crack initiation and growth in existing cracks
- Inspections result in radiation dose to personnel
  - There is a need to minimize inspections, particularly those that do not have an impact to safety

Scope

- All NUREG-0313 categories to be addressed
- Existing BWRVIP/BWROG work will be used where possible
  - Existing documents will be incorporated by reference where applicable
- Work is to be generic so that all BWRVIP/BWROG members benefit from the effort
- Product must be formatted to comply with BWRVIP after the 2000 transition

Approach

- Categories A through E will be evaluated for appropriate changes to inspection frequencies
- Service experience and physically based arguments will be used to evaluate performance
- IHSI and MSIP effectiveness and means to evaluate effectiveness of treatment will be incorporated

Approach (continued)

- HWC credit will be addressed, including how to evaluate effectiveness, using previous BWROG work and current BWRVIP efforts
- When appropriate, the results from BWRVIP crack growth studies for stainless steel and nickel-base alloys will be used

Approach (continued)

- Risk-Informed methodology will be used
  - Support the technical basis for new inspection frequencies
  - Guidance for the use of Risk-Informed tools

Schedule

- NUREG-0313
  - Document to be submitted to NRC by 9/99
  - Approval requested by 12/99
- BWROG programs (already submitted) related to HWC and weld overlays
  - Continuation of NRC review and approval in 1999
- Consider scheduler relief of current NUREG-0313 inspections pending resolution of this activity

Resources

- BWRVIP/BWROG Executive approval for this activity hinges on NRC commitment to provide necessary resources
- BWRVIP/BWROG activities must not be negatively impacted by this activity

Conclusions

- NRC requirements and IGSCC countermeasures have been effective in managing IGSCC
- A revision of the inspection frequencies in NUREG-0313 is warranted
- BWRVIP, in conjunction with efforts by the BWROG, is committed to develop a generic basis for revising NUREG-0313

H. Carpenter
O-7D4

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

April 22, 1999

Mr. Art Kenny
Intellectual Property Attorney
Intellectual Property Department
Electric Power Research Institute
3412 Hillview Avenue
Post Office Box 10412
Palo Alto CA 94303

SUBJECT: REQUEST FOR WITHHOLDING INFORMATION FROM PUBLIC DISCLOSURE TR-110172 "TECHNICAL JUSTIFICATION FOR THE EXTENSION OF THE INTERVAL BETWEEN INSPECTIONS OF WELD OVERLAY REPAIRS," FEBRUARY 1999

Dear Mr. Kenny

By letter dated February 17, 1999, you submitted a technical report, TR-110172, "Technical Justification for the Extension of the Interval between Inspections of Weld Overlay Repairs," and requested that it be withheld from public disclosure pursuant to 10 CFR 2.790.

In an affidavit dated February 17, 1999, you stated that the submitted information contains trade secrets of the Electric Power Research Institute (EPRI), which EPRI has developed, and should be considered exempt from mandatory public disclosure for the following reasons:

(a) The report is of a type customarily held in confidence by EPRI. It contains trade secrets and is held in confidence because to disclose it would prevent EPRI from licensing the report at fees, which would allow EPRI to recover its investment.

(b) If a competitor were able to publicly obtain the report, they would be able to use it commercially for profit and avoid spending the amount of money that EPRI spent in preparation of the report.

We have reviewed your letter and the material in accordance with the requirements of 10 CFR 2.790 and, on the basis of your statements, have determined that the submitted information sought to be withheld contains trade secrets or proprietary commercial information. Therefore the version of the submitted information marked as proprietary will be withheld from public disclosure pursuant to 10 CFR 2.790(b)(5) and Section 103(b) of the Atomic Energy Act of 1954, as amended.

Withholding from public inspection shall not affect the right, if any, of persons properly and directly concerned to inspect the document. If the need arises, we may send copies of this information to our consultants working in this area. We will, of course, ensure that the consultants have signed the appropriate agreements for handling proprietary information.

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If the basis for withholding this information from public inspection should change in the future such that the information could then be made available for public inspection, you should promptly notify the NRC. You should also understand that the NRC may have cause to review this determination in the future, for example, if the scope of a Freedom of Information Act request included your information. In all review situations, if the NRC need additional information from you or makes a dete.mination adverse to the above, you will be notified in advance of any public disclosure.

Sincerely,

Original Signed By:

James H. Wilson, Senior Project Manager
Generic Issues, Environmental, Financial and
Rulemaking Branch
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

Project No. 669

cc: See next page

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Sincerely,

James H. Wilson

James H. Wilson, Senior Project Manager
Generic Issues, Environmental, Financial and
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Project No. 669

Electric Power Research Institute

Project No. 669

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K

MEMORANDUM TO: Christopher I. Grimes, Chief
License Renewal and Standardization Branch
Division of Reactor Program Management

FROM: William H. Bateman, Chief */ra/*
Materials and Chemical Engineering Branch
Division of Engineering

SUBJECT: ACCEPTANCE FOR REFERENCING OF REPORT, "BWR VESSEL AND INTERNALS PROJECT, BWR LOWER PLENUM INSPECTION AND FLAW EVALUATION GUIDELINES (BWRVIP-47)," FOR COMPLIANCE WITH THE LICENSE RENEWAL RULE (10 CFR PART 54)

By letter dated December 30, 1997, as supplemented by letter dated March 3, 1999, the Boiling Water Reactor Vessel and Internals Project (BWRVIP) submitted the Electric Power Research Institute (EPRI) Topical Report TR-108727, "BWR Vessel and Internals Project, BWR Lower Plenum Inspection and Flaw Evaluation Guideline (BWRVIP-47)."

By letter dated October 13, 1999, the NRC staff issued its safety evaluation report (SER), which found the BWRVIP-47 guidelines acceptable for the current operating period of BWRs. The topical report submittal also included "Appendix A, BWR Lower Plenum. Demonstration of Compliance with the Technical Information Requirements of the License Renewal Rule (10 CFR 54.21)," for U.S. Nuclear Regulatory Commission (NRC) staff review. BWRVIP submitted a non-proprietary version of this document, TR-108727NP, on April 30, 1999.

As documented in the final safety evaluation report (FSER) enclosed with this letter, the NRC staff has completed its review of the proprietary version of BWRVIP-47. As indicated in the FSER, the staff found the topical report acceptable for licensees participating in BWRVIP to reference in a license renewal application to the extent specified and under the limitations delineated in the FSER. In order for licensees participating in BWRVIP to reference the report, they must commit to the accepted aging management programs defined therein, and complete the action items described in the FSER. In referencing the BWRVIP-47 report and meeting these limitations, an applicant will provide sufficient information for the staff to make a finding that there is reasonable assurance that the applicant will adequately manage the effects of aging so that the intended functions of the reactor vessel internal components covered by the scope of the report will be maintained consistent with the current licensing basis during the period of extended operation.

The staff does not intend to repeat its review of the matters described in the report and found acceptable in the FSER when the report appears as a reference in license renewal applications, except to ensure that the material presented applies to the specified plant.

CONTACT: C. E. Carpenter, EMCB/DE
415-2169

E/15

In accordance with the procedures established in NUREG-0390, "Topical Report Review Status," the staff requests that BWRVIP publish the accepted version of BWRVIP-47 within 90 days after receiving this letter. In addition, the published version shall incorporate this letter and the enclosed FSER between the title page and the abstract.

To identify the version of the report that was accepted by the staff, BWRVIP shall include "A" following the topical report number (e.g., BWRVIP-47-A).

Attachment: As stated

cc: See next page

In accordance with the procedures established in NUREG-0390, "Topical Report Review Status," the staff requests that BWRVIP publish the accepted version of BWRVIP-47 within 90 days after receiving this letter. In addition, the published version shall incorporate this letter and the enclosed FSER between the title page and the abstract.

To identify the version of the report that was accepted by the staff, BWRVIP shall include "A" following the topical report number (e.g., BWRVIP-47-A).

Attachment: As stated

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FINAL SAFETY EVALUATION REPORT

FOR

"BWR VESSEL AND INTERNALS PROJECT, BWR LOWER PLENUM INSPECTION AND FLAW EVALUATION GUIDELINES (BWRVIP-47)"

FOR COMPLIANCE WITH THE LICENSE RENEWAL RULE (10 CFR PART 54)

1.0 INTRODUCTION

1.1 Background

By letter dated December 30, 1997, as supplemented by letter dated March 3, 1999, the Boiling Water Reactor Vessel and Internals Project (BWRVIP) submitted the Electric Power Research Institute (EPRI) Topical Report TR-108727, "BWR Vessel and Internals Project, BWR Lower Plenum Inspection and Flaw Evaluation Guideline (BWRVIP-47)." The BWRVIP-47 report presents generic guidelines for the inspection of components in the lower plenum region of BWR vessels. Components addressed include control rod drive (CRD) housing and stub tube, control rod guide tube and orificed fuel support, and in-core housing, guide tube and dry tube assemblies. Shroud support legs and core delta pressure systems/standby liquid control system (SLC) are addressed in separate guidelines. The BWRVIP-47 guidelines provide recommendations for non-destructive evaluation (NDE) methods, inspection locations, and inspection frequencies. The BWRVIP-47 report also recommends acceptable methods for evaluating the structural integrity significance of flaws that are detected during these examinations. By letter dated October 13, 1999, the NRC staff issued its safety evaluation (SE) on the BWRVIP-47 report's acceptability for the current operating period.

Included within the topical report submittal, BWRVIP also provided the document, "Appendix A, BWR Lower Plenum. Demonstration of Compliance with the Technical Information Requirements of the License Renewal Rule (10 CFR 54.21)," for NRC staff review in accordance with the License Renewal Rule (10 CFR Part 54).

Section 54.21 of the License Renewal Rule requires, in part, that each application for license renewal contain an integrated plant assessment (IPA) and an evaluation of time-limited aging analyses (TLAA). The IPA must identify and list those structures and components subject to an aging management review and demonstrate that the effects of aging will be adequately managed so that their intended functions will be maintained consistent with the current licensing basis (CLB) for the period of extended operation. In addition, 10 CFR 54.22 requires that each application include any technical specification changes or additions necessary to manage the effects of aging during the period of extended operation as part of the renewal application.

If a license renewal applicant participating in BWRVIP confirms that the BWRVIP-47 report applies to it and that the results of the Appendix A IPA and TLAA evaluation are in effect at its plant, then no further review by the NRC staff of the issues described in the documents is necessary, except as specifically identified by the staff in its SE below. With this exception, such an applicant may rely on the BWRVIP-47 report for the demonstration required by Section 54.21(a)(3) with respect to the components and structures within the scope of the report.

Under such circumstances, the NRC staff intends to rely on the evaluation in this SE report to make the findings required by 10 CFR 54.29 with respect to a particular application.

1.2 Purpose

The staff reviewed the BWRVIP-47 report to determine whether its guidance will provide acceptable levels of quality for inspection and flaw evaluation of the subject safety-related RPV internal components. The review also considered compliance with the License Renewal Rule in order to allow applicants the option of incorporating the BWRVIP-47 guidelines by reference in a plant-specific IPA and associated time-limited aging analyses (TLAA).

1.3 Organization of this Report

Because the BWRVIP report is proprietary, this SE was written so as not to repeat information contained in the propriety portions of the report. The staff does not discuss in any detail the provisions of the guidelines nor the parts of the guidelines it finds acceptable. A brief summary of the contents of the BWRVIP-47 report is given in Section 2.0 of this SE, with the NRC staff's evaluation presented in Section 3.0. The conclusions are summarized in Section 4.0. The presentation of the evaluation is structured according to the organization of the BWRVIP-47 report.

2.0 SUMMARY OF BWRVIP-47 REPORT

The BWRVIP-47 report and its Appendix A contain a generic evaluation of the management of the effects of aging of the subject safety-related RPV internal components so that the intended functions will be maintained consistent with the CLB for the period of extended operation. This evaluation applies to BWR applicants who have committed to implementing the BWRVIP-47 report and want to incorporate the report and Appendix A by reference into a plant-specific IPA and associated TLAA.

The BWRVIP-47 report addresses the following topics:

1. <u>Component Description and Function</u> – The various lower plenum components are described in considerable detail by a series of illustrations along with brief descriptions of each component's function and materials/welding characteristics. Differences among the various models of BWRs (BWR/2, BWR-3-5, and BWR/6) are identified.

2. <u>Susceptibility Factors</u> – The various types of material degradation mechanisms (fatigue, stress corrosion cracking, age embrittlement) that could impact lower plenum components are described. Materials, stress, and environmental factors are described in general terms, and followed by specific references to actual occurrences for each degradation mechanism relative to plant operating experience for particular mechanisms and components.

3. <u>Potential Failure Locations and Safety Consequences</u> - Each of the lower plenum components are addressed from the standpoint of inspection history, future susceptibility to degradation, and consequences of failures in terms of component functions and plant safety. Based in these qualitative considerations, the BWRVIP-47 report makes

recommendations as to the need for inspections for each of the lower plenum components.

4. Background and Inspection History – Data on service related failures of components are summarized. The major sources of such data are the various GE SILs and Rapid Information Communication Service Information Letters (RICSILs). Another source of data is the reported findings from inspections of lower plenum components performed at plants in the U.S. and Spain. Inspection requirements are evaluated according to the following four criteria: 1) the potential consequences of a failure to plant safety, 2) the ability of leak monitoring to detect degradation as a complementary measure to inservice inspections, 3) field cracking history as a means to identify the most likely locations for material degradation, and 4) the extent to which results from prior inspections provide a high level of confidence that no degradation mechanisms are active for the components of concern.

5. BWRVIP Inspection Guidelines – The guidelines recommend the specific locations, NDE methods, and inspection frequencies for examinations of lower plenum components. The recommended NDE methods are limited to visual examinations, with reference made to the BWRVIP-03 report (Reference 3) for detailed requirements for implementing these visual examinations. The BWRVIP-47 report recommends only a limited number of inspections for the lower plenum components, based mainly on the relatively good service experience to date that indicates no evidence of generic cracking. The relatively small safety consequences of structural failures and the ability of leak detection to provide an early indication of structural degradation are also cited to justify the recommended level of inspection. The plant-specific recommendations focus on a one-time baseline inspection at each BWR plant, with no periodic re-inspections being required unless justified by evidence of the actual occurrence of degradation. The scope of sample inspections is expanded to address similar components if flaws are found.

5. Loads – This section briefly states that the loads used in fracture mechanics evaluations to address the effects of detected flaws on structural integrity should be based on the plant design and licensing basis. The various types of loads (e.g., pressures, seismic, etc.) of concern are listed.

Appendix A discusses the following topics:

2.1 Identification of Structures and Components Subject to an Aging Management Review

10 CFR 54.21(a)(1) requires that an IPA identify and list those structures and components within the scope of license renewal that are subject to an aging management review (AMR). Structures and components subject to an AMR shall encompass those structures and components that (1) perform an intended function, as described in 10 CFR 54.4, without moving parts or without a change in configuration or properties and (2) are not subject to replacement based on a qualified life or specified time period. These structures and components are also referred to as "passive" and "long-lived" structures and components, respectively.

In Section 2.0 of the BWRVIP-47 report, BWRVIP describes the intended functions of the BWR lower plenum. The components include the control rod guide tubes (CRGT) and orificed fuel

support (OFS), control rod drive (CRD) housings (CRDH) and stub tubes (CRDST), incore housing, guide tube and dry tube assemblies. The primary function of the CRGT, CRDH and CRDST is to facilitate control rod movement into the reactor core to achieve reactivity control as well as shutdown conditions. The OFS supports the weight of the fuel assemblies and distributes the core flow into the fuel bundles. The incore housing and guide tubes laterally supports and positions the instrumentation tubes such that the Neutron Monitoring System (NMS) has access to the interior of the reactor core.

In Appendix A, BWRVIP identified the passive and long-lived components as required by 10 CFR 54.21(a)(1). BWRVIP noted that all of the components in the lower plenum are subject to AMR.

2.2 Effects of Aging

BWRVIP identified the aging mechanisms and aging effects for the lower plenum using the guidance from NUMARC 90-02, "BWR Reactor Pressure Vessel License Renewal Industry Report," Revision 1, dated August 1992, and the resolution to the NRC's questions on that Industry Report. BWRVIP also used NUREG-1557, "Summary of Technical Information and Agreements from Nuclear Management and Resources Council Industry Reports Addressing License Renewal," dated October 1996, to correlate the aging effects and their associated aging mechanisms. Using these reports, BWRVIP determined that crack initiation and growth is the only aging effect that requires AMR for the lower plenum.

In Section 2.0 of the BWRVIP-47 report, BWRVIP discussed the causes of crack initiation and growth and provided a susceptibility assessment, and also discussed the susceptibility factors of environment, materials, and stress state. BWRVIP's review of the degradation history determined that there have been no significant field cracking of the lower plenum components.

2.3 Aging Management Programs

10 CFR 54.21(a)(3) requires that the applicant demonstrate, for each component identified, that the effects of aging will be adequately managed so that the intended function will be maintained consistent with the CLB for the period of extended operation.

In Section 3.0 of the BWRVIP-47 report, BWRVIP discussed the inspection strategy to be used for ensuring that cracks that might occur in the lower plenum are detected in a timely manner. The following components are recommended for visual (modified VT-1 or VT-3) examination: 1) CRGT-1 - CRD guide tube sleeve-to-alignment lug weld, 2) CRGT-2 - CRD guide tube body-to-sleeve weld and CRGT-3 - CRD guide tube base to body weld, 3) FS/GT-ARPIN-1 - guide tube and fuel support alignment pin-to-core plate weld, and the pin itself. No inspections are required for the other locations in the CRD housing/stub tube/guide tube/fuel support assemblies and the in-core housing/guide tube/dry tube assemblies. BWRVIP concluded that both its inspection program and plant-specific considerations will result in verification of the structural integrity in the CLB for the subject safety-related RPV internal components.

2.4 Time-Limited Aging Analyses

10 CFR 54.21(1)(c) requires that each application for license renewal contain an evaluation of TLAA as defined in 10 CFR 54.3. TLAA are those licensee calculations and analyses that:

(1) involve the lower plenum within the scope of license renewal,

(2) consider the effects of aging,

(3) involve time-limited assumptions defined by the current operating term,

(4) were determined to be relevant by the licensee in making a safety determination,

(5) involve conclusions or provide the basis for conclusions related to the capability of the lower plenum to perform their intended function, and

(6) are contained or incorporated by reference in the CLB.

If a plant-specific analysis identified by an applicant meets all six criteria above, the analysis will be considered a TLAA for license renewal and evaluated by the applicant.

Based on this criteria there are no generic TLAA issues that require evaluation for the lower plenum internals. BWRVIP considered this issue under 10 CFR 54.21(c)(1)(ii) by projecting the analysis to the end of the period of extended operation. BWRVIP stated that the typical cumulative usage factors (CUF) may exceed the threshold specified in NUMARC 90-02 for some plants' lower plenum boundary components during the current and extended operating periods. For those plants, a plant-specific fatigue analysis will be needed.

3.0 STAFF EVALUATION

The staff reviewed the BWRVIP-47 report to determine if it demonstrated that the effects of aging on the subject reactor vessel components covered by the report will be adequately managed so that the components' intended functions will be maintained consistent with the CLB for the period of extended operation, in accordance with 10 CFR 54.21(a)(3). This is the last step in the IPA described in 10 CFR 54.21(a).

Besides the IPA, Part 54 requires an evaluation of TLAA in accordance with 10 CFR 54.21(c). The staff reviewed the BWRVIP-47 report to determine if the TLAA covered by the report were evaluated for license renewal in accordance with 10 CFR 54.21(c)(1).

3.1 Structures and Components Subject to Aging Management Review

The staff agrees that the lower plenum components are subject to AMR because they perform intended functions without moving parts or without a change in configuration or properties. The staff concludes that BWR applicants for license renewal must identify the appropriate subject safety-related RPV internal components as subject to AMR to meet the applicable requirements of 10 CFR 54.21(a)(1).

3.2 Intended Functions

The staff agrees that the intended functions of the lower plenum are as stated. The primary function of the CRGT, CRDH and CRDST is to facilitate control rod movement into the reactor core to achieve reactivity control as well as shutdown conditions. The OFS supports the weight of the fuel assemblies and distributes core flow into the fuel bundles. The incore housing and guide tubes laterally supports and positions the instrumentation tubes such that the Neutron Monitoring System (NMS) has access to the interior of the reactor core.

3.3 Effects of Aging

The information necessary to demonstrate compliance with the requirements of the license renewal rule 10 CFR 54.21 is provided in Appendix A of BWRVIP-47. The BWR Reactor Pressure Vessel Industry Report NUMARC 90-02, Revision 1, August 1992, and the resolution to the NRC's questions on that industry report were used to identify the aging mechanisms for the lower plenum. If the industry report concluded that the aging mechanism is significant then the aging mechanism was included in the AMR. Using this methodology it was determined that crack initiation and growth is the only aging effect that required AMR.

Accordingly, NUREG-1557 states that crack initiation and growth are the aging effects that need to be considered. The staff agrees that this mechanism is the only one applicable to the lower plenum.

3.4 Aging Management Programs

The staff evaluated BWRVIP's aging management program (AMP) to determine if it contains the following 10 elements constituting an adequate AMP for license renewal: scope of program, preventive actions, parameters monitored or inspected, detection of aging effects, monitoring and trending, acceptance criteria, corrective actions, confirmation process, administrative controls, and operating experience.

The staff's evaluation of the BWRVIP-47 report was transmitted by letter dated October 13, 1999, to Carl Terry, BWRVIP Chairman. For the reasons set forth in that SER, the staff concluded that the inspection strategy and evaluation methodologies discussed in the BWRVIP-47 report are acceptable. Implementation of the above inspection program provides reasonable assurance that crack initiation and growth will be adequately managed such that the intended functions of the subject safety-related RPV internal components will be maintained consistent with the CLB in the extended operating period.

3.5 Time Limited Aging Analyses

One of the mechanisms that may potentially cause degradation of the ferritic-austenitic stainless steel lower plenum components is fatigue. Possible sources of fatigue in the lower plenum internals are low cycle fatigue resulting from system thermal and pressure cycling, or high cycle fatigue due to flow-induced vibrations (FIV). In those instances where a fatigue analysis was required in the case of low cycle fatigue, the CUF were extended to 40 years and determined to be less than the Code allowable value of 1.0. High cycle fatigue has not been identified as a significant issue for the lower plenum based on many reactor years with no vibration failures.

4.0 CONCLUSIONS

The staff has reviewed the BWRVIP-47 report submitted by BWRVIP. On the basis of its review, the staff concludes that the BWRVIP-47 report provides an acceptable demonstration that the BWRVIP member utilities referencing this topical report will adequately manage the aging effects of reactor vessel components within the scope of the report, with the exception of the noted renewal applicant action items set forth in Section 4.1 below, so that there is reasonable assurance that the lower plenum internals will perform their intended functions in accordance with the CLB during the period of extended operation. The staff also concludes that, upon completion of the renewal applicant action items, the BWRVIP-47 report provides an acceptable evaluation of TLAAs for the lower plenum for the BWRVIP member utilities for the period of extended operation.

Any BWRVIP member utility may reference this report in a license renewal application to satisfy the requirements of (1) 10 CFR 54.21(a)(3) for demonstrating that the effects of aging on the reactor vessel components within the scope of this topical report will be adequately managed and (2) 10 CFR 54.21(c)(1) for demonstrating the appropriate findings regarding evaluation of TLAAs for the lower plenum for the period of extended operation. The staff also concludes that, upon completion of the renewal applicant action items set forth in Section 4.1 below, referencing this topical report in a license renewal application and summarizing in an FSAR supplement the AMPs and the TLAA evaluations contained in this topical report will provide the staff with sufficient information to make the necessary findings required by Sections 54.29(a)(1) and (a)(2) for components within the scope of this topical report.

4.1 Renewal Applicant Action Items

The following are license renewal applicant action items to be addressed in the plant-specific license renewal application when incorporating the BWRVIP-47 report in a renewal application:

(1) The license renewal applicant is to verify that its plant is bounded by the topical report. Further, the renewal applicant is to commit to programs described as necessary in the BWRVIP report to manage the effects of aging on the functionality of the lower plenum during the period of extended operation. Applicants for license renewal will be responsible for describing any such commitments and identifying how such commitments will be controlled. Any deviations from the aging management programs within the BWRVIP-47 report described as necessary to manage the effects of aging during the period of extended operation and to maintain the functionality of the reactor vessel components or other information presented in the report, such as materials of construction, will have to be identified by the renewal applicant and evaluated on a plant-specific basis in accordance with 10 CFR 54.21(a)(3) and (c)(1).

(2) 10 CFR 54.21(d) requires that an FSAR supplement for the facility contain a summary description of the programs and activities for managing the effects of aging and the evaluation of TLAA for the period of extended operation. Those applicants for license renewal referencing the BWRVIP-47 report for the lower plenum shall ensure that the programs and activities specified as necessary in the BWRVIP-47 document are summarily described in the FSAR supplement.

(3) 10 CFR 54.22 requires that each application for license renewal include any technical specification changes (and the justification for the changes) or additions necessary to manage the effects of aging during the period of extended operation as part of the renewal application. In its Appendix A to the BWRVIP-47 report, BWRVIP stated that there are no generic changes or additions to technical specifications associated with the lower plenum as a result of its AMR and that the applicant will provide the justification for plant-specific changes or additions. Those applicants for license renewal referencing the BWRVIP-47 report for the lower plenum shall ensure that the inspection strategy described in the BWRVIP-47 report does not conflict or result in any changes to their technical specifications. If technical specification changes do result, then the applicant should ensure that those changes are included in its application for license renewal.

(4) Due to fatigue of the subject safety-related components, applicants referencing the BWRVIP-47 report for license renewal should identify and evaluate the projected CUF as a potential TLAA issue. This issue is discussed in more detail in Section 3.5 of this report.

5.0 REFERENCES

1. NUREG-1557, Summary of Technical Information and Agreements from Nuclear Management and Resources Council Industry Reports Addressing License Renewal, October 1996.

2. Carl Terry, BWRVIP, to USNRC, "BWR Vessel and Internals Project, BWR Lower Plenum Inspection and Flaw Evaluation Guideline (BWRVIP-47)," EPRI Report TR-108727, dated December 1997.

3. C. E. Carpenter, USNRC, to Carl Terry, BWRVIP, "Safety Evaluation of BWR Vessel and Internals Project Report, BWR Lower Plenum Inspection and Flaw Evaluation Guideline (BWRVIP-47)," dated October 13, 1999.

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

August 10, 1999

MEMORANDUM TO: Christopher I. Grimes, Chief
License Renewal and Standardization Branch
Division of Reactor Program Management

FROM: William H. Bateman, Chief William H. Bateman
Materials and Chemical Engineering Branch
Division of Engineering

SUBJECT: SAFETY EVALUATION OF THE BWRVIP VESSEL AND INTERNALS PROJECT, EPRI REPORT TR-107286, FOR COMPLIANCE WITH LICENSE RENEWAL REQUIREMENTS

The NRC staff has completed its review of the Electric Power Research Institute (EPRI) proprietary report TR-107286, "BWR Vessel and Internals Project, BWR Standby Liquid Control System / Core Plate ΔP Inspection and Flaw Evaluation Guidelines (BWRVIP-27)," transmitted by letter dated April 25, 1997, and supplemented by letter dated July 13, 1998. The BWRVIP-27 report provides generic guidelines intended to present the appropriate inspection recommendations to assure safety function integrity of the subject safety-related reactor pressure vessel (RPV) internal components.

In addition, a separate document, "Appendix B, BWR Standby Liquid Control System / Core Plate ΔP Inspection And Flaw Evaluation Guidelines, Demonstration of Compliance with the Technical Information Requirements of the License Renewal Rule (10 CFR 54.21)," was submitted by letter dated May 15, 1998. The purpose of Appendix B is to demonstrate that the BWRVIP-27 report provides the necessary information to comply with the technical information requirements pursuant to 10 CFR 54.21(a) and §54.22, and the NRC's findings under 10 CFR 54.29(a) of the License Renewal Rule.

By letter dated April 27, 1999, the NRC staff issued its safety evaluation (SE) on the BWRVIP-27 report's acceptability for the current operating period. The attached SE is for the license renewal period and finds that the BWRVIP-27 report ensures: 1) that its guidance is acceptable for the inspection and flaw evaluation of the subject safety-related RPV internal components and 2) compliance with paragraph 54.21 of the License Renewal Rule. The EMCB staff has concluded that licensee implementation of the guidelines in the BWRVIP-27 report, as supplemented, will provide an acceptable level of quality for inspection and flaw evaluation of the safety-related components addressed throughout the renewal term. In addition, the EMCB staff found the report acceptable for referencing in license renewal applications for BWR licensees who have committed to the accepted aging management programs in the BWRVIP-27 report.

Attachment: As stated

cc: see next page

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FINAL SAFETY EVALUATION REPORT FOR BWR STANDBY LIQUID CONTROL SYSTEM / CORE PLATE ΔP INSPECTION AND FLAW EVALUATION GUIDELINES (BWRVIP-27) FOR COMPLIANCE WITH LICENSE RENEWAL REQUIREMENTS

1.0 INTRODUCTION

1.1 Background

By letter dated April 25, 1997, the Boiling Water Reactor Vessel and Internals Project (BWRVIP) submitted the Electric Power Research Institute (EPRI) proprietary report TR-107286, "BWR Vessel and Internals Project, BWR Standby Liquid Control System / Core Plate ΔP Inspection And Flaw Evaluation Guidelines (BWRVIP-27)," April 1997. This report was supplemented by a letter dated July 13, 1998, which was in response to the staff's request for additional information (RAI), dated December 14, 1997. The BWRVIP-27 report provides generic guidelines intended to present the appropriate inspection recommendations to assure safety function integrity of the subject safety-related reactor pressure vessel (RPV) internal components. It also provides design information on the core plate ΔP/Standby Liquid Control (ΔP/SLC) system, geometries, weld locations, and potential failure locations for the several categories of boiling water reactors (BWR/2 through BWR/6). By letter dated April 27, 1999, the NRC staff issued its safety evaluation (SE) on the BWRVIP-27 report's acceptability for the current operating period.

By letter dated May 15, 1998, the BWRVIP submitted a separate document, "Appendix B, BWR Standby Liquid Control System / Core Plate ΔP Inspection and Flaw Evaluation Guidelines, Demonstration of Compliance with the Technical Information Requirements of the License Renewal Rule (10 CFR 54.21)," for NRC staff review in accordance with the License Renewal Rule (10 CFR Part 54).

Paragraph 54.21 of the License Renewal Rule requires, in part, that each application for license renewal contain an integrated plant assessment (IPA) and an evaluation of time-limited aging analyses (TLAA). The IPA must identify and list those structures and components subject to an aging management review and demonstrate that the effects of aging will be adequately managed. In addition, §54.22 requires that each application include any technical specification changes or additions necessary to manage the effects of aging during the period of extended operation as part of the renewal application.

If a license renewal applicant confirms that the BWRVIP-27 document applies to its plant's current licensing design basis (CLB) and that the results of the Appendix B IPA and TLAA evaluation are in effect at its plant, then no further review by the NRC staff of the issues described in the documents is necessary, unless specifically identified by the staff in its safety evaluation below.

ATTACHMENT

1.2 Purpose

The staff reviewed the BWRVIP-27 report to determine whether its guidance will provide acceptable levels of quality for inspection and flaw evaluation of the subject safety-related RPV internal components. The review also considered compliance with the License Renewal Rule in order to allow licensees the option of incorporating the BWRVIP-27 guidelines by reference in a plant-specific integrated plant assessment (IPA) and associated time-limited aging analyses (TLAAs).

1.3 Organization of this Report

Because the BWRVIP report is proprietary, this SE was written so as not to repeat information contained in the propriety portions of the report. The staff does not discuss in any detail the provisions of the guidelines nor the parts of the guidelines it finds acceptable. A brief summary of the contents of the BWRVIP-27 report is given in Section 2.0 of this SE, with the NRC staff's evaluation presented in Section 3.0. The conclusions are summarized in Section 4.0. The presentation of the evaluation is structured according to the organization of the BWRVIP-27 report.

2.0 SUMMARY OF BWRVIP-27 REPORT

The BWRVIP-27 report and its Appendix B contain a generic evaluation of the management of the effects of aging of the subject safety-related RPV internal components so that the intended functions will be maintained for the period of extended operation. This evaluation applies to BWR licensees who have committed to implementing the BWRVIP-27 report and want to incorporate the report and Appendix B by reference into a plant-specific IPA and associated TLAAs.

The BWRVIP-27 report addresses the following topics:

1. Component Description and Function – The various core plate ΔP/SLC system configurations are described in detail by a series of illustrations along with brief descriptions of each configuration's function and characteristics. Differences among the various models of BWRs (BWR/2, BWR/3-5 and BWR/6) are identified.

2. Susceptibility Factors – The various types of material degradation mechanisms (fatigue, stress corrosion cracking, age embrittlement) that could impact the ΔP/SLC internals are characterized. Materials, stress, and environmental factors are described in general terms, and followed by specific references to actual occurrences for each degradation mechanism relative to plant operating experience for particular mechanisms and components.

3. Potential Failure Locations and Safety Consequences - Each of the vessel penetration configurations are addressed from the standpoint of inspection history, future susceptibility to degradation, and consequences of failures in terms of component functions and plant safety. Based in these qualitative considerations, the BWRVIP-27 report makes recommendations as to the need for inspections for each of the ΔP/SLC system configurations.

4. Boron Mixing and Leakage Considerations - The mixing and leakage issues related to the degradation of ΔP/SLC internals are qualitatively and quantitatively addressed in this section.

5. Background and Inspection History – Data on service related failures of components are summarized. The major sources of such data are the various GE SILs and rapid information communication service information letters (RICSILs).

6. BWRVIP Inspection Guidelines – The guidelines recommend the specific locations, NDE methods, and inspection frequencies for examinations of core plate ΔP/SLC internals. The BWRVIP-27 report recommends that for most configurations, the current ASME inspection requirements be followed. For some configurations, however, an additional ultrasonic (UT) examination is recommended.

Appendix B discusses the following topics:

2.1 Identification of Structures and Components Subject to an Aging Management Review

10 CFR 54.21(a)(1) requires that an IPA identify and list those structures and components within the scope of license renewal that are subject to an aging management review. Structures and components subject to an aging management review shall encompass those structures and components that (1) perform an intended function, as described in §54.4, without moving parts or without a change in configuration or properties and (2) are not subject to replacement based on a qualified life or specified time period. These structures and components are also referred to as "passive" and "long-lived" structures and components, respectively.

In Section 2.0 of the BWRVIP-27 report, the BWRVIP describes the intended function of the core plate ΔP/Standby Liquid Control (ΔP/SLC) system . The function is to provide direct redundancy to the control rod system to achieve safe shutdown of the reactor. This is accomplished via injection of sodium pentaborate into the bottom head region of the vessel.

In Appendix B, the BWRVIP identified the passive and long-lived components as required by 10 CFR 54.21(a)(1). The BWRVIP noted that the ΔP/SLC vessel penetration/nozzle and safe-end extensions are subject to aging management review.

2.2 Effects of Aging

The BWRVIP identified the aging mechanisms and aging effects for the ΔP/SLC vessel penetration/nozzle and safe-end extensions using the guidance from NUMARC 90-02, "BWR Reactor Pressure Vessel License Renewal Industry Report," Revision 1, dated August 1992, and the resolution to the NRC's questions on the Industry Report. The BWRVIP also used NUREG-1557, "Summary of Technical Information and Agreements from Nuclear Management and Resources Council Industry Reports Addressing License Renewal," dated October 1996, to correlate the aging effects and their associated aging mechanisms. Using these reports, the BWRVIP determined that crack initiation and growth is the only aging effect that requires aging management review for the ΔP/SLC vessel penetration/nozzle and safe-end extensions.

In Section 2.0 of the BWRVIP-27 report, the BWRVIP discussed the causes of crack initiation and growth and provided a susceptibility assessment, and also discussed the susceptibility factors of environment, materials, and stress state. The BWRVIP's review of the degradation history determined that there have been no leaks due to cracking at penetration welds.

2.3 Aging Management Programs

10 CFR 54.21(a)(3) requires that the applicant demonstrate, for each component identified, that the effect of aging will be adequately managed so that the intended function will be maintained consistent with the CLB for the period of extended operation.

In Section 3.0 of the BWRVIP-27 report, the BWRVIP discussed the inspection strategy to be used for ensuring that cracks that might occur in the ΔP/SLC vessel penetration/nozzle and safe-end extensions are detected in a timely manner. The program requires an ASME Section XI VT-2 visual inspection for leakage and ASME Section XI, IWB-2500, Category B-D volumetric inspections (for low alloy steel nozzle configurations). The BWRVIP concluded that both its inspection program and plant-specific considerations will result in verification of the structural integrity in the CLB for the subject safety-related RPV internal components.

2.4 Time-Limited Aging Analyses (TLAAs)

10 CFR 54.21(1)(c) requires each application for license renewal contain an evaluation of time-limited aging analyses (TLAA) as defined in §54.3. TLAAs are those licensee calculations and analyses that:

(1) involve the instrument penetrations within the scope of license renewal,

(2) consider the effects of aging,

(3) involve time-limited assumptions defined by the current operating term,

(4) were determined to be relevant by the licensee in making a safety determination,

(5) involve conclusions or provide the basis for conclusions related to the capability of the instrument penetrations to perform their intended function, and

(6) are contained or incorporated by reference in the CLB.

If a plant-specific analysis identified by an applicant meets all six criteria above, the analysis will be considered a TLAA for license renewal and evaluated by the applicant.

The susceptibility of the low alloy nozzle designs to fatigue results in a potential TLAA issue. The BWRVIP evaluated this issue using the requirements in 10 CFR 54.21(c)(1)(ii). It projected the analysis to the end of the period of extended operation and found that the typical cumulative usage factors are below the 0.4 threshold specified in NUMARC 90-02 for all the nozzle designs during the current and extended operating periods.

3.0 STAFF EVALUATION

The staff reviewed the BWRVIP-27 report submitted by the BWRVIP to determine if it demonstrated that the effects of aging on the subject reactor vessel components covered by the report will be adequately managed so that the components' intended functions will be

maintained consistent with the CLB for the period of extended operation, in accordance with 10 CFR 54.21(a)(3). This is the last step in the IPA described in 10 CFR 54.21(a).

Besides the IPA, Part 54 requires an evaluation of TLAAs in accordance with 10 CFR 54.21(c). The staff reviewed the BWRVIP-27 report submitted by the BWRVIP to determine if the TLAAs covered by the report were evaluated for license renewal in accordance with 10 CFR 54.21(c)(1).

3.1 Structures and Components Subject to Aging Management Review

The staff agrees that the ΔP/SLC vessel penetration/nozzle and safe-end extensions are subject to aging management review because they perform intended functions without moving parts or without a change in configuration or properties. The staff concludes that BWR applicants for license renewal must identify the appropriate subject safety-related RPV internal components as subject to aging management review to meet the applicable requirements of 10 CFR 54.21(a)(1).

3.2 Intended Functions

The staff agrees that the intended functions of the core plate ΔP/SLC system are as stated. The function is to provide direct redundancy to the control rod system to achieve safe shutdown of the reactor. This is accomplished via injection of sodium pentaborate into the bottom head region of the vessel

3.3 Effects of Aging

The information necessary to demonstrate compliance with the requirements of the license renewal rule 10 CFR 54.21 is provided in Appendix B of BWRVIP-27. The BWR Reactor Pressure Vessel Industry Report NUMARC 90-02, Revision 1, August 1992, and the resolution to to the NRC's questions on the Industry Report were used to identify the aging mechanisms for the ΔP/SLC vessel penetration/nozzle and safe-end extensions. If the industry report concluded that the aging mechanism is significant then the aging mechanism was included in the aging management review. Using this methodology it was determined that crack initiation and growth is the only aging effect that required aging management review

The staff agrees that crack initiation and growth are the aging effects that need to be considered. This mechanism was the only one applicable to the subject safety-related RPV internal components as stated in NUREG 1557 (2).

3.4 Aging Management Programs

The staff evaluated the BWRVIP's aging management program to determine if it contains the following 10 elements constituting an adequate aging management program for license renewal: scope of program, preventive actions, parameters monitored or inspected, detection of aging effects, monitoring and trending, acceptance criteria, corrective actions, confirmation process, administrative controls, and operating experience.

The staff's evaluation of the BWRVIP-27 report was transmitted by letter dated April 27, 1999, to Carl Terry, BWRVIP Chairman. The staff concluded in that SER that the inspection strategy and evaluation methodologies discussed in the BWRVIP-27 report are acceptable. Implementation of the above inspection program provides reasonable assurance that crack initiation and growth will be adequately managed such that the intended functions of the subject safety-related RPV internal components will be maintained consistent with the CLB in the extended operating period.

3.5 Time Limited Aging Analyses

One of the mechanisms which can cause degradation of the low alloy steel nozzle designs is fatigue. During the initial design process, the influence of fatigue on the nozzle designs was considered. In a majority of instances, the nozzle designs were determined to be exempt from the requirements of a detailed ASME Code Section III fatigue analysis. In those instances where a fatigue analysis was required, the fatigue usage factors were extended to 60 years and determined to be very low when compared to the Code allowable of 1.0.

4.0 CONCLUSIONS

The staff has reviewed the subject BWRVIP-27 report submitted by the BWRVIP. On the basis of its review, the staff concludes that the BWRVIP-27 report provides an acceptable demonstration that the aging effects of reactor vessel components within the scope of this topical report will be adequately managed for the BWRVIP member utilities with the exception of the noted renewal applicant action items set forth in Section 4.1 below, so that there is reasonable assurance that the ΔP/SLC vessel penetration/nozzle and safe-end extensions will perform their intended functions in accordance with the CLB. The staff also concludes that, upon completion of the renewal applicant action items, the BWRVIP-27 report provides an acceptable evaluation of time-limited aging analyses for the ΔP/SLC vessel penetration/nozzle and safe-end extensions for the BWRVIP member utilities for the period of extended operation.

Any BWRVIP member utility may reference this report in a license renewal application to satisfy the requirements of (1) 10 CFR 54.21(a)(3) for demonstrating that the effects of aging on the reactor vessel components within the scope of this topical report will be adequately managed and (2) 10 CFR 54.21(c)(1) for demonstrating that appropriate findings be made regarding evaluation of time-limited aging analyses for the ΔP/SLC vessel penetration/nozzle and safe-end extensions for the period of extended operation. The staff also concludes that, upon completion of the renewal applicant action items set forth in Section 4.1 below, referencing this topical report in a license renewal application and summarizing in an FSAR supplement the aging management programs and the TLAA evaluations contained in this topical report will provide the staff with sufficient information to make the necessary findings required by Sections 54.29(a)(1) and (a)(2) for components within the scope of this topical report.

4.1 Renewal Applicant Action Items

The following are license renewal applicant action items to be addressed in the plant-specific license renewal application when incorporating the BWRVIP-27 report in a renewal application:

(1) The license renewal applicant is to verify that its plant is bounded by the topical report. Further, the renewal applicant is to commit to programs described as necessary in the BWRVIP report to manage the effects of aging during the period of extended operation on the functionality of the ΔP/SLC vessel penetration/nozzle and safe-end extensions. Applicants for license renewal will be responsible for describing any such commitments and identifying how such commitments will be controlled. Any deviations from the aging management programs within this BWRVIP report described as necessary to manage the effects of aging during the period of extended operation and to maintain the functionality of the reactor vessel components or other information presented in the report, such as materials of construction, will have to be identified by the renewal applicant and evaluated on a plant-specific basis in accordance with 10 CFR 54.21(a)(3) and (c)(1).

(2) 10 CFR 54.21(d) requires that an FSAR supplement for the facility contain a summary description of the programs and activities for managing the effects of aging and the evaluation of TLAAs for the period of extended operation. Those applicants for license renewal referencing the BWRVIP-27 report for the ΔP/SLC vessel penetration/nozzle and safe-end extensions shall ensure that the BWRVIP-27 document is included in a FSAR supplement that summarizes the description of the programs and activities for managing the effects of aging.

(3) 10 CFR 54.22 requires that each application for license renewal include any technical specification changes (and the justification for the changes) or additions necessary to manage the effects of aging during the period of extended operation as part of the renewal application. In its Appendix B to the BWRVIP-27 report, the BWRVIP stated that there are no generic changes or additions to technical specifications associated with the ΔP/SLC vessel penetration/nozzle and safe-end extensions as a result of its aging management review and that the applicant will provide the justification for plant-specific changes or additions. Those applicants for license renewal referencing BWRVIP-27 for the ΔP/SLC vessel penetration/nozzle and safe-end extensions shall ensure that the inspection strategy described in the BWRVIP-27 document does not conflict or result in any changes to their technical specifications. If technical specification changes do result, then the applicant should ensure that those changes are included in its application for license renewal.

5.0 REFERENCES

1. Carl Terry, BWRVIP, to USNRC, "BWR Vessel and Internals Project, BWR Standby Liquid Control System / Core Plate ΔP Inspection And Flaw Evaluation Guidelines (BWRVIP-27)," EPRI Report TR-107286, dated April 1997.

2. NUREG-1557, Summary of Technical Information and Agreements from Nuclear Management and Resources Council Industry Reports Addressing License Renewal, October 1996.

3. C. E. Carpenter, USNRC, to Carl Terry, BWRVIP, "Safety Evaluation of BWR Vessel and Internals Project Report, BWR Standby Liquid Control System / Core Plate ΔP Inspection And Flaw Evaluation Guidelines (BWRVIP-27)," dated April 27, 1999.

4. C. E. Carpenter, USNRC, to Carl Terry, BWRVIP, "Propriety Request for Additional Information - Review of BWR Vessel and Internals Project Report, BWR Standby Liquid Control System / Core Plate ΔP Inspection And Flaw Evaluation Guidelines (BWRVIP-27)," dated December 14, 1997.

5. Carl Terry, BWRVIP, to USNRC, "BWRVIP Response to NRC Request for Additional Information on BWRVIP-27," July 13, 1998.