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THE B&W OWNERS GROUP

MATERIALS COMMITTEE

MASTER INTEGRATED REACTOR VESSEL SURVEILLANCE PROGRAM

Babcock & Wilcox
a McDermott company

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MASTER INTEGRATED REACTOR VESSEL SURVEILLANCE PROGRAM

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Prepared for

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CONTENTS

APPENDICES

Contents (Cont'd)

APPENDICES (Cont'd) Page

List of Tables

Table

Tables (Cont'd)

Tables (Cont'd)

Tables (Cont'd)

List of Figures

Figures (Cont'd)

Figures (Cont'd)

Figure | Page

1. INTRODUCTION

This document describes a Master Integrated Reactor Vessel Surveillance Program (MIRVP) designed to provide the data required to insure the continued licensability of 17 reactor vessels fabricated by Babcock and Wilcox (B&W). These reactor vessels include 8 B&W-designed 177-Fuel Assembly (FA) plants and 9 Westinghouse-designed plants with B&W-fabricated reactor vessels.

The program builds on the integrated surveillance program developed by the B&W Owners Group for the B&W 177-FA plants[1] and is organized as shown in Figure 1-1. This program will obtain data from several sources, redundant irradiation sites, and, as a contingency, a test reactor irradiation program. As shown in Figure 1-1, this program includes integration of the following phases:

a. plant-specific reactor vessel surveillance programs from 17 reactor vessels

b. the existing supplemental B&W Owners Group irradiation capsules

c. additional supplemental irradiation capsules to assure the availability of high fluence and thermal annealing data for all 17 reactor vessels

d. existing test reactor irradiation data sources

e. provisions for additional test reactor irradiation data sources

The 11 utility participants in the B&WOG Reactor Vessel Integrity Program* have 17 reactor vessels with beltline regions that were fabricated using the

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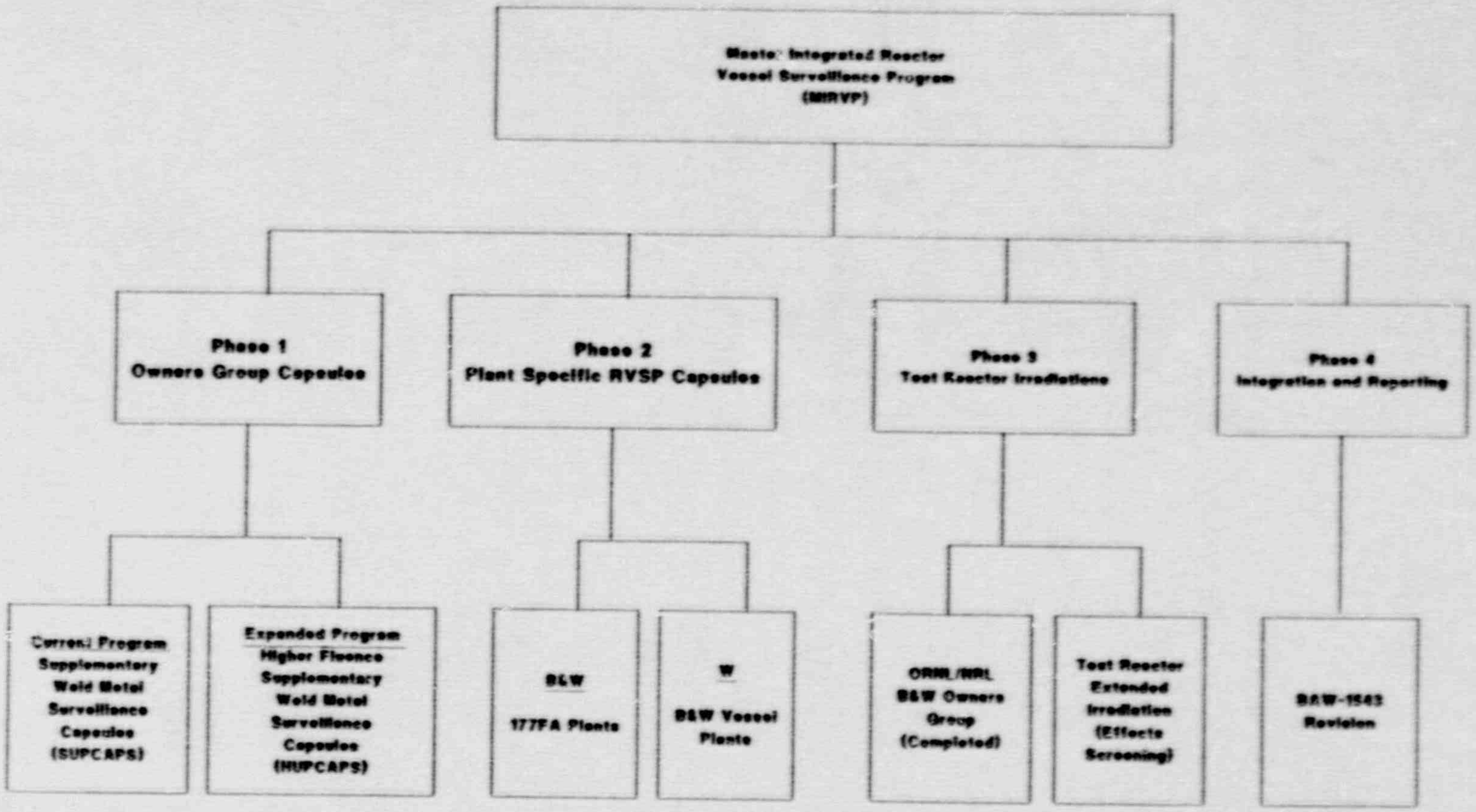
*Arkansas Power & Light Company, Commonwealth Edison Company, Duke Power Company, Florida Power Corporation, Florida Power & Light Company, General Public Utilities Nuclear Corporation, Rochester Gas & Electric Corporation, Sacramento Municipal Utility District, Toledo Edison Company, Virginia Electric & Power Company, Wisconsin Electric Power Company (Sacramento Municipal Utility District has withdrawn from the Program as a result of the shutdown of Rancho Seco Unit 1.)

Figure 1-1 Elements of Master Integrated Reactor Vessel Surveillance Program



automatic submerged arc (ASA) process with 32 Mn-Mo-Ni filler wire and Linde 80 flux lot combinations. These were fabricated from 15 heats of wire and 23 Linde 80 flux lots. The Charpy and fracture toughness properties of several wire/flux combinations (8 of the 15 heats of wire) are being characterized in existing Reactor Vessel Surveillance Programs (RVSPs), [i.e., B&WOG Integrated Reactor Vessel Surveillance Program (IRVSP) and Westinghouse plant-specific RVSPs]. The goal of the MIRVP is to develop fracture toughness data for all beltline welds to enable the performance of necessary analyses to ensure continued compliance with 10CFR50,(2) Appendix G.

The objectives of this effort are as follows:

1. Provide a unified power reactor data base for Linde 80 welds necessary to perform the analysis required by 10CFR50, Appendix G for materials that may exhibit < 50 ft-lb Charpy upper-shelf energy.

2. Maximize the effectiveness of data sharing among all participants to assure that required data is available to all participants for current and extended plant operation.

3. Provide the materials, specimens, irradiation capsules, and power reactor irradiation sites required to obtain data that can be used to evaluate the thermal annealing process.

4. Minimize testing of redundant capsules (those which do not provide useful information) in existing plant specific RVSPs to insure optimum utilization of all data sources.

5. Simplify the licensing process by providing a single document that covers the RVSP integration and capsule withdrawal schedules and which can be referenced in each utility's Technical Specifications.

This report provides a detailed description of each element of the MIRVP, explains their interrelationships, provides irradiation capsule withdrawal schedules, and provides a discussion on the technical and regulatory aspects of the integrated RVSP concept. The appendices to this report document the relevant details for each reactor vessel plant-specific surveillance program and the supplementary capsules that either are or will become part of the MIRVP.

2. BACKGROUND

It became apparent in the late 1950's that neutron embrittlement could seriously degrade the mechanical properties of steels used in reactor vessels. This was a phenomenon that varied significantly from one type of steel to another, from one heat to another, and from one weld to another. Accordingly, a number of research programs were conducted to evaluate the phenomenon. By the time the first commercial nuclear plants were designed, enough data were accumulated to confirm that the neutron irradiation damage to the reactor vessel materials could significantly degrade the properties. Not all of the first generation reactor vessels were equipped with surveillance capsules to monitor irradiation damage; however, out of this initial period of nuclear development, the guidelines for establishing an RVSP were adopted. ASTM Standard E 185-61,[3] "Standard Practice for Conducting Surveillance Tests for Light Water-Cooled Nuclear Power Reactor Vessels" was issued and conveyed the current state-of-the-art technology for designing a surveillance program.

Research on neutron irradiation damage to reactor vessel steels continued in the 1960's. Specifically, the effects of the principal parameters influencing neutron embrittlement sensitivity was studied, including the effects of differing neutron spectra, neutron flux rates, irradiation temperature, and chemical composition on pressure vessel steels. These studies were conducted using both commercial power plants and test reactors with the primary objective of determining the sensitivity of commonly used reactor vessel steels to neutron irradiation.[4-7]

In the late 1960's, a significant discovery was made when the copper and phosphorus contents in reactor vessel materials were identified by the Naval Research Laboratory as principal parameters contributing to mechanical property degradation.[8-11] Further work, in cooperation with Babcock &

Wilcox, confirmed the role of these two elements and led to their exclusion from steels and weld metal to be exposed to neutron irradiation.

The ASTM Standard E 185-61 was revised in 1966, 1970, 1973, 1979, and 1982 to reflect the knowledge gained through the above research efforts. These revisions are compared in Table 2-1.

In 1973, in a concerted effort to improve the quality of reactor pressure vessel integrity and to base the assessment of vessel integrity on a theoretical rather than an empirical basis, the concept of fracture mechanics techniques was implemented through Nuclear Regulatory Commission (NRC) regulation. These requirements are included in 10CFR50, Appendix G, "Fracture Toughness Requirements." Also included was a requirement for monitoring the neutron embrittlement of the reactor vessel beltline region, which is described in 10CFR50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements." With the issue of Appendix H, a justification was established for a concerted effort to acquire the necessary information by testing irradiated specimens from surveillance capsules and to standardize the existing surveillance program to the extent possible. Also, these Appendices made an RVSP mandatory. Up to this point, the data gathered from an RVSP had received low priority and RVSPs were nonuniform in format and content since the requirements were broadly defined and gave considerable latitude to the RVSP designer. As can be seen in Table 2-1, it was not until 1979 that ASTM E 185 required that the specific "controlling" weld be included in the RVSP capsules. ASTM E 185-73 was further revised to support the requirements of 10CFR50, Appendix H through a cooperative effort between the standard development committee and the regulators.

Today, the requirements of 10CFR50, Appendixes G and H, together with ASTM Standard E 185-82, are recognized throughout the nuclear industry as the standards and procedures for ensuring the integrity of nuclear reactor pressure vessels subject to inservice environmental degradation. These regulations require the Owners of light-water cooled nuclear power plants to monitor the neutron radiation-induced changes in impact toughness and mechanical properties of materials comprising the reactor vessel. Test data obtained from RVSPs allow determination of the conditions under which the reactor vessel may be operated to avoid nonductile failure within a prescribed

Table 2-1. Significant Differences Between Revisions of ASTM E185

ASTM E185 Revision	Materials Monitored by Program	No. of Capsules	No. of Specimens/ Capsule/Material	No. of Baseline Specimens/Mat's	Specimen Orientation	Index Temp for Measuring ΔT	Capsule Withdrawal Schedule	Dosimetry Requirements	Temperature Monitor Requirements	Special Requirements and Recommendations
1966	1. Base metal with the highest trans temp. 2. Any weld metal 3. HAZ metal	13 or more	8 Charpy 2 Tension	15 Charpy 3 Tension	Parallel to major working direc'n	Charpy energy fix temp as identified by NDT weight drop wt tests (normally 30 ft-lb)	One at neutron fluence corresponding to EOL; others not specified	Refer to ASTM E261; selection given to designer	Low melting point elements or alloys may be employed	1. Desirable to include correlation monitor 2. Thermal control specimen desirable
1970	1. Base metal with the highest trans temp. 2. Representative weld metal (same wire or rod & flux as one of the high flux region welds) 3. HAZ of base metal	3 or more	8 Charpy 2 Tension	15 Charpy 3 Tension	Parallel to major working direc'n	Same as above	One corresponding to 30% of design life; one to 100% life; others not specified	Determined per ASTM E261; Fe and unshielded Co dosimeters to be included; Ni; Cd-shielded Co & Cu suggested also	Same as above	1. Desirable to include correlation monitor spec's. 2. Thermal control specimens desirable 3. Consider inserting capsules at later time 4. Test material chemistry shall be determined
1973 Case B	Detailed selection procedure (beltline reg.) 1. Base metal 2. Weld metal (same wire or rod and type of flux as one of the welds) 3. HAZ of base metal	5	12 Charpy 2 Tension	15 Charpy 3 Tension	Normal to major working direc'n	Measured at 30 ft-lb	First 3 capsules withdrawn at specific times; 4th & 5th capsules standby	Determined per ASTM E261; Fe & unshielded Co dosimeter required	Same as above	1. Capsule neutron lead factor shall not exceed 3 2. Chemistry (including Cu,P,S,V) of test materials shall be determined 3. Consider inserting capsules at later time

Table 2-1. Significant Differences Between Revisions of ASTM E185 (Cont'd)

ASTM E185 Revision	Materials Monitored by Program	No. of Capsules	No. of Specimens/ Capsule/Material	No. of Baseline Specimens/Mat's	Specimen Orientation	Index Temp for Measuring T	Capsule Withdrawal Schedule	Dosimetry Requirements	Temperature Monitor Requirements	Special Requirements and Recommendations
1979	General guidance for selection controlling materials 1. Controlling base metal 2. Controlling weld metal (same heat of weld wire and lot of flux as beltline region controlling weld) 3. HAZ of base metal	5	12 Charpy 3 Tension & fracture mechanics	18 Charpy 3 Tension & fracture mechanics	Normal to major working direc'n	T @ 30 ft-lb RT_{NDT} T @ 50 ft-lb & T @ 35 mils for information only	5 capsules -- first 4 capsules withdrawn at specific times; 5th capsule standby 4 capsules -- first 3 capsules withdrawn at specific times; 4th capsule standby	Selected per ASTM E482 to measure integrated flux, fast neutron spectrum, & therm neutron spectrum	Same as above	1. Correlation monitor specimens are optional 2. Capsule neutron lead factor shall be between 1 and 3 3. Complete chemistry of test materials shall be determined 4. ASTM fracture toughness specimen per ASTM E636 shall be determined 5. Capsule and attachment design shall permit insertion of replacement capsule 6. Accelerated capsule optional 7. Test equipment shall be calibrated
1982	Same as above	5	Same as above	Same as above	Same as above	Same as above	Same as above	Same as above	Same as above	1. Charpy index temperatures, upper shelf energy determined from average curves

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margin of safety. Fracture mechanics techniques are used to quantitatively define plant operating conditions in terms of pressure-temperature (P-T) limits. The fracture mechanics analyses are performed in accordance with 10CFR50, Appendix G. The input information for these analyses includes material properties, applied stresses, neutron fluence, a reference flaw size, and system operating considerations.

On July 26, 1983, a revision to 10CFR50, Appendixes G and H, became effective. The most significant revisions were to (1) extend the coverage of Appendix G to include steels with specified minimum yield strengths from 50,000 to 90,000 psi, (2) determine the temperature shift at the 30 ft-lb level (this does not change the 50 ft-lb minimum upper-shelf energy criterion), (3) satisfy predicted end-of-life fracture toughness requirements using radiation conditions at the "critical location on the crack front of the assumed flaw," and (4) extend Appendix H rules to define the basic requirements of an integrated surveillance program.

The unirradiated C_vUSE level of Linde 80 welds was not high enough to accommodate regulatory requirements regarding the effects of neutron irradiation. At the time these early reactor vessels were fabricated, applicable codes and regulations did not specify minimum irradiated and unirradiated C_vUSE levels. Even though these conditions existed before the current requirements for reactor vessel fracture toughness were established, it is now required that all reactor vessel materials, regardless of the date of manufacture, must exhibit adequate toughness to prevent nonductile failure. 10CFR50, Appendix G, requires that when significant radiation-induced degradation of material fracture toughness properties occurs, corrective measures must be determined and submitted to the NRC for review three years before the material's C_vUSE is predicted to drop below 50 ft-lbs. If corrective actions are not applied in a timely manner, plant availability may be severely limited.

Imposition of these restrictions is described in 10CFR50, Appendix G, and the ASME Boiler and Pressure Vessel (B&PV) Code, Section III.[12] Paragraph V.B of 10CFR50, Appendix G, in part, states the following requirements:

> Reactor vessels may continue to be operated only for that service period within which the requirements of Section IV of this Appendix

are satisfied using the predicted value of the adjusted reference temperature and the predicted value of the upper-shelf energy at the end of the service period to account for the effects of radiation on the fracture toughness of the beltline materials.

In the event that these requirements cannot be satisfied as stated in 10CFR50, Appendix G, or by alternative procedures acceptable to the NRC, reactors may continue to operate provided all the following requirements of 10CFR50, Appendix G, paragraph V.C are satisfied:

1. A volumetric examination of 100 percent of the beltline materials that do not satisfy the requirements of Section V.B of this Appendix is made and any flaws characterized according to Section XI[13] of the ASME B&PV Code and as otherwise specified by the Director, Office of Nuclear Reactor Regulation.

2. Additional evidence of the fracture toughness of the beltline materials after exposure to neutron irradiation is to be obtained from results of supplemental fracture toughness tests.

3. A fracture analysis shall be performed that conservatively demonstrates, making appropriate allowances for all uncertainties, the existence of equivalent margins of safety for continued operations.

Paragraph V.D further states, "If the procedures of Section V.C of this Appendix do not indicate the existence of an equivalent safety margin, the reactor vessel beltline region may, subject to the approval of the Director, Office of Nuclear Reactor Regulation, be given a thermal annealing treatment to recover the fracture toughness of the material." Appendix A provides a detailed discussion on reactor vessel thermal annealing. All nuclear plants, regardless of the fabrication date, must meet the requirements stated above.

Since Appendix G also applies to the early fabrication period reactor vessels, continued operation must be justified by demonstrating that equivalent margins of safety exist for any beltline material suspected to exhibit C_vUSE less than 50 ft-lb. This requires obtaining fracture toughness data for the affected materials and performing a fracture mechanics analysis using these data.

As mentioned earlier, a revision in 1983 to 10CFR50, Appendix H defined the basic requirements of a integrated surveillance program. However, the integrated RVSP approach was accepted by the NRC and has been utilized by B&W since 1976. For an integrated RVSP to be acceptable to the NRC, a number of criteria, as provided by 10CFR50, Appendix H, must be met. Paragrpah II. C of Appendix H states the following:

A. An integrated surveillance program may be considered for a set of reactors that have similar design and operating features.

B. The representative materials chosen for surveillance from each reactor in the set may be irradiated in one or more of the reactors, but there must be an adequate dosimetry program for each reactor.

C. No reduction in the requirements for number of materials to be irradiated, specimen types, or number of specimens per reactor is permitted, but the amount of testing may be reduced if the initial results agree with predictions.

D. Integrated surveillance programs must be approved by the Director, Office of Nuclear Reactor Regulation, on a case-by-case basis. Criteria for approval include the following considerations.

E. The design and operating features of the reactors in the set must be sufficiently similar to permit accurate comparison of the predicted amount of radiation damage as a function of total power output.

F. There must be adequate arrangement for data sharing between plants.

G. There must be a contingency plan to assure that the surveillance program for each reactor will not be jeopardized by operation at reduced power level or by an extended outage of another reactor from which data are expected.

H. There must be substantial advantages to be gained, such as reduced power outages or reduced exposure to radiation, as a direct result of not requiring surveillance capsules in all reactors in the set.

The above criteria and considerations are satisfied by the MIRVP approach. A detailed discussion of these criteria and considerations is given in Section 4 of this report.

3. MASTER INTEGRATED REACTOR VESSEL SURVEILLANCE PROGRAM

3.1. General Description

The master integrated reactor vessel surveillance program combines 17 separate RVSPs and, where appropriate or necessary, provides for sharing of irradiation sites. Additionally, it addresses both the short- and long-term requirements for acquiring irradiation data and the need to improve the quality and quantity of fracture toughness data to support the continued licensability of the participating reactor pressure vessels.

The MIRVP correlates data from both power reactor surveillance monitoring and test reactor research programs. The principal sources of information are the power reactor surveillance efforts; this discussion, therefore, is mainly concerned with the power reactor program, which is comprised of three parts. The first part is the continuation of the plant-specific surveillance programs that monitor the irradiation damage to selected materials, as originally planned. The capsules contain samples of weld metal, plate or forging material, and heat-affected zone (HAZ) material from the vessel beltline and neutron dosimetry and thermal monitors; this part of the program will therefore continue to monitor the long-term effects of neutron irradiation on the reactor materials.

The second part of the program consists of a series of specially designed supplementary weld metal surveillance capsules (SUPCAPS) to study the effects of irradiation on a number of weld metals, which are anticipated to be highly sensitive to irradiation damage because of their chemical composition and low initial Charpy upper shelf energies. These capsules differ from regular plant-specific RVSP capsules in that they contain the necessary specimens to obtain fracture toughness properties of individual weld metals. The capsules are located in the same irradiation holder tubes as the regular plant-specific surveillance capsules at Crystal River-3 and Davis-Besse.

The third part of the MIRVP consists of higher fluence supplementary weld metal surveillance capsules (HUPCAPS) to obtain irradiated weld metal data (primarily fracture toughness properties) to satisfy the requirements of 10CFR50, Appendixes G and H for the current license and license renewal of the plants involved in this program. Additional objectives are to (1) provide for a capsule of Westinghouse design for correlation of irradiation data in the Westinghouse neutronic environment with the B&W 177-FA environment; (2) provide irradiation of reconstituted specimens (to accelerate data gathering); and, (3) provide definitive information on the annealing response of this family of materials.

The MIRVP also provides for the comparing of the above capsule data with data obtained on the same material by various test reactor research programs. The high flux available in a test reactor makes it possible to achieve high fluence in specimens in a relatively short time, e.g., EOL in six months. However, the neutron damage mechanism in this high flux and particular neutron energy spectrum and temperature can be different than that experienced in PWRs. Data comparisons for fluences up to ~1.4E19 have been completed to date. However, analysis of the high fluence data is not complete and additional test reactor irradiations may be necessary to fully evaluate the effects of flux density and neutron energy spectrum on the irradiation damage to these materials.

The surveillance materials in the capsules of the plant-specific RVSP were not selected in accordance with ASTM E 185-82. Hence, the materials monitored by the RVSP are not always the materials judged in 10CFR50, Appendix H, to most likely be the controlling beltline region materials with regard to radiation embrittlement for the reactor vessel for which the RVSP was designed. Consequently, the applicability of the data to be generated by the plant-specific RVSP becomes limited. However, by combining the data from all the RVSPs, it is practical to develop a data base to determine the probable values and predict the irradiation behavior of those welds for which there are no specific data. This does not preclude a plant-specific materials characterization should sufficient data be available.

3.2. Plant-Specific Surveillance Programs

The plant-specific surveillance programs include irradiation (1) in host reactors of surveillance capsules that were removed from the B&W 177-FA reactors without capsule holder tubes, and (2) of capsules from those plants in which the irradiations are being conducted. Each plant participating in the MIRVP has a plant-specific surveillance program that was designed to meet the requirements of the NRC and the ASTM E 185 revision at the time the program was developed. Table 3-1 shows typical withdrawal schedules from ASTM E 185-82. The following sections describe the B&W 177-FA and Westinghouse-designed plant RVSPs.

3.2.1. Babcock and Wilcox-Designed Reactor Vessel Surveillance Programs

There are eight B&W-designed reactor vessels that contain high-copper, Linde 80 ASA weld seams. Plant parameters are compared in Table 3-2. Irradiation of RVSP capsules for these eight reactors is currently being performed in two "host" reactors, Crystal River-3 and Davis-Besse. Originally, TMI-2 was a host reactor for TMI-1, however, the incident on March 29, 1979 at TMI-2 terminated its use. The capsules in TMI-2 were requalified[14] for continued irradiation (except for those which were destructively tested for requalification).

The following pairings of capsules and reactors are agreed upon by the Owners:

Guest Reactor/Owner	Host Reactor/Owner
Oconee Unit 1/Duke Power Company	Crystal River Unit 3/Florida Power Corporation
Oconee Unit 2/Duke Power Company	Crystal River Unit 3/Florida Power Corporation
Oconee Unit 3/Duke Power Company	Crystal River Unit 3/Florida Power Corporation
Arkansas Nuclear One Unit 1/Arkansas Power and Light Company	Davis-Besse Unit 1/Toledo Edison Company
Rancho Seco Unit 1/Sacramento Municipal Utility District	Davis-Besse Unit 1/Toledo Edison Company
Three Mile Island Unit 1/GPU Nuclear Corporation	Crystal River Unit 3/Florida Power Corporation

Table 3-1. Recommended Withdrawal Schedules in Accordance with ASTM Specification E 185-82

Sequence	Time of Withdrawal
	Six-Capsule Program
First	Earliest of 1.5 EFPY; capsule fluence >5 x 10^{18} n/cm^2; highest ΔRT_{NDT} of an encapsulated material equals 50F.
Second	Earliest of 3 EFPY; capsule fluence midway between that of the first and third capsules.
Third	Earliest of 6 EFPY; capsule fluence corresponds to that of the EOL fluence of the reactor vessel 1/4T location.
Fourth	Earliest of 15 EFPY; capsule fluence corresponds to that of the EOL fluence of the reactor vessel inside surface location.
Fifth	Standby; not less than once nor greater than twice the EOL fluence of the reactor vessel inside surface location. Capsule may be held without testing after withdrawal.
Sixth	Not required; will be treated as a standby capsule.

Table 3-2. Comparison of Plant Parameters for the B&W 177-FA RVSPs

Plant Parameters	Davis-Besse Unit 1	Rancho Seco Unit 1	Arkansas Nuclear One Unit 1	Crystal River Unit 3	Oconee Unit 1	Oconee Unit 2	Oconee Unit 3	Three Mile Island Unit 1
Design heat output (core), MWt	2772	2772	2568	2544	2568	2568	2568	2568
Design overpower, %	112	112	112	112	112	112	112	112
System pressure (nom), psia	2200	2200	2200	2200	2200	2200	2200	2200
Coolant flow rate, 10^6 lbm/h; gpm	143.8; 387,000	143.8; 387,000	139.7; 375,000	139.6; 375,000	139.7; 375,000	139.7; 375,000	139.7; 375,000	139.7; 375,000
Coolant temperatures, F								
Nominal inlet	558	558	556	556	556	556	556	556
Avg rise in vessel	49	49	47	46	47	47	47	47
Avg in vessel	582	582	579	579	579	579	579	579
No. of fuel assemblies	177	177	177	177	177	177	177	177
Type of fuel assemblies	Mark B (15x15)	Mark B (15x15)	Mark B (15x15)	Mark B (15x15)	Mark B (15x15)	Mark B (15x15)	Mark B (15x15)	Mark B (15x15)
Core barrel ID/OD, in.	141/145	141/145	141/145	141/145	141/145	141/145	141/145	141/145
Thermal shield ID/OD, in.	147/151	147/151	147/151	147/151	147/151	147/151	147/151	147/151
Core structural characteristics								
Core equiv diam, in.	128.9	128.9	128.9	128.9	128.9	128.9	128.9	128.9
Core active fuel height, in.	143.2	141.8	141.8	141.8	141.8	141.8	141.8	142.3

Table 3-2. Comparison of Plant Parameters for the B&W 177-FA RVSPs (Cont'd)

Plant Parameters	Davis-Besse Unit 1	Rancho Seco Unit 1	Arkansas Nuclear One Unit 1	Crystal River Unit 3	Oconee Unit 1	Oconee Unit 2	Oconee Unit 3	Three Mile Island Unit 1
Reactor vessel design parameters								
Principal material	SA508, Cl.2	SA533 Tp B Cl.1	SA533 Tp B Cl.1	SA533 Tp B C.1	SA302 GrB Cl.1[b]	SA508 Cl.2	SA508 Cl.2	SA302 GrB[b]
Design pressure, psig	2500	2500	2500	2500	2500	2500	2500	2500
Design temperature, F	650	650	650	650	650	650	650	650
Shell ID, in.	171	171	171	171.375	171	171	171	171
Shell thickness, in.	8.44	8.44	8.44	8.44	8.44	8.44	8.44	8.44
OD across nozzles, in.	261[c]	2.49	2.49	249	249	249	249	249
Overall vessel-closure head height[a], ft in.	40' 8 7/8"	40' 8 7/8"	40' 8 7/8"	40' 8 7/8"	40' 8 7/8"	40' 8 7/8"	40' 8 7/8"	40' 8/78"
Core barrel-thermal shield principal material	Type 304 SS	Type 304 SS	Type 304 SS	Type 304 SS	Type 304 SS	Type 304 SS	Type 304 SS	Type 304 SS

(a)Over cladding and instrumentation nozzles.

(b)As modified by Code Case 1339.

(c)For Davis-Besse Unit 1 this is the nominal OD across the inlet nozzles. The OD across the outlet nozzles is 245 inches.

The plant-specific RVSP for each of the eight B&W 177-FA nuclear plants participating in the MIRVP is described in a topical report, as follows:

Nuclear Plant*	Applicable Topical Report
Oconee Unit 1	BAW-10006A, Rev. 3[15]
Oconee Unit 2	BAW-10006A, Rev. 3
Oconee Unit 3	BAW-10006A, Rev. 3
Three Mile Island Unit 1	BAW-10006A, Rev. 3
Crystal River Unit 3	BAW-10100A[16]
Arkansas Nuclear One, Unit 1	BAW-10006A, Rev. 3
Rancho Seco Unit 1	BAW-10100A
Davis-Besse Unit 1	BAW-10100A

*The types and properties of the RVSP materials for each plant are described in Appendix A.

Each plant-specific RVSP consists of six surveillance capsules, four of which are the prime data-collecting capsules, and the others are "standby" capsules. The prime capsules are withdrawn at designated time intervals so that the data collected are for irradiation levels ranging from low fluence to that equal to the vessel inner surface (IS) at end of life (EOL). The standby capsules provide any necessary additional data late in the operating life of the plant.

Three basic types of specimens, in varying combinations, are placed in these capsules: Charpy V-notch, tension test, and compact fracture toughness (CT). (Appendix C describes the specimens in detail.) The Charpy V-notch specimens are 0.394 inch square, 2.165 inches long, and conform to ASTM E 23-72.[17] The tension test specimens are 4.25 inches long and conform to ASTM E 8-69T.[18] The compact fracture toughness specimens are 0.5 inch thick by 1.25 by 1.20 inches, and conform to the basic requirements of ASTM E 399-81[19] and E 813-81.[20] Specimen identity is maintained throughout the program by a die-stamped identification code (a combination of letters and numbers) on the top and bottom of each specimen.

In addition to the specimens, each capsule contains neutron dosimeters and thermal monitors. Figures 3-1 through 3-3 show typical capsules and the orientation of their specimens, neutron dosimeters, and thermal monitors. The voids in the capsule are filled with aluminum alloy spacers to minimize movement of the specimens inside and improve heat transfer. The capsule is helium-filled.

The B&W 177-FA integrated RVSP organizes and evaluates the data from the individual surveillance programs. Within this common network are 3 types of surveillance programs (types A, B, and C), in which 6 capsule types (I-VI) are irradiated. Surveillance program A uses capsule types I and II; program B uses types III and IV; and program C uses types V and VI.

The physical characteristics of the specimen holder tube and the capsule are described in paragraph 3.2.1.1, while the neutron dosimeters and thermal monitors are discussed in paragraphs 3.2.1.2 and 3.2.1.3. The 3 separate programs (A-C) and the capsule types (I-VI) are described in paragraph 3.2.1.4.

3.2.1.1. Structural, Hydraulic, and Thermal Characteristics of Specimen Holder Tube and Capsule

The surveillance capsule holder tubes are attached to the thermal shield and position the capsules in the downcomer annulus near the reactor vessel wall. The holder tube is located such that the midspan elevation of the tube is at the core midplane, as shown in Figure 3-4. The azimuthal locations of the holder tubes are shown in Figures 3-5 and 3-6 for the Crystal River-3 and Davis-Besse host reactors, respectively.

The thermal characteristics of the specimen holder tube and the capsule were analyzed to obtain a design in which the temperature of the specimens is approximately equal to that of the reactor vessel inside wall. This analysis was performed to determine the maximum temperature of the surveillance capsule Charpy specimens that can be expected to occur during steady state (100% power) and during an overheating transient. The perforated tube design allows enough coolant to reach the surveillance capsules to cool them to less than 9F above the temperature of the entering coolant water. This is well within the ±25F temperature criterion used for comparison to the 1/4-thickness vessel wall location.

Figure 3-1 Surveillance Capsule Arrangement -- Types I and II

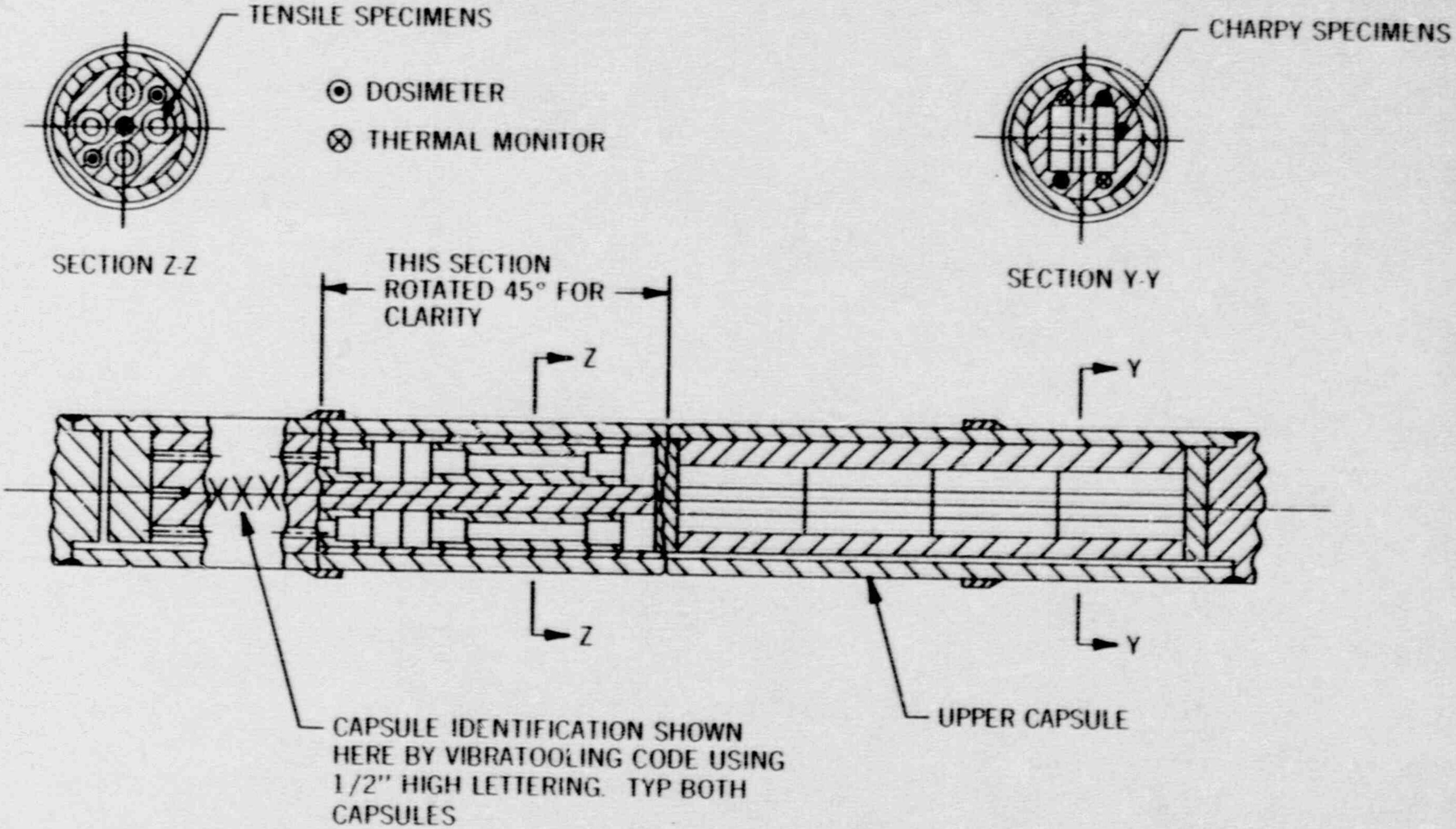


Figure 3-2 Surveillance Capsule Arrangement -- Type IV

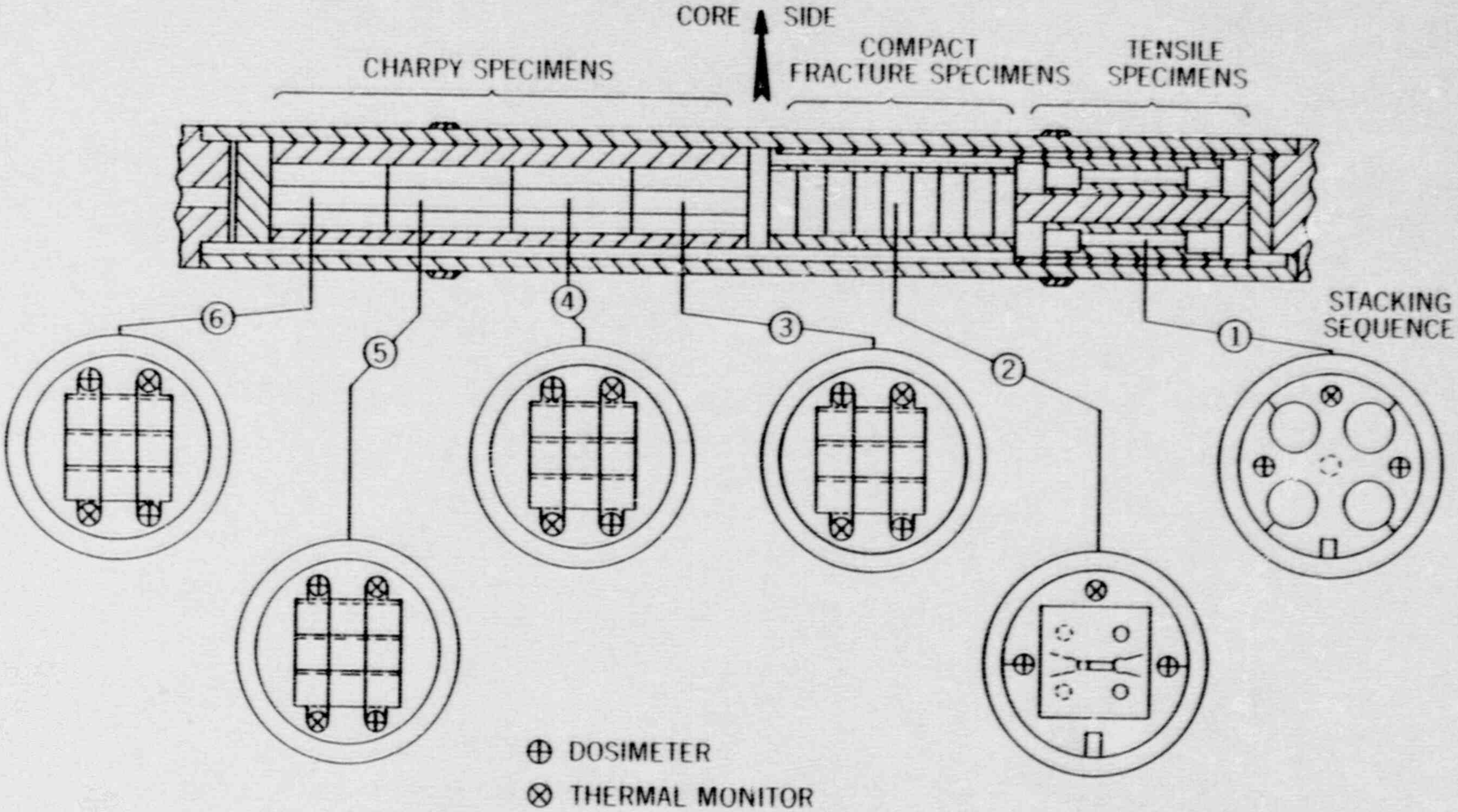


Figure 3-3 Surveillance Capsule Arrangement -- Types III, V, and VI

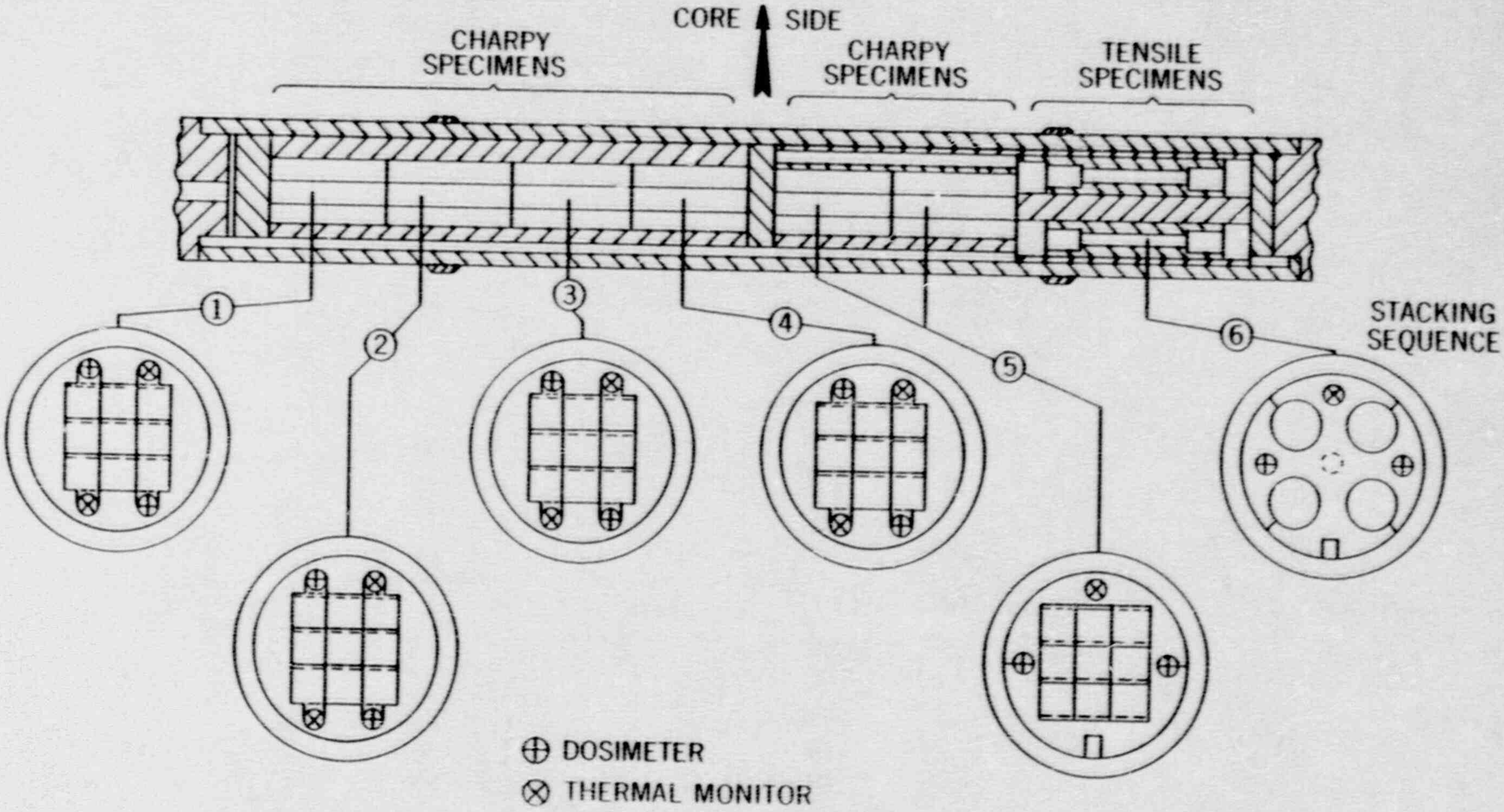


Figure 3-4 Reactor Vessel Arrangement Showing Current Surveillance Capsule Holder Tube Locations

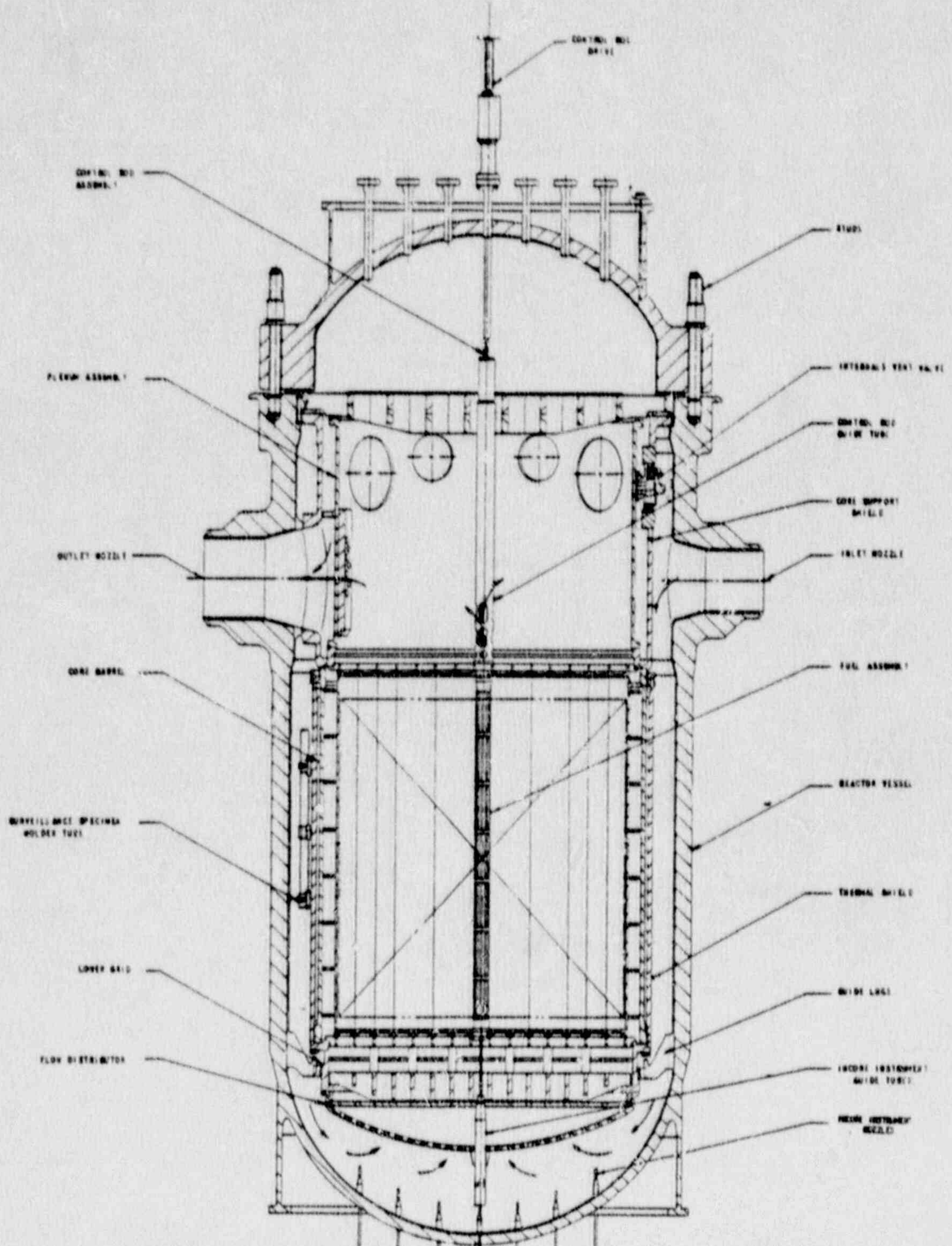


Figure 3-5 Surveillance Capsule Holder Tube Location and Identification for Crystal River Unit 3

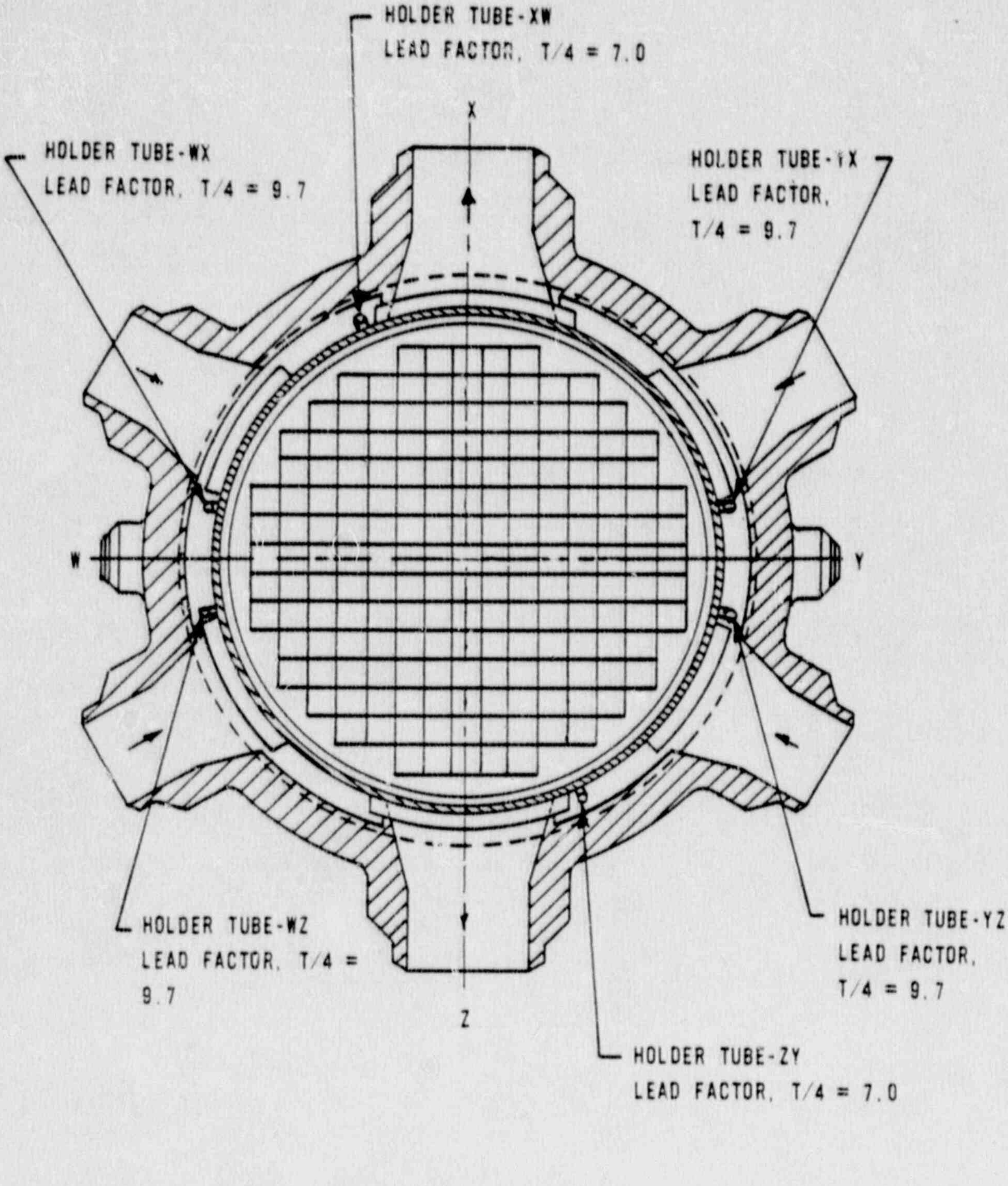
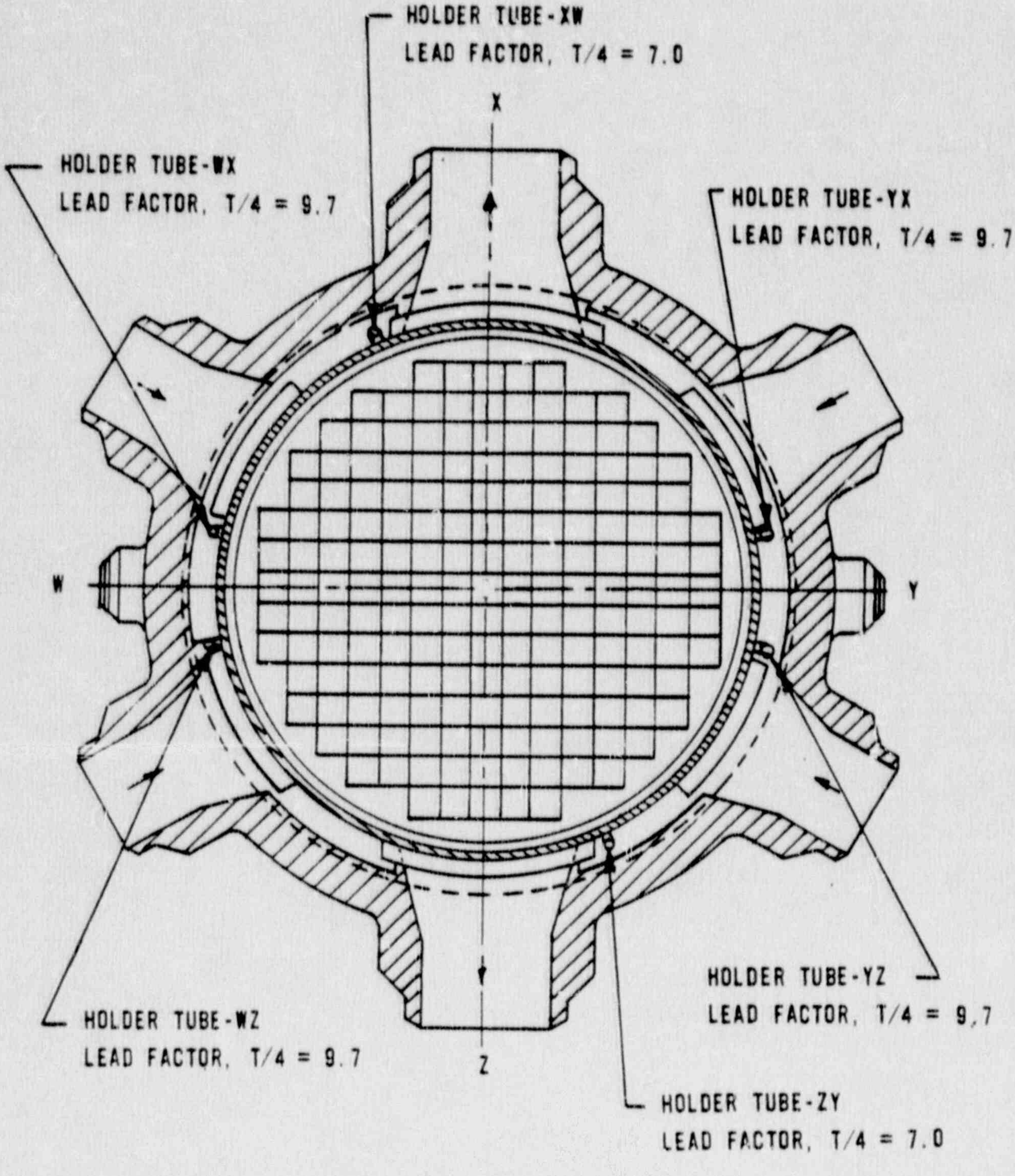


Figure 3-6 Surveillance Capsule Holder Tube Location and Identification for Davis-Besse Unit 1



The capsules are locked into the holder tube by a removable closure device that subjects the capsules to a compressive load and the holder tube to an equal tensile load. This loading is designed to minimize flow-induced vibration. (The tight inner packing also minimizes flow-induced vibrations within the capsule.) The perforated holder tube exposes the capsule to the reactor coolant. Each capsule is a stainless steel cylinder approximately 2.4 feet long, 2.5 inches in outside diameter, and 2.0 inches in inside diameter. Structurally, the capsules are designed to withstand the compressive preload and the external pressure without failure.

The capsule is designed to maintain specimens at temperature within ±25F of the reactor vessel temperature at the 1/4-thickness (1/4T) vessel wall location.* Figure 3-7 illustrates the calculated vessel wall temperature distribution for steady-state normal operation. The heat transfer analysis for the capsule considers the differences in thermal properties of the materials and the helium-filled gaps between internal components of the capsule. Conservative maximum temperatures were calculated for each different cross section within the capsule and these were within the upper bound 25F of the vessel 1/4T temperature. The coolant temperature serves as the lower bound and is also within 25F of the vessel temperature at 1/4T.[21]

The capsules are placed in the holder tubes (two per tube) which are positioned so that both the time-averaged axial distribution of the axial peak neutron flux and the initial azimuthal distribution of fast neutron flux are maximized.

3.2.1.2. Neutron Dosimetry

Neutron dosimeters are placed in the specimen capsules to determine the actual neutron fluence levels experienced by the specimens. Each capsule contains 4 dosimeter tubes, each tube accommodating 6 different dosimeter wires. The dosimeter types are listed in Table 3-3. Dosimeter tube placement within the capsules is shown in Figures 3-1 through 3-3.

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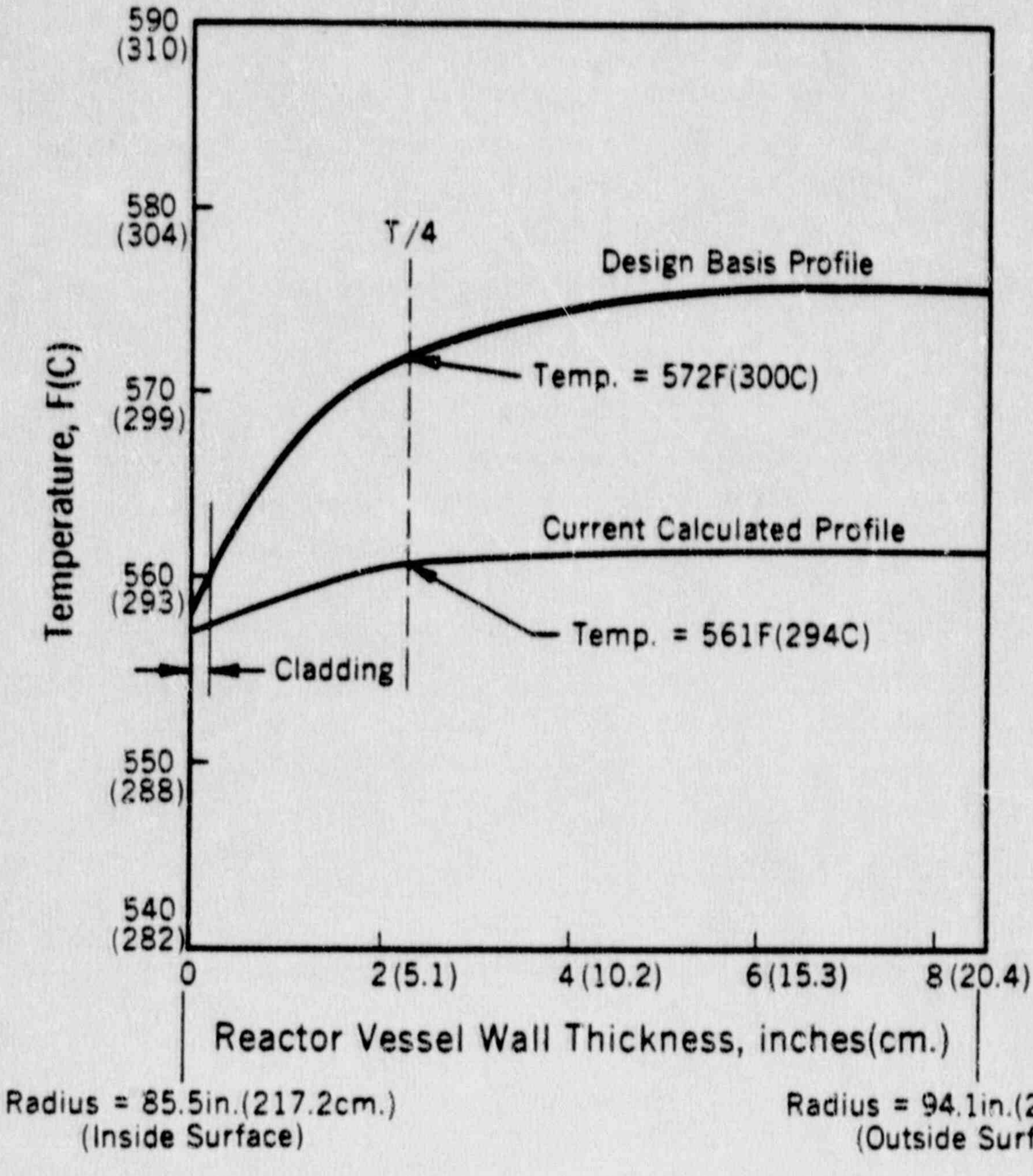
*The properties at the 1/4T vessel location contribute to the basis for periodic adjustments of the pressure-temperature relationships for normal, upset, and test conditions throughout the vessel service life.

Figure 3-7 Reactor Vessel Wall Temperature Profile During Full Power Steady State Operation



3.2.1.3. Thermal Monitors

A number of low-melting fusible alloy thermal monitors are included in each capsule to determine the maximum temperature achieved during irradiation exposure. The thermal monitors and their alloy composition and melting temperatures are given in Table 3-4. The locations of the thermal monitors within the capsule are shown in Figures 3-1 through 3-3.

3.2.1.4 Types of Surveillance Programs and Capsules

The basic surveillance programs and capsule types are briefly described below; more detailed information is presented in Appendix D. An overview of the programs and capsule types is given in Table 3-5. The materials contained in the capsules are described in Appendix A.

Surveillance Program A

Surveillance program A consists of capsule types I and II; it is described in Topical Report BAW-10006A, Rev. 3. Types I and II were originally the upper and lower capsules in the holder tubes, respectively.

Capsule Type I -- Capsule type I contains 8 tension test specimens and 36 Charpy specimens. Tension test specimens were prepared from weld metal and base metal A in the longitudinal direction.* Charpy specimens were prepared from weld metal, the HAZ of base metal A in the longitudinal direction, base metal A in both longitudinal and transverse directions, and correlation monitor plate.

Capsule Type II -- Capsule type II contains 8 tension test specimens and 36 Charpy specimens. Tension test specimens were prepared from the HAZ of heat B in the longitudinal direction and base metal heat B in the longitudinal direction. Charpy specimens were prepared from the HAZ of heat B in the longitudinal direction, base metal heat B in both the longitudinal and transverse directions, and correlation monitor plate.

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*A detailed discussion of the convention used in defining the orientation of test specimens is given in BAW-1820.[22]

Table 3-3. B&W 177-FA Plant-Specific Surveillance Capsule Dosimeters

Neutron-Sensitive Element (13-15)	Shield	Reaction Cross-Section Threshold Energy	Product Isotope
^{59}Co	Cd-Ag*	0.5 eV	5.3 yr ^{60}Co
^{59}Co	Cd-Foil*	0.5 eV	5.3 yr ^{60}Co
^{237}Np	Cd-Ag	0.5 MeV	Appropriate fission products
^{238}U	Cd-Ag	1.1 MeV	Appropriate fission products
^{58}Ni	Cd-Ag	2.3 MeV	71d ^{58}Co
^{54}Fe	None	2.5 MeV	314d ^{54}Mn
^{59}Co	None	Thermal	5.3 yr ^{60}Co

*Both shielding methods were used.

Table 3-4. B&W Capsule Thermal Monitor Wires

Approximate Melting Point, F*	Reference Materials
558	90% Pb, 5.0% Ag, 5.0% Sn
580	** 94.5% Pb, 2.5% Ag, 3.0% Sn 97.5% Pb, 2.5% Ag
588	97.5% Pb, 1.5% Ag, 1.0% Sn
610	100% Cd
621	100% Pb

*The melting point of each alloy heat or batch has been verified in its final form.

**Both alloy compositions were used.

Table 3-5. B&W 177-FA Plant-Specific Reactor Vessel Surveillance Program - Detailed Summary

Capsule ID Type	Table of Mat'l Specs	Table of Capsule Specs	Report Date	Applicable Report
Oconee Unit 1				
A I	A-1	D-1	Aug 84	BAW-1837[23]
B II	A-1	D-1	---	
C I	A-1	D-1	Oct 88	BAW-2050[24]
D II	A-1	D-1	---	
E I	A-1	D-1	Sept 77	BAW-1436[25]
F II	A-1	D-1	Sept 75	BAW-1421, Rev. 1[26]
Topical Report BAW-10006A, Rev. 3				
Oconee Unit 2				
A I	A-2	D-2	Dec 81	BAW-1699[27]
B II	A-2	D-2	---	
C I	A-2	D-2	May 77	BAW-1437[28]
D II	A-2	D-2	---	
E I	A-2	D-2	Oct 88	BAW-2051[29]
F II	A-2	D-2	---	
Topical Report BAW-10006A, Rev. 3				
Oconee Unit 3				
A V	A-3	D-3	July 77	BAW-1438[30]
B VI	A-3	D-3	Oct 81	BAW-1697[31]
C V	A-3	D-3	---	
D VI	A-3	D-3	---	
E V	A-3	D-3	---	
F VI	A-3	D-3	---	
Topical Report BAW-10100A*				
Three Mile Island Unit 1				
A I	A-4	D-4	Untested	BAW-2042[14]
B II	A-4	D-4	---	
C I	A-4	D-4	March 86	BAW-1901[32]
D II	A-4	D-4	---	
E I	A-4	D-4	June 76	BAW-1439[33]
F II	A-4	D-4	---	
Topical Report BAW-10006A, Rev. 3				

*The OC-3 capsules were fabricated before BAW-10100A was published; however, it was the OC-3 program that was described in BAW-10100A.

Table 3-5. B&W 177-FA Plant-Specific Reactor Vessel Surveillance Program - Detailed Summary (Cont'd)

Capsule ID Type	Table of Mat'l Specs	Table of Capsule Specs	Report Date	Applicable Report
Crystal River Unit 3				
A III	A-5	D-5	---	
B IV	A-5	D-5	June 81 & March 82	BAW-1679[34] and BAW-1718[35]
C III	A-5	D-5	March 86	BAW-1898[36]
D IV	A-5	D-5	March 86 & April 86	BAW-1899[37] and BAW-1914[38]
E III	A-5	D-5	---	
F IV	A-5	D-5	Sept 88	BAW-2049[39]
Topical Report BAW-10100A				
Arkansas Nuclear One Unit 1				
A I	A-6	D-6	July 84	BAW-1836[40]
B II	A-6	D-6	Nov 81	BAW-1698[41]
C I	A-6	D-6	April 89	BAW-2075[42]
D II	A-6	D-6	---	
E I	A-6	D-6	April 77	BAW-1440[43]
F II	A-6	D-6	---	
Topical Report BAW-10006A, Rev. 3				
Rancho Seco Unit 1				
A III	A-7	D-7	---	
B IV	A-7	D-7	Feb 82 & March 82	BAW-1702[44] and BAW-1720[45]
C III	A-7	D-7	---	
D IV	A-7	D-7	Oct 83	BAW-1792[46] and BAW-1793P[47]
E III	A-7	D-7	---	
F IV	A-7	D-7	April 89	BAW-2074[48]
Topical Report BAW-10100A				
Davis-Besse Unit 1				
A III	A-8	D-8	Sept 85	BAW-1882[49]
B IV	A-8	D-8	May 84 & June 85	BAW-1834[50] and BAW-1867[51]
C III	A-8	D-8	---	
D IV	A-8	D-8	---	
E III	A-8	D-8	---	
F IV	A-8	D-8	Jan 82 & March 82	BAW-1701[52] and BAW-1719[53]
Topical Report BAW-10100A				

Surveillance Program B

Surveillance program B consists of capsule types III and IV. The program is described in Topical Report BAW-10100A (referred to therein as the modified program). In addition to tension test and Charpy specimens, compact fracture toughness specimens 0.5 inch thick (0.5T CT) are included in capsule type IV. Types III and IV were originally the upper and lower capsules in the holder tubes, respectively.

Capsule Type III -- Capsule type III contains 4 tension test specimens and 54 Charpy specimens. Tension test specimens were prepared from the weld metal and base metal heat A in the transverse direction. Charpy specimens were prepared from the weld metal, HAZ heats A and B in the transverse direction, base metals heats A and B in the transverse direction, and correlation monitor plate.

Capsule Type IV -- Capsule type IV contains 4 tension test specimens, 36 Charpy specimens, and 8 compact fracture specimens 0.5 inch thick. Tension test specimens were prepared from the weld metal and base metal heat A in the transverse direction. Charpy specimens were prepared from the weld metal, the HAZ of heat A in the transverse direction, and base metal heat A in the transverse direction. The compact fracture specimens were prepared from the weld metal.

Surveillance Program C

Surveillance program C consists of capsule types V and VI. The program, described in topical report BAW-10100A, is referred to as the basic program. Capsule types V and VI were originally the upper and lower capsules in the holder tubes, respectively.

Capsule Type V -- Capsule type V contains 4 tension test specimens and 54 Charpy specimens. Tension test specimens were prepared from the weld metal and base metal heat A in the transverse direction. Charpy specimens were prepared from weld metal, the HAZ of heat A in the longitudinal direction, base metal heat A in the longitudinal and transverse directions, and heat B in the transverse direction.

Capsule Type VI -- Capsule type VI contains 4 tension test specimens and 54 Charpy specimens. The tension test specimens were prepared from the weld metal and base metal A in the transverse direction. Charpy specimens were prepared from the weld metal, the HAZ of heats A and B in the longitudinal direction, base metal of heats A and B in the transverse direction, and correlation monitor plate.

3.2.2. Westinghouse-Designed Reactor Vessel Surveillance Programs

There are nine Westinghouse-designed, B&W-fabricated reactor vessels that contain high-copper, Linde 80 ASA weld seams. Plant parameters are compared in Table 3-6. Each of these plants has an RVSP that consists of either six or eight surveillance capsules. Each capsule contains a combination of specimens that include Charpy V-notch, tension test, and WOL* specimens representative of reactor vessel material. The capsules also contain neutron dosimeters and thermal monitors. The specimens are described in further detail in Appendix C.

Each plant-specific RVSP was designed to meet the requirements of the NRC and the ASTM E 185 revision in effect at the time the program was developed. For each plant a WCAP (Westinghouse Commercial Atomic Power) report was prepared that describes the fabrication and design of the RVSP capsules. The Westinghouse-designed plant's groupings and the associated surveillance program WCAP are as follows:

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*Wedge-Opening Loading fracture toughness specimen.

Table 3-6. Comparison of Plant Parameters for the Westinghouse RVSPs

Plant Parameters	R.E. Ginna	Point Beach Unit 1	Point Beach Unit 2	Turkey Point Unit 3	Turkey Point Unit 4	Surry Unit 1	Surry Unit 2	Zion Unit 1	Zion Unit 2
Design heat output (core), MWt	1520	1518.5	1518.5	2200	2200	2241/2546 (uprated)	2241/2546 (uprated)	3250/3391 (uprated)	3250/3391 (uprated)
Design overpower, %	110	110	110	110	110	110	110	110	110
System pressure (nominal), psig	2235	2235/1985	2235/1985	2235	2235	2235	2235	2235	2235
Coolant flow rate, 10^6 lb/h; gpm	31.7;85,700	66.7;180,000	66.7;180,000	101.5;274,000	101.5;274,000	110.7;271,900	116.7;271,900	135;364,500	135;364,500
Coolant temperatures, F									
Nominal inlet	546.8	552.5	552.5	546.2	546.2	543	543	530.2	530.2
Avg rise in vessel	54	57.6	57.6	55.9	55.9	62.8	62.8	64	64
Avg in vessel	573.8	610.1	610.1	574.2	574.2	574.4	574.4	562.2	562.2
No. of fuel assemblies	121	121	121	157	157	157	157	193	193
Type of fuel assemblies	14x14	14x14	14x14	15x15	15x15	15x15	15x15	15x15	15x15
Assembly design	OFA/V-5	OFA/V-5	OFA	OFA/LOPAR	OFA/LOPAR	Standard	Standard	OFA	OFA
Core barrel ID/OD, in.	109/112.5	109/112.5	109/112.5	133.9/137.9	133.9/137.9	133.9/137.9	133.9/137.9	148/152.5	148/152.5
Thermal shield ID/OD, in.	115.3/122.5	115.3/122.5	115.3/122.5	142.6/148.0	142.6/148.0	142.6/148.0	142.6/148.0	158.5/164	158.5/164
Core structural characteristics									
Core equiv diameter, in.	96.9	96.1	96.1	119.7	119.7	119.7	119.7	132.7	132.7
Core active fuel height, in.	141.4	144	144	144	144	144	144	144	144

Table 3-6. Comparison of Plant Parameters for the Westinghouse RVSPs (Cont'd)

Plant Parameters	R.E. Ginna	Point Beach Unit 1	Point Beach Unit 2	Turkey Point Unit 3	Turkey Point Unit 4	Surry Unit 1	Surry Unit 2	Zion Unit 1	Zion Unit 2
Reactor vessel design parameters									
Principal material	A-508 Cl2	SA-302 GrB	A-508 Cl2	A-508Cl2	A-508Cl2	A-533 GrB Cl1	A-533 GrB Cl1	A-533 GrB Cl1	A-533 GrB Cl1
Design pressure, psig	2485	2485	2485	2485	2485	2485	2485	2485	2485
Design temperature, F	650	650	650	650	650	650	650	650	650
Shell ID, in.	132	132	132	155.5	155.5	157	157	173	173
Shell thickness, in.	6.50	6.50	6.50	7.75	7.75	7.75	7.75	8.44	8.44
OD across inlet/outlet nozzles, in.	230/219	230/219	230/219	174	174	174	174	262/258	262/258
Overall vessel-closure head height	39' 1.3"	37' 5"	37' 5"	42' 7"	42' 7"	40' 5"	40' 5"	43' 9.7"	43' 9.7"
Core barrel-thermal shield principal material	A240 Type 304	A240 Type 304	A240 Type 304	A240 Type 304	A240 Type 304	A240 Type 304	A240 Type 304	A240 Type 304	A240 Type 304

Nuclear Plant/Owner	Applicable Report
Group 1 (2-Loop, 14 x 14 Fuel Array)	
R. E. Ginna/Rochester Gas & Electric Corp.	WCAP-7254[54]
Point Beach Unit 1/Wisconsin Electric Power Co.	WCAP-7513[55]
Point Beach Unit 2/Wisconsin Electric Power Co.	WCAP-7712[56]
Group 2 (3-Loop, 15 x 15 Fuel Array)	
Surry Unit 1/Virginia Electric & Power Co.	WCAP-7723[57]
Surry Unit 2/Virginia Electric & Power Co.	WCAP-8085[58]
Turkey Point Unit 3/Florida Power & Light Co.	WCAP-7656[59]
Turkey Point Unit 4/Florida Power & Light Co.	WCAP-7660[60]
Group 3 (4-Loop, 15 x 15 Fuel Array)	
Zion Unit 1/Commonwealth Edison Co.	WCAP-8064[61]
Zion Unit 2/Commonwealth Edison Co.	WCAP-8132[62]

The capsules are approximately 1-inch square and are fabricated from stainless steel sheet, seal welded after being helium-filled. The capsules are autoclaved at reactor operating pressure and temperature to collapse the "can" onto the specimens to optimize thermal conductivity.

Figures 3-8 through 3-11 show the various types of capsules, the orientation of their specimens, and the location of neutron dosimeters and thermal monitors.

The capsules are attached to the thermal shield. The 1-inch square stainless steel specimen container is approximately 63 inches in height. The containers are positioned axially such that the specimens are centered on the core midplane, thereby spanning the central 5.25 feet of the 12-foot high reactor core. Additional details of capsule locations and lead factors for the various reactors are shown in Figures 3-12 through 3-16. The Westinghouse capsules are designed to meet the requirements of ASTM E185 (i.e., to duplicate the reactor vessel neutron environment as closely as practical). Therefore, they would be expected to respond to their environment in a similar manner as the B&W-design capsules. However, the thinner cladding and

Figure 3-8 Schematic Showing Specimens, Thermal Monitors and Dosimeter Placement and Orientation with Respect to the Core and Vessel Wall for Westinghouse Capsule Types I, II, III, IV, and V

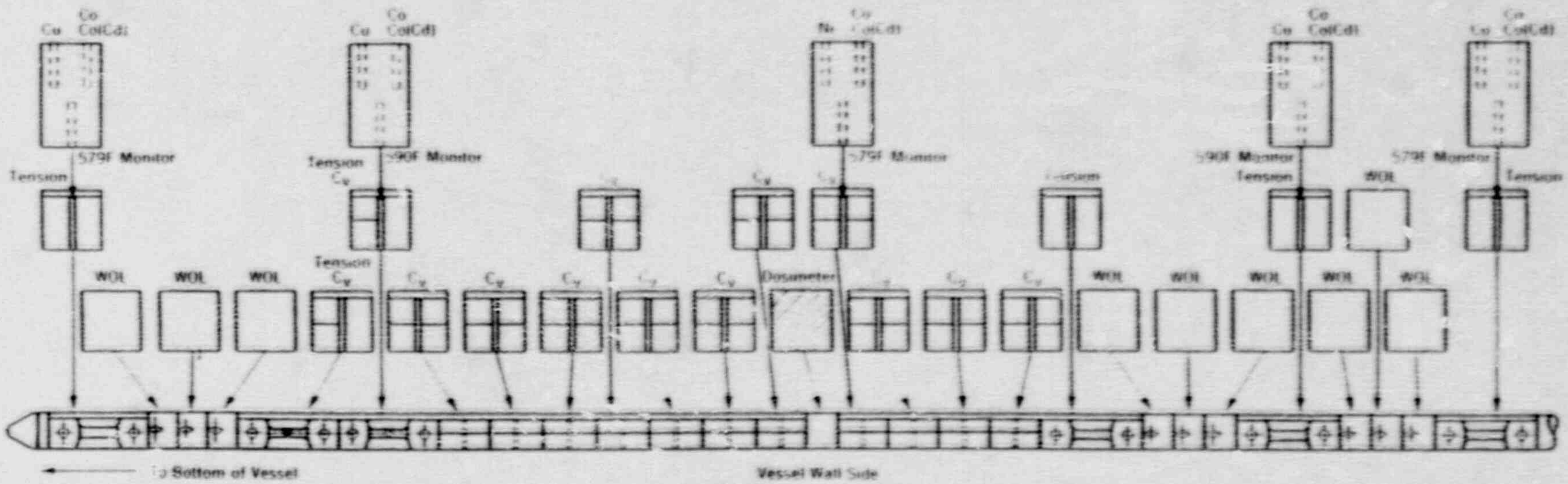


Figure 3-9 Schematic Showing Specimens, Thermal Monitors and Dosimeter Placement and Orientation with Respect to the Core and Vessel Wall for Westinghouse Capsule Type VI

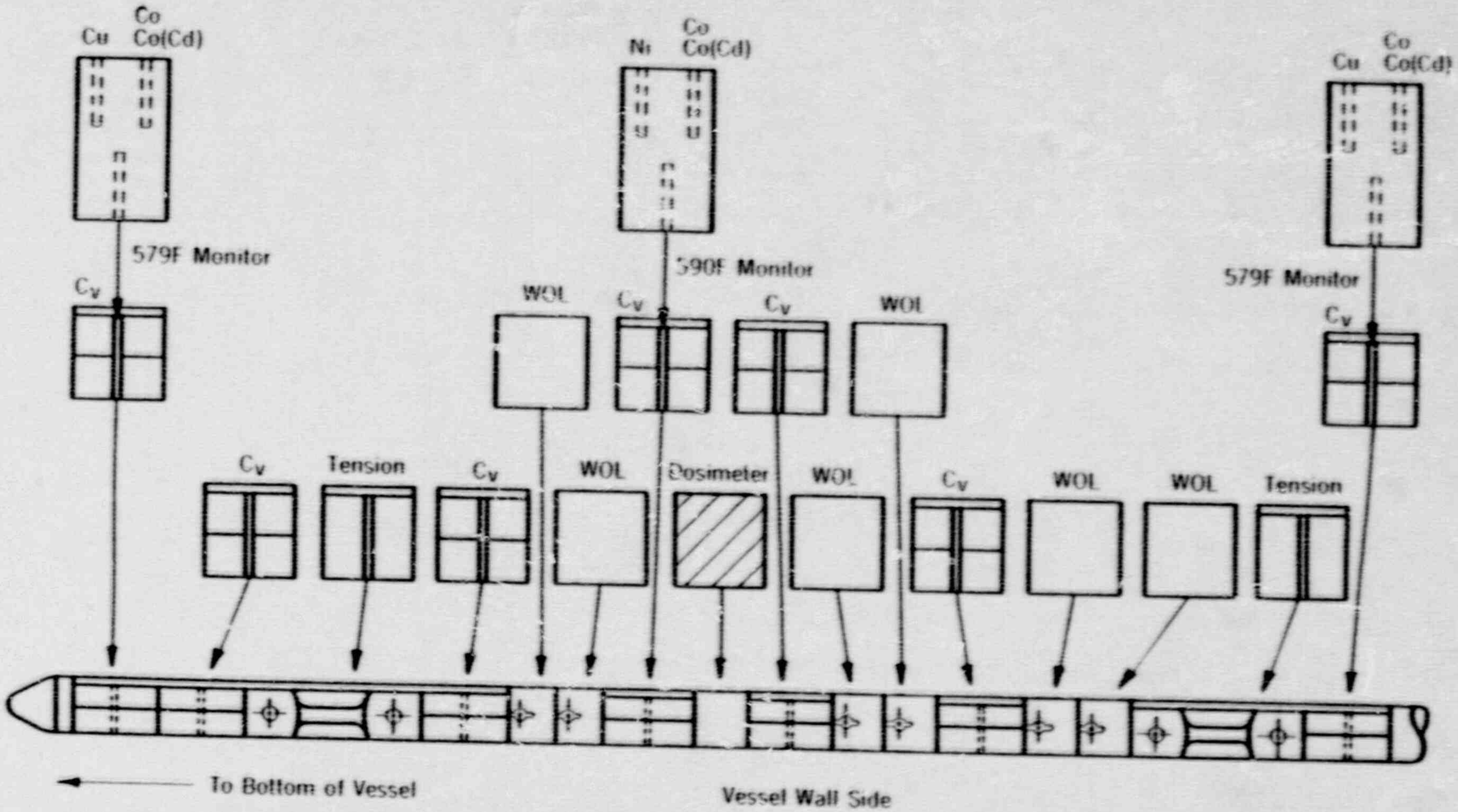


Figure 3-10 Schematic Showing Specimens, Thermal Monitors and Dosimeter Placement and Orientation with Respect to the Core and Vessel Wall for Westinghouse Capsule Type VII

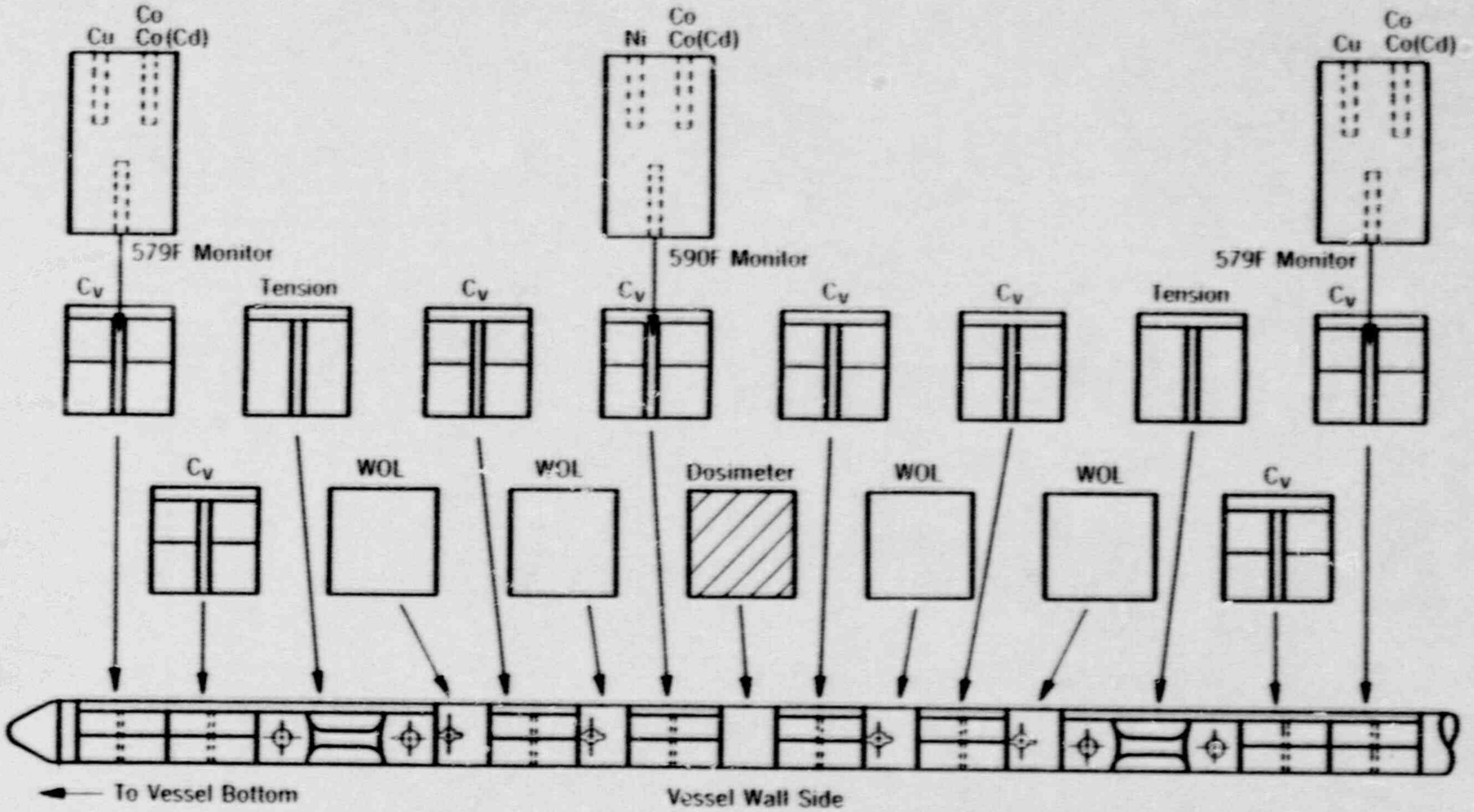


Figure 3-11 Schematic Showing Specimens, Thermal Monitors and Dosimeter Placement and Orientation with Respect to the Core and Vessel Wall for Westinghouse Capsule Types VIII and IX

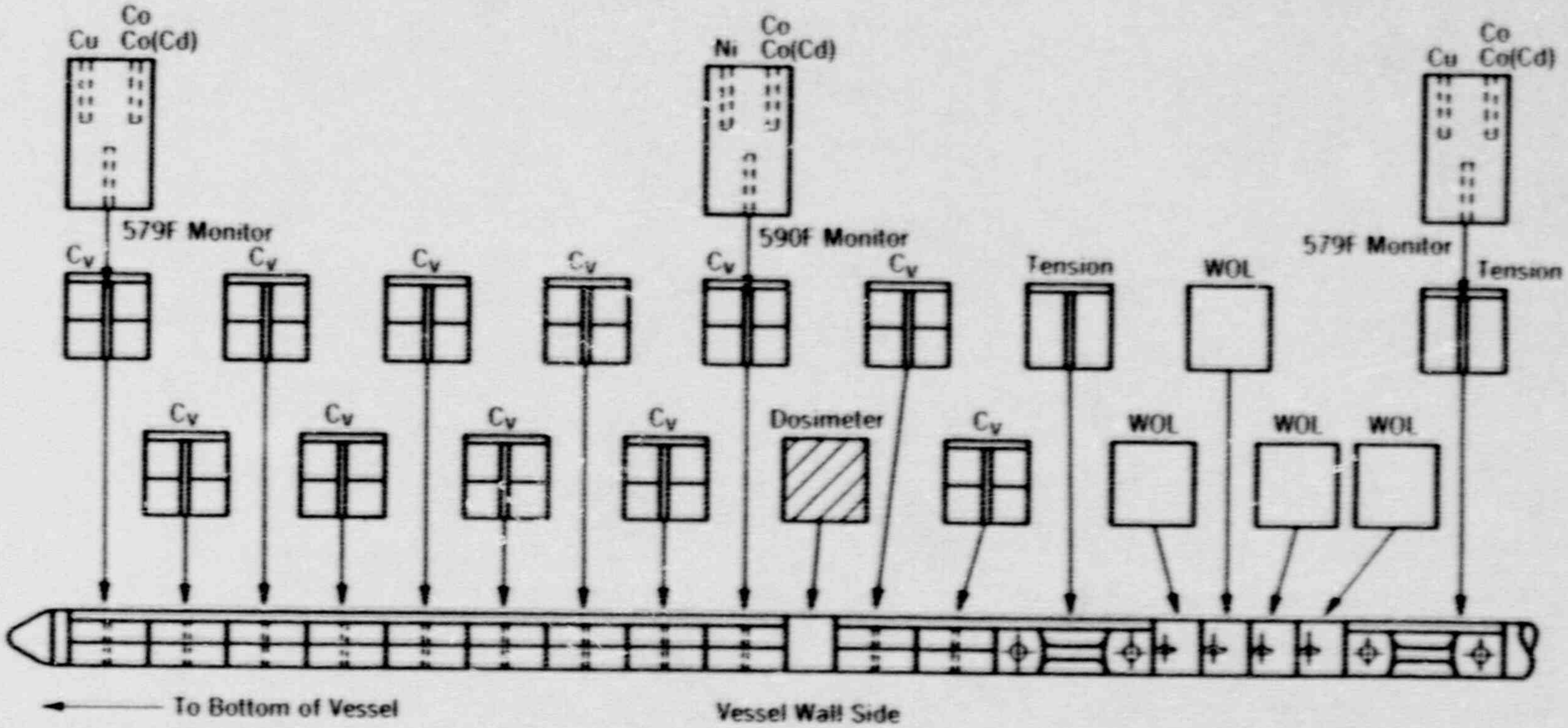


Figure 3-12 Arrangement of Surveillance Capsules in the R. E. Ginna Unit No. 1, and Point Beach Units No. 1 and 2 Reactor Vessels (Lead Factors for the Capsules Shown in Parentheses are for the Original Fuel Management)

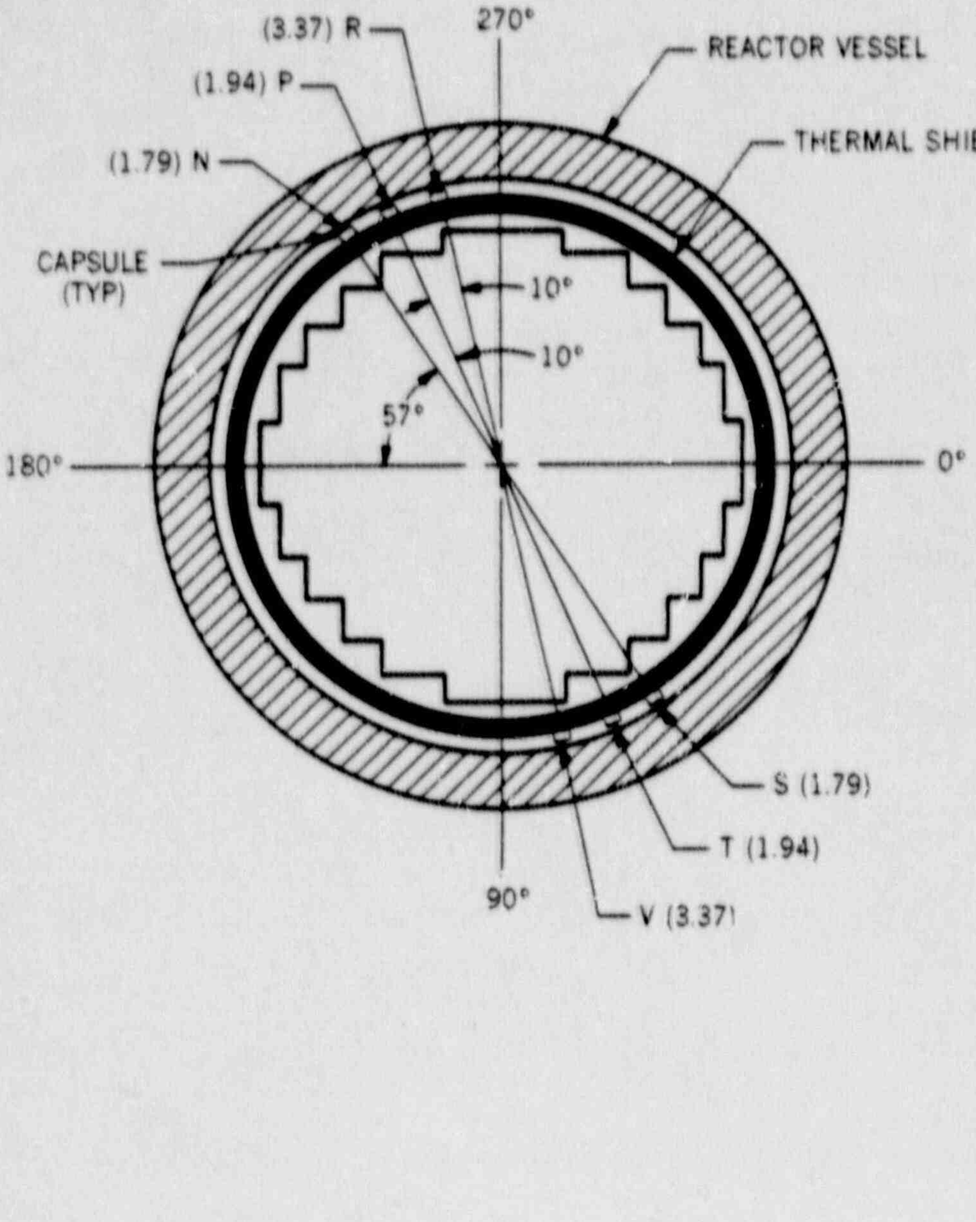


Figure 3-13 Arrangement of Surveillance Capsules in the Surry Unit 1 Vessel (Lead Factors for the Capsules Shown in Parentheses are for the Original Fuel Management)

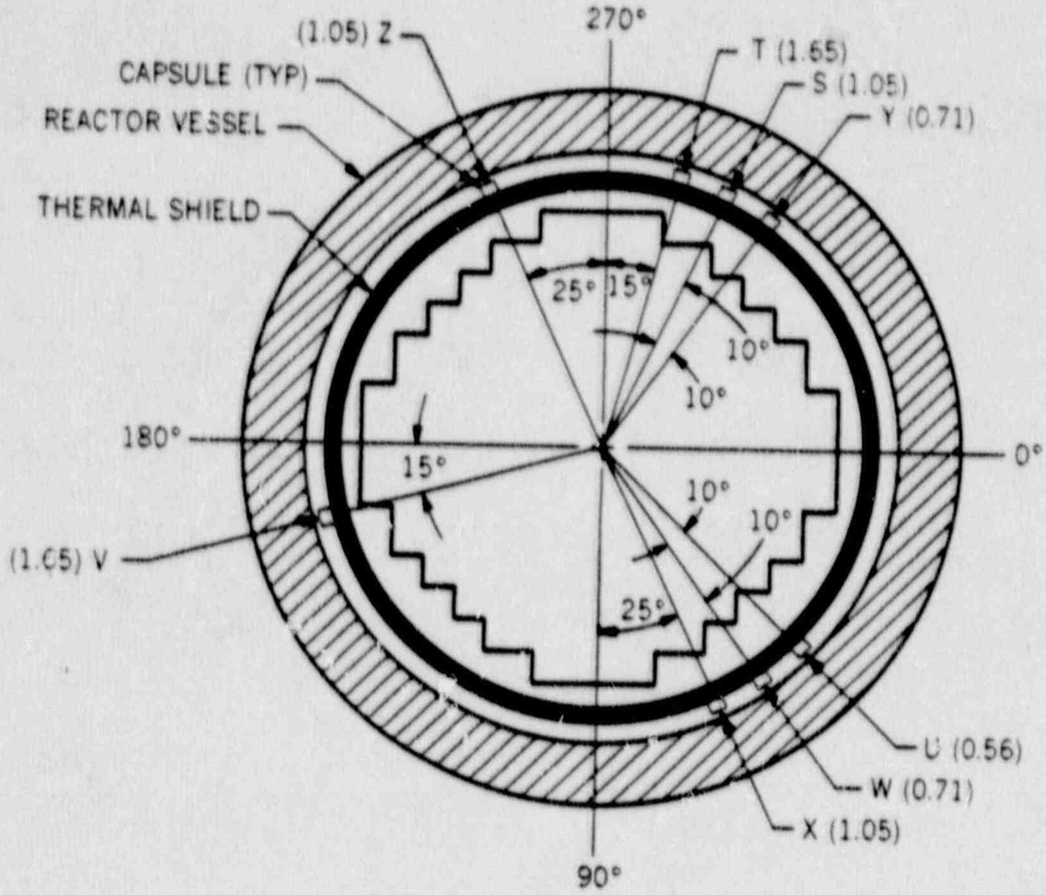


Figure 3-14 Arrangement of Surveillance Capsules in the Surry Unit 2 Reactor Vessel (Lead Factors for the Capsules Shown in Parentheses are for the Original Fuel Management)

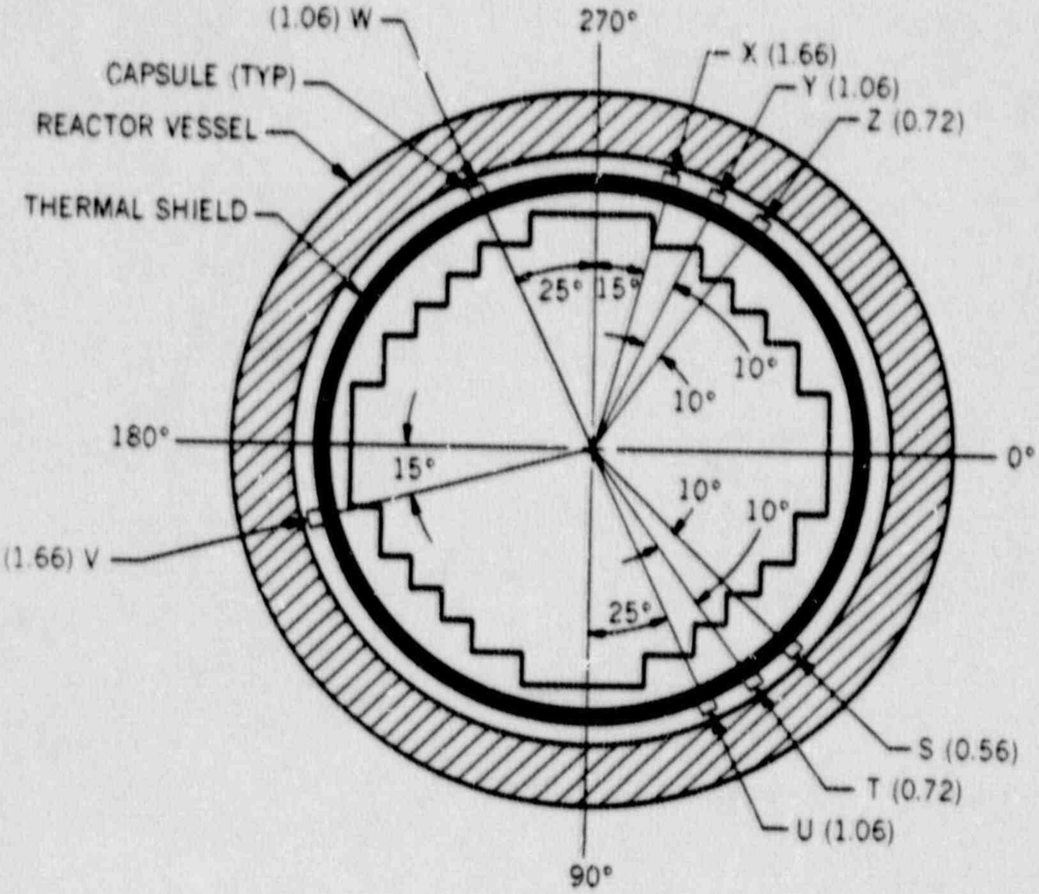


Figure 3-15 Arrangement of Surveillance Capsules in the Turkey Point Units No. 3 and 4 Reactor Vessels (Lead Factors for the Capsules Shown in Parentheses are for the Original Fuel Management)

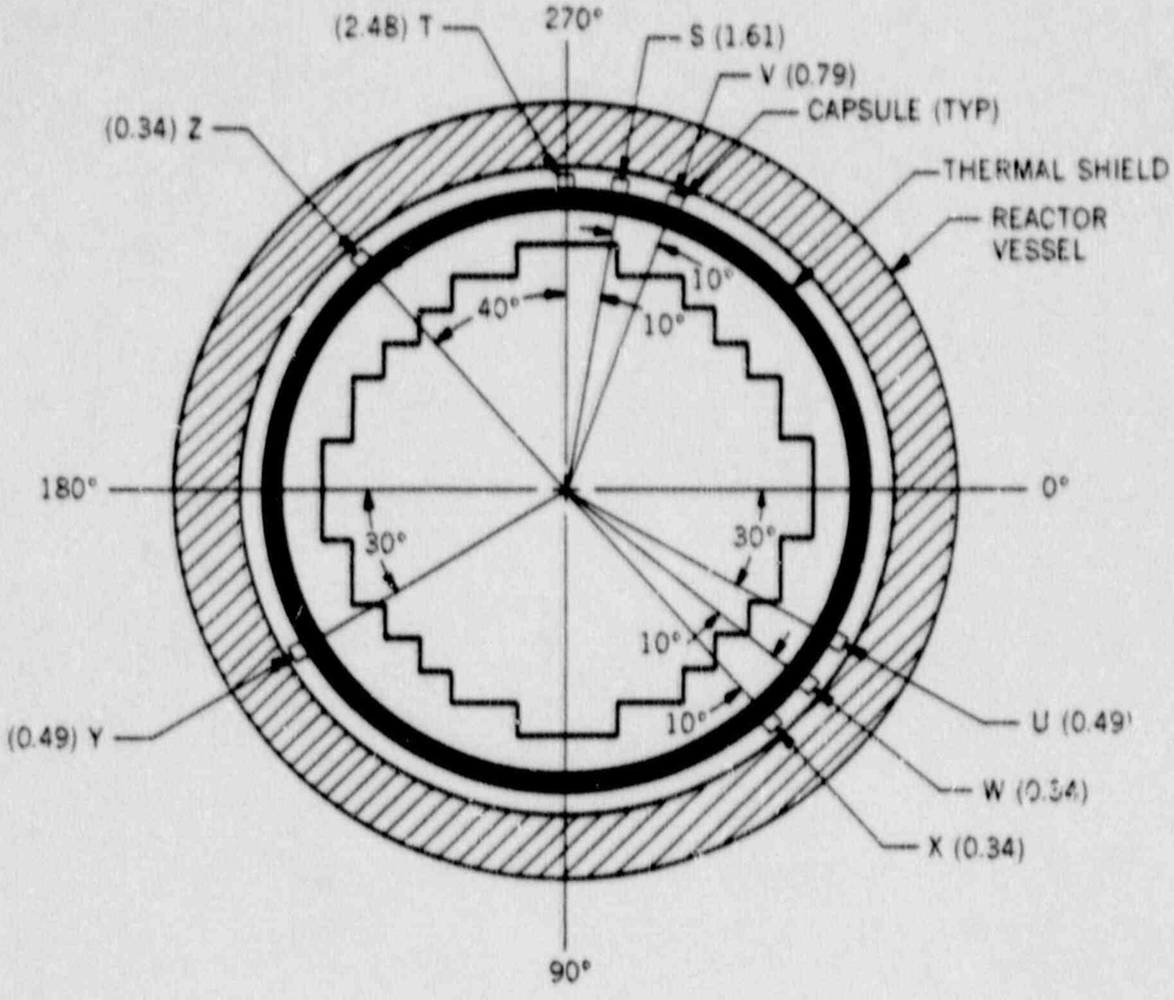
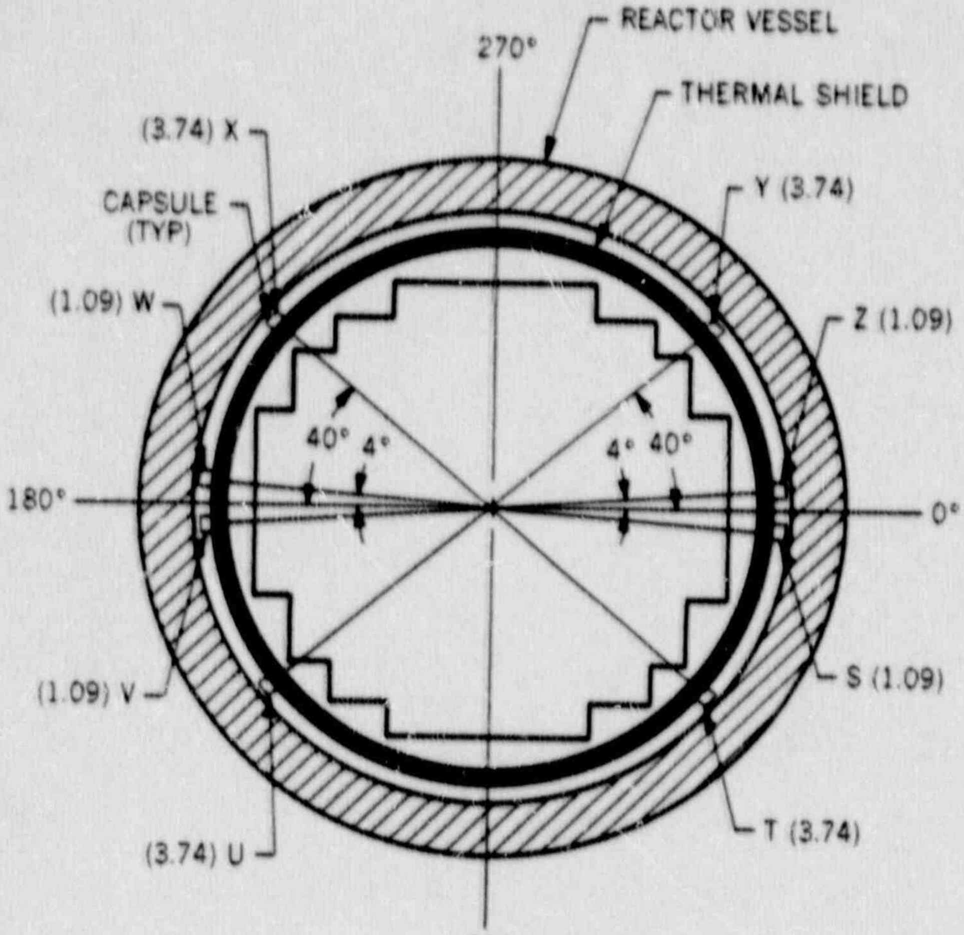


Figure 3-16 Arrangement of Surveillance Capsules in the Zion Units 1 and 2 Reactor Vessels (Lead Factors for the Capsules Shown in Parentheses are for the Original Fuel Management)



smaller cross-section of the Westinghouse-design capsules would have less sensitivity to gamma heating and greater response to the reactor vessel inlet water temperature. Because of basic differences in operating characteristics of the two designs, as shown in Figure 3-17, it is anticipated that a difference in temperature environment may exist at levels of reduced power. Relatively speaking, these periods of reduced power operation are small compared to normal operation. The differences that can exist between the two designs depending on power levels, are within the defined temperature range stated in Regulatory Guide 1.99, Revision 2, and, therefore, the data from both sets of capsule designs can be compared to the Regulatory Guide as a reference data base.

3.2.2.1. Neutron Dosimetry

Neutron dosimeters are placed in the specimen capsules to determine the actual neutron fluence levels experienced by the specimens. Four different dosimeter arrangements (location and dosimetry material selection) are utilized by the 9 plants, as shown in Table 3-7. For those capsules that do not have iron flux wires, material is removed from test specimens at a number of locations to provide iron dosimetry. Dosimeter placement within the capsules is shown in Figures 3-8 through 3-11.

3.2.2.2. Thermal Monitors

Each capsule contains a number of fusible alloy thermal monitors. The melting temperatures, alloy compositions, and arrangement of the thermal monitors for each plant are shown in Table 3-8. The locations of the thermal monitors within the capsules are shown in Figures 3-8 through 3-11.

3.2.2.3. Types of Surveillance Programs and Capsules

The nine Westinghouse-designed plants have individually arranged surveillance programs with regard to capsule type, specimen loading, and withdrawal schedule. There are nine different capsule types associated with these plants. An overview of the plant-specific programs and capsule types is given in Table 3-9. The basic programs and capsule types are briefly described below, and more detailed information is presented in Appendix D. The materials contained in the capsules are described in Appendix A.

Figure 3-17 Comparison of B&W and Typical Westinghouse Reactor Vessel Steady State Relationships Between Temperature and Power

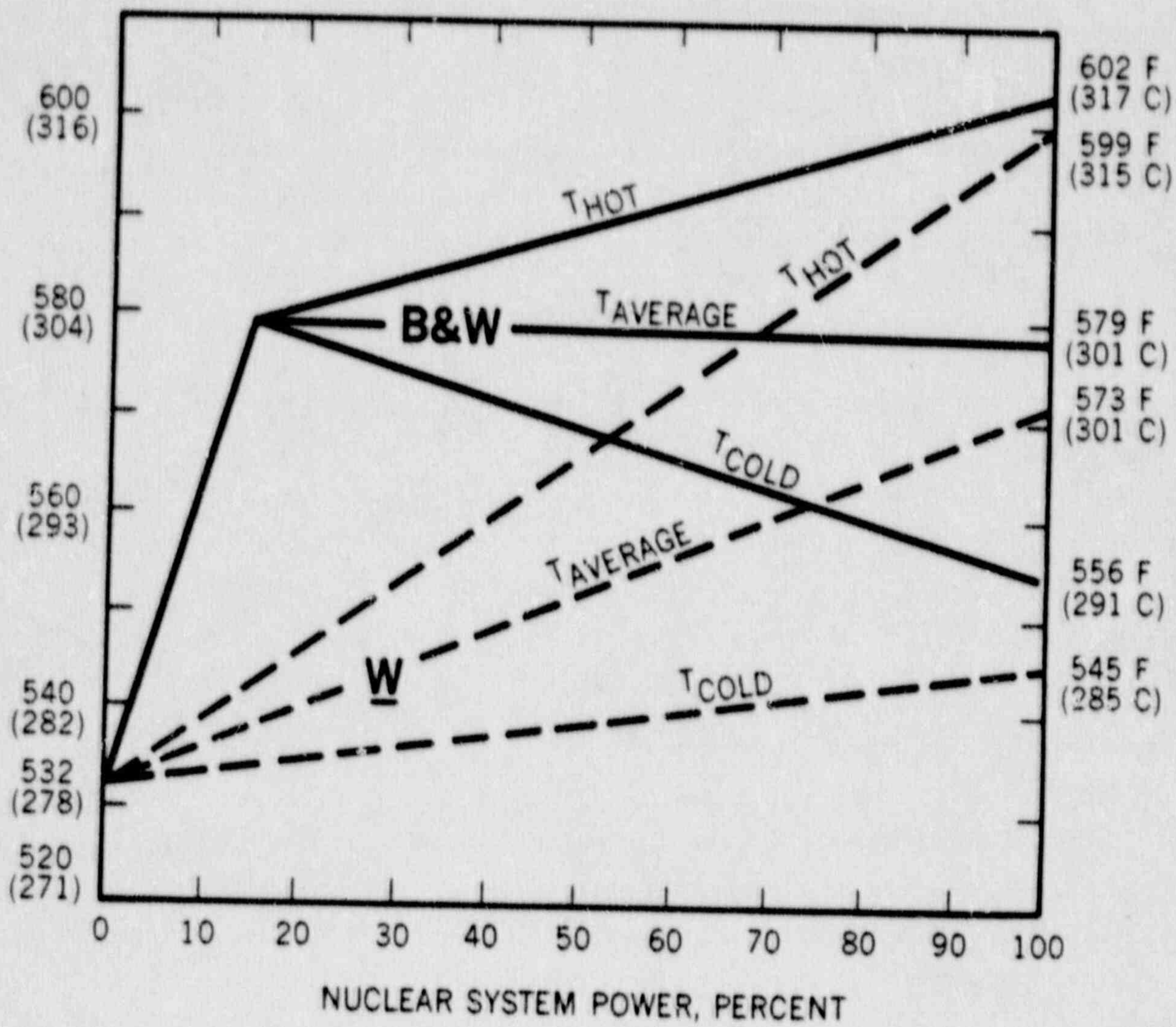


Table 3-7. Westinghouse Plant-Specific Surveillance Capsule Dosimetry

Neutron-Sensitive Element	Shield	Reaction Cross-Section Threshold Energy	Product Isotope
^{59}Co	Cd*	0.5 eV	5.3 yr ^{60}Co
^{237}Np	Cd	0.5 MeV	Appropriate fission products
^{238}U	Cd	1.1 MeV	Appropriate fission products
^{58}Ni	None	2.3 MeV	71d ^{58}Co
^{63}Cu	None	6.1 MeV	5.3 yr ^{60}Co
^{59}Co	None	Thermal	5.3 yr ^{60}Co
^{54}Fe**	None	2.5 MeV	314d ^{54}Mn

*Cadmium metal used for shielding the cobalt. Cadmium oxide used for shielding neptunium and uranium.

**Iron wires used only in Zion Unit 1, Zion Unit 2, and Surry Unit 2. Test specimens serve as iron dosimeters in the other plant RVSP capsules.

Table 3-8. Westinghouse Capsule Thermal Monitors

Plant	Top	Mid-Top	Middle	Mid-Bottom	Bottom
Ginna Point Beach-1 Point Beach-2	579F	590F	579F	590F	579F
Zion-1 Zion-2 Surry-2	590F				579F
Surry-1 Turkey Point-3 Turkey Point-4	579F		590F		579F

1. The 579F melting point alloy is 97.5 Pb-2.5 Ag.

2. The 590F melting point alloy is 97.5 Pb-1.75 Ag-0.75 Sn.

Table 3-9. Westinghouse Plant-Specific Reactor Vessel Surveillance Program - Detailed Summary

Capsule ID Type		Table of Material Specification	Table of Capsule Specification	Report Date	Applicable Report
R. E. Ginna Unit 1					
N	II	A-9	D-9	---	
P	II	A-9	D-9	---	
R	I	A-9	D-9	Nov 74	WCAP-8421(63)
S	II	A-9	D-9	---	
T	I	A-9	D-9	April 82	WCAP-10086(64)
V	I	A-9	D-9	March 73	W Report(65)
Point Beach Unit 1					
N	IV	A-10	D-10	---	
P	IV	A-10	D-10	---	
R	III	A-10	D-10	Aug 78	WCAP-9357(66)
S	IV	A-10	D-10	Nov 76	WCAP-8739(67)
T	III	A-10	D-10	Dec 84	WCAP-10736(68)
V	III	A-10	D-10	June 73	BCL Report(69)
Point Beach Unit 2					
N	IV	A-11	D-11	---	
P	IV	A-11	D-11	---	
R	V	A-11	D-11	Dec 79	WCAP-9635(70)
S	V	A-11	D-11	---	
T	IV	A-11	D-11	Aug 78	WCAP-9331(71)
V	V	A-11	D-11	June 75	BCL Report(72)
Surry Unit 1					
S	VI	A-12	D-12	---	
T	VII	A-12	D-12	June 75	BCL Report(73)
U	VI	A-12	D-12	---	
V	VII	A-12	D-12	Feb 87	WCAP-11415(74)
W	VI	A-12	D-12	March 79	BCL-585-8R(75)
X	VII	A-12	D-12	---	
Y	VI	A-12	D-12	---	
Z	VII	A-12	D-12		
Surry Unit 2					
S	VIII	A-13	D-13	---	
T	VIII	A-13	D-13	---	
U	VIII	A-13	D-13	---	
V	VIII	A-13	D-13	June 87	WCAP-11499(76)
W	VIII	A-13	D-13	Feb 81	BCL585-026(77)
X	VIII	A-13	D-13	Sept 75	BCL Report(78)
Y	IX	A-13	D-13	---	
Z	IX	A-13	D-13	---	

Table 3-9. Westinghouse Plant-Specific Reactor Vessel Surveillance Program - Detailed Summary (Cont'd)

Capsule ID Type		Table of Material Specification	Table of Capsule Specification	Report Date	Applicable Report
Turkey Point Unit 3					
S	VI	A-14	D-14	May 79	SWRI-02-5131(79)
T	VII	A-14	D-14	Sept 76	WCAP-8631(80)
U	VI	A-14	D-14	---	
V	VII	A-14	D-14	Aug 86	SWRI-06-8575(81)
W	VI	A-14	D-14	---	
X	VII	A-14	D-14	---	
Y	VI	A-14	D-14	---	
Z	VI	A-14	D-14	---	
Turkey Point Unit 4					
S	VI	A-15	D-15	May 79	SWRI-02-5380(82)
T	VII	A-15	D-15	June 76	SWRI-02-4221(83)
U	VI	A-15	D-15	---	
V	VII	A-15	D-15	---	
W	VI	A-15	D-15	---	
X	VII	A-15	D-15	---	
Y	VI	A-15	D-15	---	
Z	VI	A-15	D-15	---	
Zion Unit 1					
S	VIII	A-16	D-16	---	
T	VIII	A-16	D-16	March 78	BCL-585-4(84)
U	VIII	A-16	D-16	March 81	WCAP-9890(85)
V	VIII	A-16	D-16	---	
W	VIII	A-16	D-16	---	
X	VIII	A-16	D-16	March 84	SWRI-06-7484-001(86)
Y	IX	A-16	D-16	---	
Z	IX	A-16	D-16	---	
Zion Unit 2					
S	VIII	A-17	D-17	---	
T	VIII	A-17	D-17	July 83	SWRI Report(87)
U	VIII	A-17	D-17	March 78	BCL-585-4(88)
V	VIII	A-17	D-17	---	
W	VIII	A-17	D-17	---	
X	VIII	A-17	D-17	---	
Y	IX	A-17	D-17	---	
Z	IX	A-17	D-17	---	

R.E. Ginna Unit 1

Two types of capsules, here designated as types I and II, are utilized in the surveillance program of R.E. Ginna Unit 1. Capsule type I contains 9 tension test specimens, 48 Charpy specimens, and 9 WOL specimens. Tension test specimens were prepared from weld metal and base metal from each of the intermediate and lower shell course forgings in the longitudinal (or hoop) direction. Charpy V-notch specimens were prepared from weld metal, the HAZ of the intermediate shell course in the longitudinal direction, base metal from each of the shell courses in the longitudinal direction, base metal from each of the shell courses in the longitudinal direction, and correlation monitor plate. WOL fracture toughness specimens were prepared from weld metal and base metal from each of the shell courses in the longitudinal direction.

Capsule type II contains 9 tension test specimens, 48 Charpy specimens, and 9 WOL specimens. Tension test specimens were prepared from weld metal and base metal from each of the intermediate and lower shell course forgings in the longitudinal (or hoop) direction. Charpy V-notch specimens were prepared from weld metal, the HAZ of the intermediate shell course in the longitudinal direction, and base metal from each of the shell courses in the longitudinal direction. WOL fracture toughness specimens were prepared from weld metal and base metal from each of the shell courses in the longitudinal direction.

Point Beach Unit 1

Two types of capsules, here designated as types III and IV, are utilized in the surveillance program of Point Beach Unit 1. Capsule type III contains 9 tension test specimens, 48 Charpy specimens, and 9 WOL specimens. Tension test specimens were prepared from weld metal and base metal from each of the intermediate and lower shell course plates in the longitudinal direction. Charpy V-notch specimens were prepared from weld metal, the HAZ of the intermediate shell course in the longitudinal direction, base metal from each of the shell courses in the longitudinal direction, and correlation monitor plate. WOL fracture toughness specimens were prepared from weld metal and base metal from each of the shell courses in the longitudinal direction.

Capsule type IV contains 9 tension test specimens, 48 Charpy specimens, and 9 WOL specimens. Tension test specimens were prepared from base metal from each of the intermediate and lower shell course plates in the longitudinal direction. Charpy V-notch specimens were prepared from weld metal, the HAZ of the intermediate shell course in the longitudinal direction, base metal from each of the shell courses in the longitudinal direction, and correlation monitor plate. WOL fracture toughness specimens were prepared from base metal from each of the shell courses in the longitudinal direction.

Point Beach Unit 2

Two types of capsules, here designated as types IV and V, are utilized in the surveillance program of Point Beach Unit 2. Capsule type IV is the same as in Point Beach Unit 1. Capsule type V contains 9 tension test specimens, 48 Charpy specimens, and 9 WOL specimens. Tension test specimens were prepared from weld metal and base metal from each of the intermediate and lower shell course forgings in the longitudinal (or hoop) direction. Charpy V-notch specimens were prepared from weld metal, the HAZ of the intermediate shell course in the longitudinal direction, base metal from each of the shell courses in the longitudinal direction, and correlation monitor plate. WOL fracture toughness specimens were prepared from weld metal and base metal from each of the shell courses in the longitudinal direction.

Surry Unit 1

Two types of capsules, here designated as types VI and VII, are utilized in the surveillance program of Surry Unit 1. Capsule type VI contains 4 tension test specimens, 28 Charpy specimens, and 6 WOL specimens. Tension test specimens were prepared from base metal from each of the intermediate and lower shell course plates in the longitudinal direction. Charpy V-notch specimens were prepared from base metal from each of the shell courses in the longitudinal direction and correlation monitor plate. WOL fracture toughness specimens were prepared from base metal from each of the shell courses in the longitudinal direction.

Capsule type VII contains 4 tension test specimens, 32 Charpy specimens, and 4 WOL specimens. Tension test specimens were prepared from weld metal and base metal from the intermediate shell course plate in the longitudinal direction. Charpy V-notch specimens were prepared from weld metal, the HAZ of the intermediate shell course in the longitudinal direction, base metal from the intermediate shell course in the longitudinal direction, and correlation monitor plate. WOL fracture toughness specimens were prepared from weld metal and base metal from the intermediate shell course in the longitudinal direction.

Surry Unit 2

Two types of capsules, here designated as types VIII and IX, are utilized in the surveillance program of Surry Unit 2. Capsule type VIII contains 4 tension test specimens, 44 Charpy specimens, and 4 WOL specimens. Tension test specimens were prepared from weld metal and base metal from the intermediate shell course plate in the longitudinal (or transverse) direction. Charpy V-notch specimens are prepared from weld metal, the HAZ of the intermediate shell course in the longitudinal direction, base metal from the intermediate shell course in the longitudinal and transverse directions, and correlation monitor plate. WOL fracture toughness specimens were prepared from base metal from the intermediate shell course in the longitudinal (or transverse) direction.

Capsule type IX contains 4 tension test specimens, 44 Charpy specimens, and 4 WOL specimens. Tension test specimens were prepared from weld metal and base metal from the intermediate shell course plate in the longitudinal direction. Charpy V-notch specimens were prepared from weld metal, the HAZ of the intermediate shell course in the longitudinal direction, base metal from the intermediate shell course in the longitudinal and transverse directions, and correlation monitor plate. WOL fracture toughness specimens were prepared from weld metal.

Turkey Point Unit 3

Two types of capsules, here designated as types VI and VII, are utilized in the surveillance program of Turkey Point Unit 3. Capsule type VI contains 4 tension test specimens, 28 Charpy specimens, and 6 WOL specimens. Tension test specimens were prepared from base metal from each of the intermediate and lower shell course forgings in the longitudinal (or hoop) direction. Charpy V-notch specimens were prepared from the base metal from each of the shell courses in the longitudinal direction and correlation monitor plate. WOL fracture toughness specimens were prepared from base metal from each of the shell courses in the longitudinal direction.

Capsule type VII contains 4 tension test specimens, 32 Charpy specimens, and 4 WOL specimens. Tension test specimens were prepared from weld metal and base metal from the intermediate (or lower) shell course forging in the longitudinal direction. Charpy V-notch specimens were prepared from weld metal, the HAZ of the intermediate shell course in the longitudinal direction, base metal from the intermediate (or lower) shell course in the longitudinal direction, and correlation monitor plate. WOL fracture toughness specimens were prepared from weld metal and base metal from the intermediate (or lower) shell course in the longitudinal direction.

Turkey Point Unit 4

Two types of capsules, here designated as types VI and VII, are utilized in the surveillance program of Turkey Point Unit 4. Capsule loading for type VI is the same as in Turkey Point Unit 3.

Capsule type VII contains 4 tension test specimens, 32 Charpy specimens, and 4 WOL specimens. Tension test specimens were prepared from weld metal and base metal from the lower shell course forging in the longitudinal direction. Charpy V-notch specimens were prepared from weld metal, the HAZ of the intermediate shell course in the longitudinal direction, base metal from the lower shell course in the longitudinal direction, and correlation monitor plate. WOL fracture toughness specimens were prepared from weld metal and base metal from the lower shell course in the longitudinal direction.

Zion Unit 1

Two types of capsules, here designated as types VIII and IX, are utilized in the surveillance program of Zion Unit 1. Capsule type VIII contains 4 tension test specimens, 44 Charpy specimens, and 4 WOL specimens. Tension test specimens were prepared from weld metal and base metal from the intermediate shell course plate in the longitudinal (or transverse) direction. Charpy V-notch specimens were prepared from weld metal, the HAZ of the intermediate shell course in the longitudinal direction, base metal from the intermediate shell course in the longitudinal and transverse directions, and correlation monitor plate. WOL fracture toughness specimens were prepared from base metal from the intermediate shell course in the longitudinal (or transverse) direction.

Capsule type IX contains 4 tension test specimens, 44 Charpy specimens, and 4 WOL specimens. Tension test specimens were prepared from weld metal and base metal from the intermediate shell course plate in the longitudinal direction. Charpy V-notch specimens were prepared from weld metal, the HAZ of the intermediate shell course in the longitudinal direction, base metal from the intermediate shell course in the longitudinal and transverse directions, and correlation monitor plate. WOL fracture toughness specimens were prepared from weld metal.

Zion Unit 2

Two types of capsules, here designated as types VIII and IX, are utilized in the surveillance program of Zion Unit 2. Capsule loading is the same as Zion Unit 1.

3.3. Supplementary Weld Metal Surveillance Capsules

3.3.1. Introduction

The Supplementary Weld Metal Surveillance Capsules (SUPCAPS) are included in the MIRVP for the irradiation and testing of 8 weld metals [SA-1135, SA-1526, SA-1585, WF-25(6), WF-25(9), WF-67, WF-70(N), WF-112] contained in 6 capsules. The capsules are being irradiated in the two B&W 177-FA host reactors. The 6 SUPCAPS are labeled TMI2-LG1, TMI2-LG2, CR3-LG1, CR3-LG2, DB1-LG1, and DB1-LG2. Each SUPCAP contains Charpy V-notch, tension test, and compact fracture specimens from 3 weld metals. There are two capsule designs, types R-1 and R-2, as shown in Figures 3-18 and 3-19. The type R-2 capsule represents an improved design over type R-1 since it utilizes subsize (Charpy size) tension test specimens. The subsize specimens allow the addition of 5 more tension test and 3 more compact fracture specimens per capsule. In addition, there are small variations between types R-1 and R-2 in the location of the thermal monitors and neutron dosimeters.

The TMI2-LG1 and TMI2-LG2 capsules are type R-1, and the CR3-LG1, CR3-LG2, DB1-LG1, and DB1-LG2 capsules are type R-2. Table 3-10 identifies the weld metals irradiated in each capsule as well as the distribution of specimens. The specimens listed as 0.394TCT, 0.500TCT, and 0.936TRCT are the compact

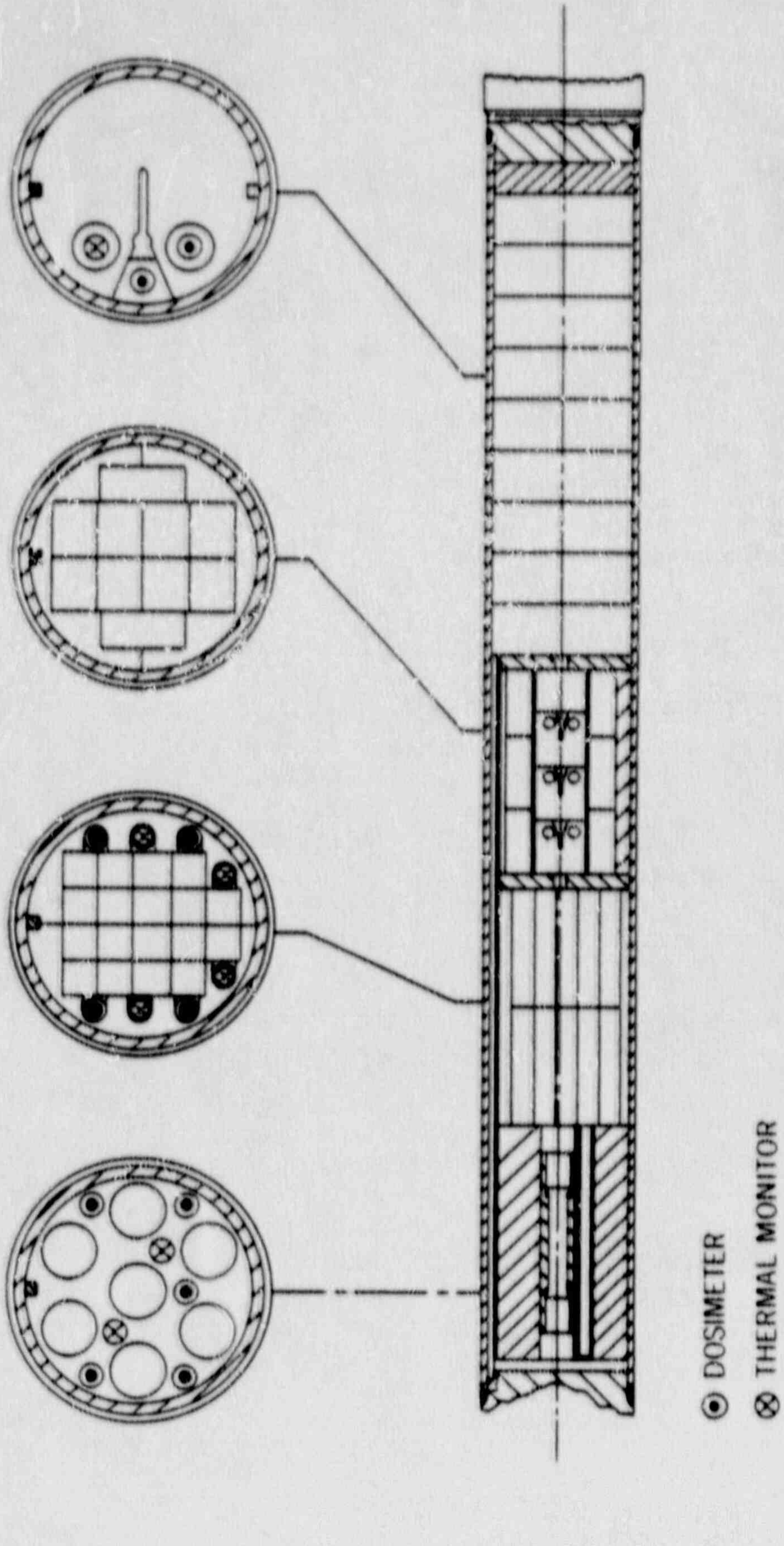


Figure 3-18 SUPCAP Capsule, Type R-1

Figure 3-19 SUPCAP Capsule, Type R-2

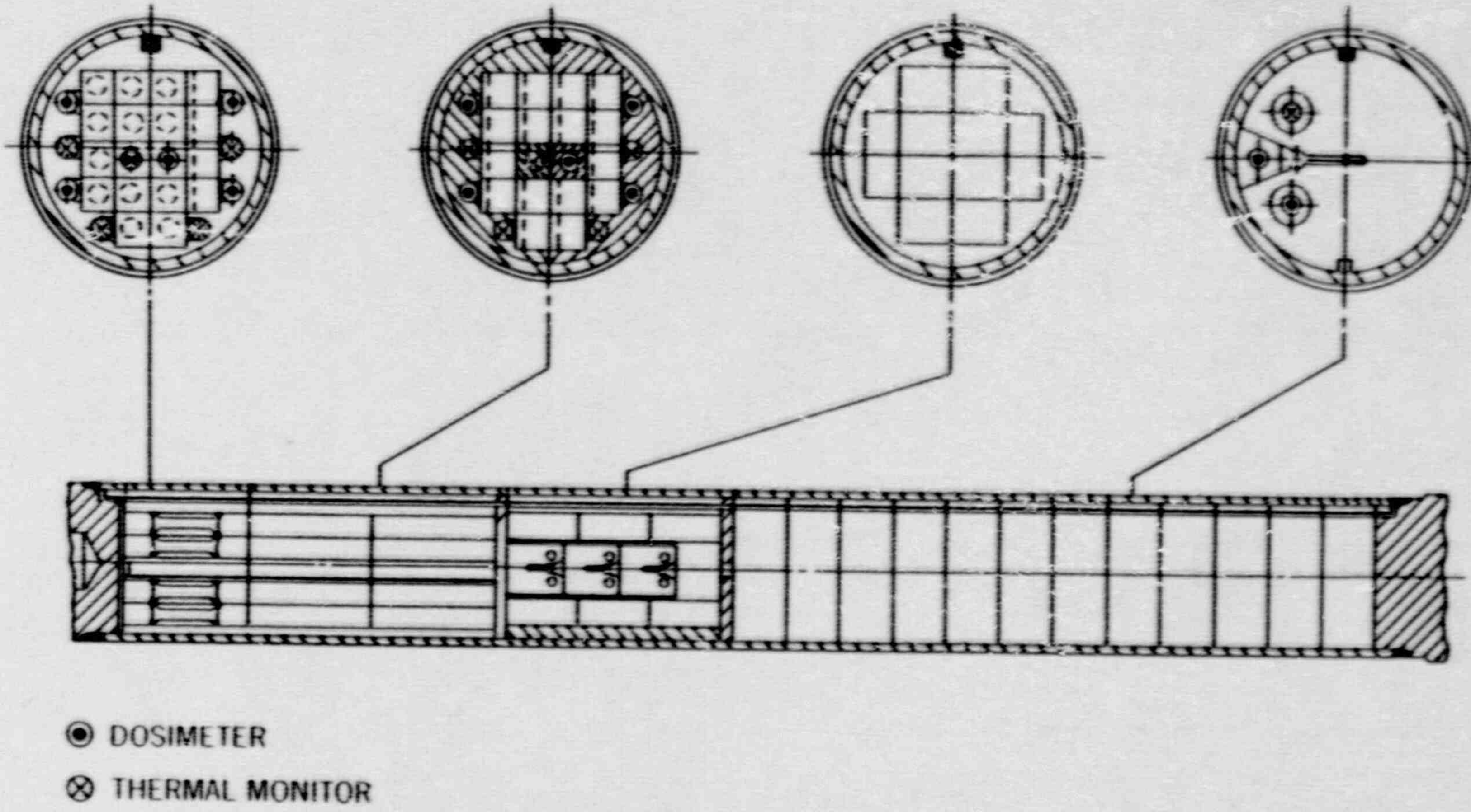


Table 3-10. SUPCAPS -- Material and Specimens Per Capsule

Weld Metal	Tension	Charpy	0.394 TCT	0.500 TCT	0.936 TRCT
	Specimens				
Capsule TMI2-LG1					
WF-70(N)*	3	12	2	4	3
SA-1526	2	12	2	4	3
WF-25(6)**	2	12	2	4	3
Capsule TMI2-LG2					
SA-1526	2	12	2	4	3
WF-25(6)	2	12	2	4	3
WF-25(9)***	3	12	2	4	3
Capsule CR3-LG1					
SA-1535	4	12	2	4	4
WF-67	4	12	2	4	4
WF-25(9)	4	12	2	4	4
Capsule CR3-LG2					
WF-70(N)	4	12	2	4	4
SA-1585	4	12	2	4	4
WF-67	4	12	2	4	4
Capsule DB1-LG1					
WF-70(N)	4	12	2	4	4
WF-112	4	12	2	4	4
SA-1135	4	12	2	4	4
Capsule DB1-LG2					
WF-70(N)	4	12	2	4	4
WF-112	4	12	2	4	4
SA-1135	4	12	2	4	4

*N - Weld material from a Midland-1 nozzle drop-out.

**6 - Weld material from a TMI-2 nozzle drop-out.

***9 - Weld material from a OC-3 nozzle drop-out.

fracture toughness specimens of 0.394, 0.500, and 0.936 inch thickness, respectively. The 0.394TCT and 0.500TCT specimens are rectangular, and the 0.936TRCT is round; they are modifications of ASTM E 399-81 and E 813-81 specimen geometry. The number of Charpy and tension test specimens per weld per capsule is adequate to characterize the impact toughness and tensile properties for each weld metal and irradiation condition. Other related development programs are expected to generate sufficient information to properly identify the methods (i.e., static versus dynamic) and test temperatures at which these SUPCAP compact fracture specimens are tested. The combination of compact fracture specimens is believed to be adequate to confirm the toughness curves.

3.3.2. SUPCAP Design

The cylindrical SUPCAPS, like the B&W 177-FA plant-specific RVSP capsules described previously, contain Charpy V-notch, tension test, and compact fracture specimens as well as neutron dosimeters and thermal monitors. The unique advantage of the cylindrical capsule is that it allows for easy capsule replacement and for uniform specimen temperatures. Aluminum alloy spacers hold the specimens, neutron dosimeters, and thermal monitors in place and fill the gaps within the capsule. The remaining spaces are helium-filled. The capsules are locked in place in a holder tube assembly.

The type R-1 capsules were designed before the R-2s and used the type of tension test specimens found in the standard capsule design of the B&W 177-FA RVSPs. When the R-2 SUPCAPS were designed, it was recognized that the standard-sized tension test specimens were not necessary (see Figure C-1). The use of subsize Charpy-sized tension test specimens (see Figure C-2) permitted the inclusion of additional 5 tension test specimens and 3 compact fracture specimens.

Each capsule contains specimens from 3 different weld metals. The weld metals and distribution of specimens per weld are described in Table 3-10. The tension test, Charpy V-notch, and compact fracture specimens are described in Appendix C. The materials contained in the capsules are described in Appendix B. Each capsule also contains neutron dosimeters to measure fluence and thermal monitors to measure the maximum irradiation temperature. The neutron dosimeters and thermal monitors are described below. The arrangements of the specimens, dosimeters, and temperature monitors within the capsules are illustrated in Figures 3-18 and 3-19.

3.3.2.1. Physical Characteristics of the SUPCAPS

The capsule is designed to maintain specimens at temperatures within ±25F of the reactor vessel temperature at the 1/4T vessel wall location. Figure 3-7 illustrates the calculated vessel wall temperature distribution for steady-state normal operation. The capsule heat transfer analysis accounts for the differences in thermal properties of the materials and the helium-filled gaps between internal components of the capsule. Conservative maximum temperatures were calculated for each different cross section within the capsule and these were within the upper bound 25F of the vessel 1/4T temperature. The coolant temperature serves as the lower bound and is also within 25F of the vessel temperature at 1/4T.[21]

3.3.2.2. SUPCAP Dosimetry

Each capsule contains dosimeter tubes, which in turn contain neutron dosimeter wires of a sufficient variety to measure fast neutron fluence (time integrated flux), fast neutron spectrum, and thermal neutron fluence. A variety of neutron dosimeters were chosen in accordance with ASTM Standard Recommended Practice E 419-73[89] and E 482-82.[90] The neutron dosimeters are distributed throughout the capsule to measure the neutron fluence at various locations.

Table 3-11 lists the neutron dosimetry and provides energy range and shielding requirements. The gadolinium (shield) thickness of 20 to 50 mils was sized to provide sufficient neutron absorption to effectively eliminate competing reactions (lower bound) and to prevent significant absorption of fast neutrons (upper bound). The neutron dosimeters, along with their shielding, are then stacked in aluminum alloy holder tubes.

3.3.2.3. SUPCAP Thermal Monitors

Thermal monitors are distributed throughout the capsule to measure specimen temperatures. Each set of thermal monitors contains 3 to 5 low-melting-point elements or eutectic alloys whose melting points range from 580 to

Table 3-11. Supplementary Weld Metal Surveillance Capsule Dosimetry

Neutron-Sensitive Element	Shield	Reaction Cross-Section Threshold Energy	Product Isotope
Long Tube (TMI2-LG1,-LG2)[91]			
^{59}Co	Cd-foil	0.5 eV	5.3 yr ^{60}Co
^{237}Np	Gd	0.5 MeV	Appropriate fission products
^{238}U	Gd	1.1 MeV	Appropriate fission products
^{58}Ni	Gd	2.3 MeV	71d ^{58}Co
^{54}Fe	Gd	2.5 MeV	314d ^{54}Mn
^{63}Cu	Gd	6.1 MeV	5.3 yr ^{60}Co
^{59}Co	None	Thermal	5.3 yr ^{60}Co
Short Tube (TMI2-LG1,-LG2)[91]			
^{237}Np	Gd	0.5 MeV	Appropriate fission products
^{238}U	Gd	1.1 MeV	Appropriate fission products
^{54}Fe	Gd	2.5 MeV	314d ^{54}Mn
Short Tube Type DA (CR3-LG1,-LG2; DB1-LG1,-LG2)[92,93]			
^{59}Co	Gd	0.5 eV	5.3 yr ^{60}Co
^{237}Np	Gd	0.5 MeV	Appropriate fission products
^{238}U	Gd	1.1 MeV	Appropriate fission products
^{58}Ni	Gd	2.3 MeV	71d ^{58}Co
^{54}Fe	None	2.5 MeV	314d ^{54}Mn
^{59}Co	None	Thermal	5.3 yr ^{60}Co
Short Tube Type DB (CR3-LG1,-LG2; DB1-LG1,-LG2)[92,93]			
^{237}Np	Gd	0.5 MeV	Appropriate fission products
^{238}U	Gd	1.1 MeV	Appropriate fission products
^{58}Ni	Gd	2.3 MeV	71d ^{58}Co
^{54}Fe	Gd	2.5 MeV	314d ^{54}Mn
^{63}Cu	Gd	6.1 MeV	5.3 yr ^{60}Co

Table 3-11. Supplementary Weld Metal Surveillance Capsule Dosimetry (Cont'd)

Neutron-Sensitive Element	Shield	Reaction Cross-Section Threshold Energy	Product Isotope
Long Tube Type DC (CR3-LG1,-LG2; DB1-LG1,-LG2)(92,93)			
^{59}Co	Gd	0.5 eV	5.3 yr ^{60}Co
^{237}Np	Gd	0.5 MeV	Appropriate fission products
^{238}U	Gd	1.1 MeV	Appropriate fission products
^{58}Ni	Gd	2.3 MeV	71d ^{58}Co
^{54}Fe	Gd	2.5 MeV	314d ^{54}Mn
^{63}Cu	Gd	6.1 MeV	5.3 yr ^{60}Co
^{59}Co	None	Thermal	5.3 yr ^{60}Co

621F. By determining which monitors have melted, the peak temperature at various locations within the capsule is determined. Table 3-12 lists the thermal monitors and their respective melting temperatures.

3.3.3. Unirradiated Baseline Data

The unirradiated baseline data needed to support the evaluation of the irradiated capsule data from the SUPCAPS will be obtained from two sources. The primary source for these data are sets of specimens that have been prepared from the same weld metal used in the SUPCAPS. These sets of specimens are similar to those included in the capsules but of a larger quantity to optimally expand the data base. The type and number of specimens of each material are described in Table 3-13.

Some material in excess of the needs of the program was provided to the HSST program to obtain test reactor irradiation data. Since this program would be obtaining baseline unirradiated data of the same type as needed by the SUPCAP program, it was decided not to duplicate the efforts of the HSST program. The sources of the baseline data for the eight welds in the SUPCAPS are identified in Table 3-14.

3.4. Higher Fluence Supplementary Weld Metal Surveillance Capsules

3.4.1. Introduction

The Higher Fluence Supplementary Weld Metal Surveillance Capsules (HUPCAPS) are included in the MIRVP to (1) provide for additional B&W-designed irradiation capsules to expand and enlarge the compact fracture toughness data base; (2) provide for an irradiation capsule of Westinghouse-design for correlation of irradiation data in the Westinghouse neutronic environment with the B&W 177-FA environment; and, (3) provide capsules for a weld metal annealing response investigation. Weld metals to be irradiated include Linde 80 welds from the current B&W Owners Group inventory, reconstituted Charpy specimens from Westinghouse RVSPs, and Linde 80 weld metals from other sources including a Midland Unit 1 reactor vessel circumferential weld (WF-70). The HUPCAPs will be irradiated at Crystal River-3, Davis-Besse, and Surry Unit 2. There are a total of 8 capsules in the HUPCAP program and they are designated A1, A2, A3, A4, A5, L1, L2, and W1. Capsules A1 through A4 add weld metal high-fluence compact fracture data to the data base. The A5 capsule provides

Table 3-12. SUPCAP Thermal Monitors

Capsule	Thermal Monitor	97.5% Pb 2.5% Ag	97.5% Pb 1.5% Ag 1.0% Sn	98.8% Cd 1.2% Cu	Pure Cd	Pure Pb
	Melting Point:	580°F	588°F	598°F	610°F	621°F
DB1-LG1	TS11-TS15[1]	X	X	X	X	X
	TL10[2]	X	X	X	X	X
DB1-LG2	TS21-TS15	X	X	X	X	X
	TL20	X	X	X	X	X
CR3-LG1	TS11-TS15	X	X	X	X	X
	TL10	X	X	X	X	X
CR3-LG2	TS21-TS25	X	X	X	X	X
	TL20	X	X	X	X	X
TMI2-LG1	ST1-ST6	X	N/A[3]	X	X	N/A
	LT1	X	X	X	X	X
TMI2-LG2	ST7-ST12	X	N/A	X	X	N/A
	LT2	X	X	X	X	X

[1]ST or TS - Short thermal monitor (3 fusible alloys)

[2]LT or TL - Long thermal monitor (5 fusible alloys)

[3]N/A - Not applicable

Table 3-13. Matrix of B&W Unirradiated Control Specimens for Welds in the SUPCAP Program

Weld Metal	Tension Test	Charpy	0.394 TCT	0.500 TCT	0.936 TRCT	0.936 TCT	2.000 TCT
SA-1526	4	22	5	8	5	--	2
WF-112	4	22	5	8	5	--	2
WF-67	4	22	5	8	6	2	--
WF-25(9)	8	44	10	16	10	--	4

Table 3-14. Identification of Programs for the Unirradiated Control Specimens of the SUPCAP Program Welds

Weld Metal	Program
WF-70(N)	HSST Task 3
WF-112	SUPCAP
SA-1585	HSST Task 3
SA-1526	SUPCAP
WF-25(6)	HSST Tasks 2 and 3
WF-67	SUPCAP
WF-25(9)	SUPCAP
SA-1135	HSST Task 3

the irradiation of reconstituted and previously irradiated WOL specimens to allow testing of specimens well ahead of vessel needs. The L1 and L2 capsules provide definitive information on annealing response for this class of materials. Benchmaıking data will be provided by irradiating capsule W1 in Surry Unit 2. This capsule contains material irradiated in B&W reactors and will therefore provide comparison of irradiation data from a Westinghouse and a B&W PWR.

The HUPCAPS are similar in design to the SUPCAPS with the exception of capsule W1 which is of the Westinghouse design. Table 3-15 identifies the weld metals irradiated in each capsule as well as the distribution of specimens. The compact fracture toughness specimens are rectangular or round and are modifications of ASTM E 399-81 and E 813-81 specimen geometry.

3.4.2. HUPCAP Dosimetry

Each capsule contains neutron dosimetry in accordance with ASTM Standard E844-86.[94] The neutron dosimeters are distributed throughout the capsule to measure the neutron fluence at various locations. Table 3-16 lists the neutron dosimeters and provides their energy range and shielding requirements. The gadolinium (shield) thickness of 20 to 80 mils was sized to provide sufficient neutron absorption to effectively eliminate competing reactions (lower bound) and to prevent excessive absorption of fast neutrons (upper bound).

The B&W-design HUPCAPS will contain at least 6 sets of flux wires and one full-diameter steel block containing radially spaced dosimeter wires to measure the neutron flux gradient through the cross-section of the capsule. At the center of the dosimeter block is a gadolinium case containing two complete sets of dosimeter wire plus a HAFM.*

The Westinghouse-design HUPCAP will contain approximately 5 sets of flux wires and one set of dosimetry identical to that in the gadolinium case in the center of the dosimeter block in the B&W-design HUPCAPS.

Figures 3-20 and 3-21 show the neutron dosimeter locations in the capsules.

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*Helium accumulation flux monitor.

Table 3-15. HUPCAPS -- Material and Specimens Per Capsule

Capsule	Weld Metal	Specimens: Tension Test	Specimens: Charpy	Specimens: 0.936 TRCT	Specimens: 0.417 TCT	Specimens: 3-Point Bend
A1	WF-25(6)*	4	12	5	-	-
	WF-67	4	12	5	-	-
	WF-70(N)**	4	12	5	-	-
A2	SA-1101	3	9	3	-	-
	SA-1135	3	9	4	-	-
	SA-1526	3	9	4	-	-
	SA-1585	3	9	4	-	-
A3	WF-70(B)***	4	12	5	-	-
	WF-182-1	4	12	5	-	-
	SA-1484	4	12	5	-	-
A4	WF-25(6)	4	12	5	-	-
	WF-67	4	12	5	-	-
	WF-70(N)	4	12	5	-	-
L1	WF-25(6)	4	12	5	-	-
	WF-67	4	12	5	-	-
	WF-70(N)	4	12	5	-	-
L2	WF-25(6)	4	12	5	-	-
	WF-67	4	12	5	-	-
	WF-70(N)	4	12	5	-	-
W1	WF-70(N)	3	10	-	4	3
	SA-1526	3	10	-	4	3
	SA-1585	3	10	-	4	3
A5	WF-209-1(I)****					
	SA-1101(I)	(to be determined)				
	SA-1263(I)					

*6 - Weld material from a TMI-2 nozzle drop-out.

**N - Weld material from a Midland-1 nozzle drop-out.

***B - Weld material from a Midland-1 vessel beltline.

****I - Irradiated material.

Table 3-16. Higher Fluence Supplementary Weld Metal Surveillance Capsule Dosimetry

Neutron-Sensitive Element	Shield	Reaction Cross-Section Threshold Energy	Product Isotope
Short Tube Type DA*			
^{238}U	Gd	1.1 MeV	Appropriate fission products
^{58}Ni	Gd	2.3 MeV	71d ^{58}Co
^{63}Cu	Gd	6.1 MeV	5.3y ^{60}Co
^{54}Fe	Gd	2.5 MeV	314d ^{54}Mn
^{93}Nb	Gd	0.1 MeV	13.6y ^{93}Nb
^{46}Ti	Gd	3.9 MeV	85d ^{46}Sc
Short Tube Type DB*			
^{59}Co	None	Thermal	5.3y ^{60}Co
^{58}Ni	Gd	2.3 MeV	71d ^{58}Co
^{63}Cu	Gd	6.1 MeV	5.3y ^{60}Co
^{54}Fe	Gd	2.5 MeV	314d ^{54}Mn
^{238}U	Gd	1.1 MeV	Appropriate fission products
^{237}Np	Gd	0.5 MeV	Appropriate fission products
Long Tube Type DC*			
^{237}Np	Gd	0.5 MeV	Appropriate fission products
^{237}Np	Gd	0.5 MeV	Appropriate fission products
^{54}Fe	Gd	2.5 MeV	314d ^{54}Mn
^{58}Ni	Gd	2.3 MeV	71d ^{58}Co
^{63}Cu	Gd	6.1 MeV	5.3y ^{60}Co
^{59}Co	Gd	0.5 eV	5.3y ^{60}Co
^{59}Co	None	Thermal	5.3y ^{60}Co
^{238}U	Gd	1.1 MeV	Appropriate fission products
^{238}U	Gd	1.1 MeV	Appropriate fission products

Table 3-16. Higher Fluence Supplementary Weld Metal Surveillance Capsule Dosimetry (Cont'd)

Neutron-Sensitive Element	Shield	Reaction Cross-Section Threshold Energy	Product Isotope
Long Tube Type DD*			
^{46}Ti	Gd	3.9 MeV	85d ^{46}Sc
^{237}Np	Gd	0.5 MeV	Appropriate fission products
^{54}Fe	Gd	2.5 MeV	314d ^{54}Mn
^{58}Ni	Gd	2.3 MeV	71d ^{58}Co
^{63}Cu	Gd	6.1 MeV	5.3y ^{60}Co
^{59}Co	Gd	0.5 eV	5.3y ^{60}Co
^{59}Co	None	Thermal	5.3y ^{60}Co
^{238}U	Gd	1.1 MeV	Appropriate fission products
^{93}Nb	Gd	0.1 MeV	13.6y ^{93m}Nb
Full-Section Steel Block*			
^{46}Ti	Gd	3.9 MeV	85d ^{46}Sc
^{93}Nb	Gd	0.1 MeV	13.6y ^{93m}Nb
^{54}Fe	Gd	2.5 MeV	314d ^{54}Mn
^{58}Ni	Gd	2.3 MeV	71d ^{58}Co
^{63}Cu	Gd	6.1 MeV	5.3y ^{60}Co
^{59}Co	Gd	0.5 eV	5.3y ^{60}Co
^{238}U	Gd	1.1 MeV	Appropriate fission products
^{237}Np	Gd	0.5 MeV	Appropriate fission products
^{9}Be	Gd	1.5 MeV	Helium accumulation

Table 3-16. Higher Fluence Supplementary Weld Metal Surveillance Capsule Dosimetry (Cont'd)

Neutron-Sensitive Element	Shield	Reaction Cross-Section Threshold Energy	Product Isotope
Westinghouse Capsule (Radiometric)**			
^{54}Fe	None	2.5 MeV	314d ^{54}Mn
^{58}Ni	None	2.3 MeV	71d ^{58}Co
^{63}Cu	None	6.1 MeV	5.3y ^{60}Co
^{59}Co	None	Thermal	5.3y ^{60}Co
^{59}Co	Cd	0.5 eV	5.3y ^{60}Co
^{93}Nb	Cd	0.1 MeV	13.6y ^{93m}Nb
Westinghouse Capsule (Block)**			
^{54}Fe	Gd	2.5 MeV	314d ^{54}Mn
^{63}Cu	Gd	6.1 MeV	5.3y ^{60}Co
^{58}Ni	Gd	2.3 MeV	71d ^{58}Co
^{59}Co	Gd	0.5 eV	5.3y ^{60}Co
^{93}Nb	Gd	0.1 MeV	13.6y ^{93m}Nb
^{46}Ti	Gd	3.9 MeV	85d ^{46}Sc
^{237}Np	Gd	0.5 MeV	Appropriate fission products
^{238}U	Gd	1.1 MeV	Appropriate fission products

*For capsules A1, A2, A3, A4, A5, L1, and L2

**For capsule W1

Figure 3-20a HUPCAP Capsules A1, A2, A3, A4, L1, and L2

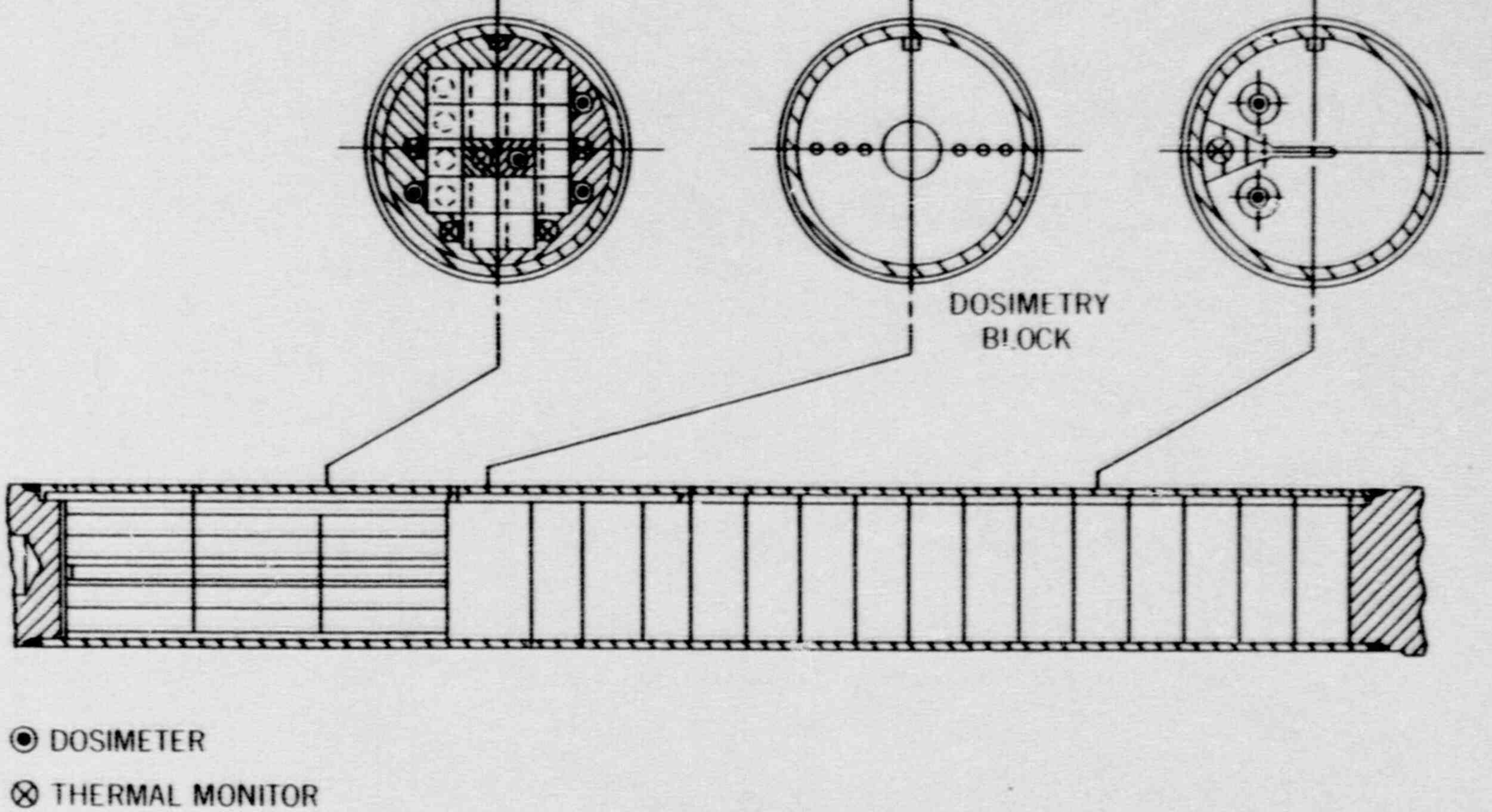


Figure 3-20b HUPCAP Capsule A5

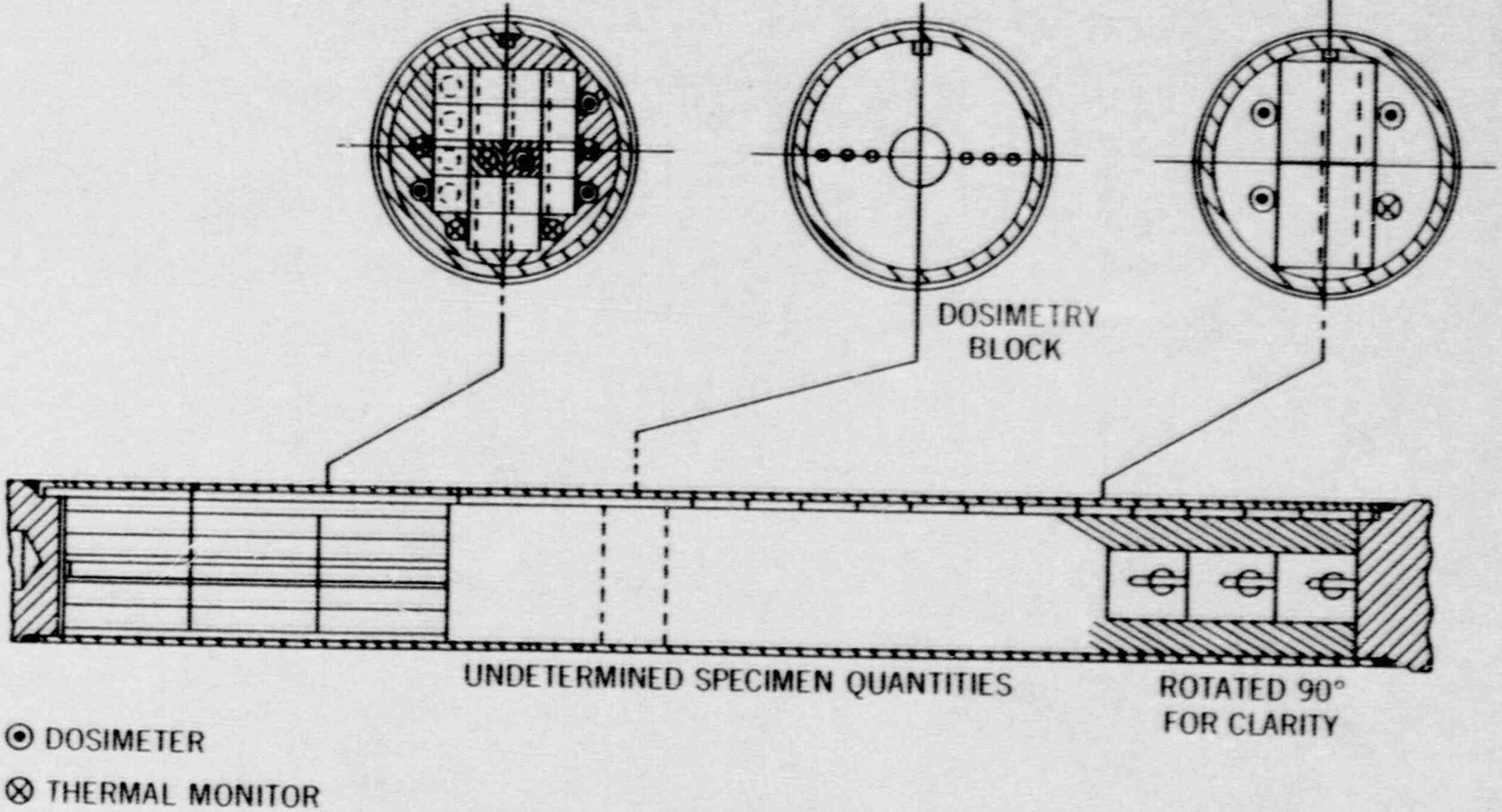
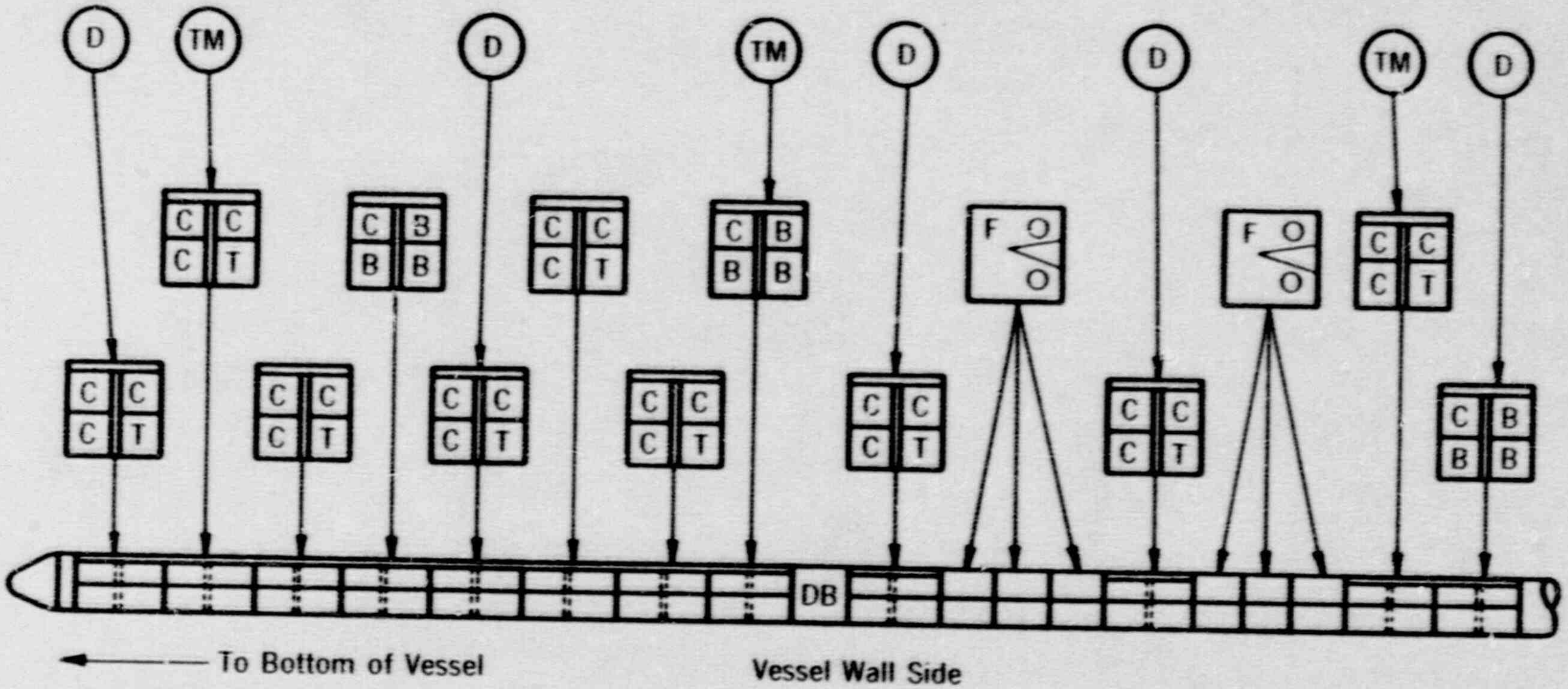


Figure 3-21 Placement of Specimens, Neutron Dosimeters and Thermal Monitors in HUPCAP W1

LEGEND

B - Slow-bend(3-point) Specimen
C - Charpy V-notch Specimen
D - Dosimeter (Radiometric) Set
DB - Dosimeter Block
F - 0.417T Compact Fracture Specimen
T - Tension Test (Subsize) Specimen
TM - Thermal Monitor Set



3.4.3. HUPCAP Thermal Monitors

Thermal monitors are distributed throughout the capsule. The thermal monitors contain low-melting-point elements or eutectic alloys whose melting points range from 580 to 621F. By determining which monitors have melted, the peak temperature at various locations within the capsule is determined. Table 3-17 lists the thermal monitors and their respective melting temperatures. The locations of these monitors are shown in Figures 3-20 and 3-21 for the B&W and Westinghouse capsules, respectively.

3.5. Test Reactor Irradiations

The high flux available in a test reactor makes it possible to achieve high fluences in specimens in a relatively short time. However, anticipating that the neutron damage mechanism in this high flux and particular neutron energy spectrum and temperature may be different than that experienced in PWRs, the B&WOG evaluated the mechanical properties and fracture toughness of Linde 80 weld metal irradiated in a high-flux neutron field at ORNL as part of the HSST program.[95] These property values were compared to those obtained from the first SUPCAP irradiations. The comparisons of the Charpy impact and tension test data indicate that a difference exists between some test reactor and power reactor data. No differences appear to exist in a similar comparison of the fracture toughness data; however, these relationships may change at extended irradiations, as one theory suggests. This is all the more reason that reactor vessel analyses be based primarily on power reactor data. The uncertainty in using power reactor data is less than that associated with test reactor data since the power reactor irradiation environment is that which the reactor vessel experiences. This is not to imply that the test reactor data is in error, however, the test reactor neutron flux and spectrum are usually significantly different from that of the power reactor. The effect of these differences is being evaluted in other programs and will be considered with the application of these data.

As a contingency, in the event that a host reactor should experience an extended outage, a test reactor irradiation program will be planned. A suitable test reactor will be located. Factors to be considered include availability, spectral characteristics, and operating temperature. Specimens will be irradiated to characterize the acceptability of such irradiation

Table 3-17. HUPCAP Thermal Monitors

Alloy (wt%)	Melting Point, F
97.5 Pb-2.5 Ag	580
97.5 Pb-1.5 Ag-1.0 Sn	588
98.8 Cd-1.2 Cu	598
Cd	610
Pb	621

experiments. Plans will include irradiation of a well-characterized material, such as WF-70, in the optimum test reactor.

3.6. Unirradiated Control Data

The unirradiated baseline data needed to support the evaluation of the irradiated capsule data from the various capsules has been prepared from the same weld metal used in the plant-specific RVSPs and is described in each plant's RVSP report, where applicable. Additional data has been obtained on specimens from the weld metals used in the SUPCAPS. The type and number of specimens of each material are described in Table 3-13. Baseline data is also available for the eight welds in the SUPCAP program from the HSST baseline studies and is identified in Table 3-14. Unirradiated SA-1484 has not been tested; specimens are being fabricated as part of the HUPCAP effort for this purpose. Table 3-18 summarizes this information.

3.7. Comparison of the B&W and Westinghouse Operating Parameters

The integrated surveillance program approach is dependent on the similarity of exposure conditions in order to compare the specific materials used in each plant. Differences in plant characteristics and operating parameters will be considered and accounted for to ensure the applicability of the data generated.

As noted earlier, the relative neutron flux energy spectrum, the irradiation dose rate, and the irradiation temperature are important parameters in evaluating the similarity of reactors. Small neutron flux spectral differences between plants are expected. Sufficient data, obtained from both B&W and Westinghouse plant irradiation, will be available to draw the required comparisons. This issue is discussed in fuller detail in Section 4.

3.8. Capsule Irradiation

3.8.1. General Discussion

Each of the vessel beltline regions in this MIRVP contain welds that were fabricated using combinations of several Mn-Mo-Ni filler wire heats and Linde 80 flux lots. All of these materials will be treated as members of a family of materials with closely correlatable properties. That is, the chemical composition of a Linde 80 weld metal is believed to determine the effect of

Table 3-18. Matrix of Unirradiated Control Specimens for Welds in the MIRVP

Weld Metal	Program	Tension Test	CVN	0.394 TCT	0.500 TCT	0.936 TRCT	0.936 TCT	2.000 TCT	4.000 TCT
WF-209-1	Z1 RVSP	6	23	--	---	--	---	---	---
	Z2 RVSP	--	--	--	---	--	---	---	---
	OC2 RVSP	6	15	--	---	--	---	---	---
	OC3 RVSP	6	21	--	---	--	---	---	---
	CR3 RVSP	--	--	--	8-9	--	6	---	---
WF-70(N)*	HSST Task 3	6	24	--	3	--	2 (.8TCT)	1 (1.6TCT)	---
WF-70(B)*	WF-70**	--	--	--	---	--	---	---	---
SA-1526	S1 RVSP	6	23	--	---	--	---	---	---
	SUPCAP	4	22	5	8	5	---	2	---
WF-25*	TMI1 RVSP	6	18	--	---	--	---	---	---
	SUPCAP	8	44	10	16	10	---	4	---
	HSST Tasks 2 & 3	15	50	--	10	--	8 (.8TCT)	8 (1.6TCT)	5
SA-1263	PB1 RVSP	6	21	--	---	--	---	---	---
SA-1585*	HSST Task 3	9	32	--	4	--	4 (.8TCT)	4 (1.6TCT)	3
WF-193	PB2 RVSP	6	20	--	---	--	---	---	---
	ANO1 RVSP	6	27	--	---	--	---	---	---
	RS1 RVSP	5	14	--	8	--	---	---	---
WF-112*	OC1 RVSP	6	21	--	---	--	---	---	---
	SUPCAP	4	22	5	8	5	---	2	---
SA-1101	TP3 RVSP	6	21	--	---	--	---	---	---
	HSST Task 2	--	--	--	---	--	---	---	---
SA-1094*	TP4 RVSP	6	18	--	---	--	---	---	---
SA-1036	G RVSP	--	--	--	---	--	---	---	---
SA-1135*	HSST Task 3	9	27	--	4	--	3 (.8TCT)	3 (1.6TCT)	2
WF-182-1	DB1 RVSP	5	19	--	8	--	4	---	---
WF-67	SUPCAP	4	22	5	8	6	2	---	---
SA-1484*	HUPCAP	6	12	--	---	5	---	---	---

*Welds made from the same weld wire as the previous weld but using a different flux lot.

**Unirradiated data to be obtained from the "WF-70 Issue Resolution," Phase XIV, Task J of B&W Owners Group Reactor Vessel Integrity Program.

neutron irradiation upon its fracture toughness properties. Since chemical composition of the controlling metallic elements of a weld deposit is mainly determined by the chemical composition of the consumable weld wire, welds which are made with the same wire heat are considered "surrogates" of each other. For example, weld material WF-209-1 can be considered as a surrogate weld for WF-70 since both welds were fabricated with wire heat number 72105.[96] Data for these surrogate welds can easily be pooled together for correlation with the other Linde 80 family of welds. The value of plant-specific capsule programs is greatly enhanced by evaluating all pertinent welds in the MIRVP fabricated with the same wire/flux combination and the surrogate welds. In some instances, the surveillance weld from one plant, although not in the vessel weldment of that plant, may be a controlling (or otherwise important) weld for another plant. Therefore, the plant-specific capsule withdrawal and testing schedules were adjusted in the MIRVP to benefit from this understanding, especially where a significant body of data exists.

The insertion and withdrawal schedules (that follow) for each of the B&W and Westinghouse plant-specific RVSPs have been prepared in accordance with ASTM E 185-82 and the criteria for integrated surveillance programs of 10CFR50, Appendix H, paragraph II.C. All capsules in the plant-specific RVSPs will be irradiated, but some of the capsules will not be tested since they would not provide sufficiently significant information. The capsules that will not be tested are primarily those that do not contain weld fracture toughness specimens. A comparison of each plant-specific RVSP with ASTM E185-82 requirements is given in Appendix E, Table E-3.

3.8.2. B&W Plant-Specific RVSPs

The B&W plant-specific RVSP portion of the MIRVP is nearing completion. A number of standby capsules are being irradiated to satisfy regulatory requirements. The 2 capsules being irradiated that are planned to be tested are

1. OC3-D (Oconee-3, IS-EOL), to be removed at CR-3, EOC 7
2. TE1-D (Davis-Besse, IS-EOL), to be removed at DB, EOC 6

The remaining capsules either do not contain weld materials, or are not expected to contribute significantly to the data base. It is planned, therefore, that these capsules be withdrawn, as scheduled, and not tested. They will be stored until required otherwise.

3.8.3. Westinghouse Plant-Specific RVSPs

Most of the Westinghouse plant-specific RVSP capsules do not contain weld material pertinent to the plant for which they are being irradiated. In some cases, a surrogate weld is included in the RVSP. Data from a plant-specific RVSP can be used for another plant where the weld metal is relevant thereby reducing the number of capsules to be tested. However, RVSP capsules must continue to be used for reactor vessel neutron dosimetry unless alternate methods, such as cavity dosimetry, are provided.

The individual Westinghouse-designed plant-specific RVSPs are discussed below. Where a plant is not provided with a relevant weld metal in its RVSP, the relevant material is sought in another plant. The order of preference in seeking out this material is (1) plants in the same Westinghouse grouping, (2) other Westinghouse-designed plants, and (3) other PWRs. If relevant material data is not found, the Linde 80 "family of materials" correlation approach will be applied.

R. E. Ginna Unit 1

All capsules in the R. E. Ginna RVSP contain SA-1036 weld material which is a surrogate for SA-847, a beltline material in R. E. Ginna and Point Beach Unit 1. SA-1135 weld material is also a surrogate for SA-847 and will be irradiated to an estimated fluence of 2.2E19 in the SUPCAPS. Capsule P will be irradiated to the estimated IS-EOL fluence of 4.1E19 to fulfill regulatory requirements. Capsules S and N are standbys and are scheduled to be withdrawn at one to two times the IS-EOL fluence and stored (without testing).

Point Beach Unit 1

Capsules in the Point Beach Unit 1 RVSP contain SA-1263 weld material which is a surrogate for SA-1585 and SA-1650. Beltline weld materials of concern in Point Beach Unit 1 are SA-1101 and SA-847. SA-847 material is "covered"* by surrogate materials SA-1036 in Ginna and SA-1135 in the SUPCAPS. SA-1101 material is in Turkey Point Unit 3 capsules and SA-1094, a surrogate material for SA-1101, is in Turkey Point Unit 4. SA-1263 weld surveillance data benefits Surry Units 1 and 2 and Oconee Unit 1. SA-1263 weld material is covered in the SUPCAPS and HUPCAPS, therefore no additional data is required for this weld material. Capsules P and N only contain weld metal Charpy V-notch specimens and would add little to the data base; these capsules are designated as standbys and are scheduled to be withdrawn at one to two times the IS-EOL fluence and stored.

Point Beach Unit 2

Point Beach Unit 2 RVSP capsules contain WF-193 weld material which is a surrogate for WF-112 and WF-154. The beltline material of concern is SA-1484. HUPCAP A3 will provide data on SA-1484 weld material with a fluence of 1.7E19. A surrogate material for SA-1484 is WF-67, which is well characterized in the SUPCAPS and HUPCAPS. Capsule S contains weld WOL specimens, and capsules P and N only contain Charpy V-notch weld specimens; these capsules are designated as standbys and are scheduled to be withdrawn at one to two times the IS-EOL fluence and stored.

Surry Unit 1

Surry Unit 1 RVSP capsules contain SA-1526 weld material for which WF-25 is a surrogate. The beltline materials of interest are SA-1585 and SA-1526. SA-1585 is covered in the SUPCAPS, HUPCAPS, and Point Beach Unit 1. SA-1526 and WF-25 weld material are covered in the SUPCAPS and HUPCAPS, therefore, no additional data is required. Capsules S, U, Y, X, and Z are designated as standbys and are scheduled to be withdrawn at one to two times the IS-EOL (or as otherwise needed) and stored.

Surry Unit 2

Surry Unit 2 RVSP capsules do not contain Linde 80 weld material. The RVSP weld material was fabricated by Rotterdam and is not believed to be as

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*The term "covered" is here taken to denote that material properties will be (or were) obtained by irradiation and test of another weld material that is of the same wire heat and different flux lot (surrogate material).

susceptible to irradiation damage as the Linde 80 welds. Rotterdam weld material is only in the Surry Unit 2 beltline. However, surveillance data to date has shown that the shift in transition temperature and the drop in Charpy upper-shelf energy are comparable to predictions using Regulatory Guide 1.99, revision 2. Two capsules, V and X, have been evaluated at fluences of 3.0E18 and 1.9E19. The third capsule that was withdrawn, W, was only evaluated for dosimetry at a fluence of 6.0E18. Capsules Y and Z contain weld metal tension test, Charpy V-notch, and WOL specimens. It is recommended that capsule Y be withdrawn and tested at a fluence of 2.7E19 and capsule Z be designated as a standby capsule with a target fluence of 3.4E19 for possible testing, if necessary. The remaining capsules will be withdrawn at times to satisfy regulatory requirements and for dosimetry; specimens do not need to be tested. The beltline Linde 80 weld of interest to the B&W Owners Group in Surry Unit 2 is SA-1585. This material is covered in the SUPCAPS, HUPCAPS, and in Point Beach Unit 1 capsules.

Turkey Point Unit 3

Turkey Point Unit 3 RVSP capsules contain SA-1101 weld material which is a surrogate for SA-1094 and SA-1769. SA-1101 is a beltline material for Turkey Point Units 3 and 4 and Point Beach Unit 1. Turkey Point Units 3 and 4 have an NRC approved integrated surveillance program.[97] The only capsules to be tested in accordance with ASTM E 185 requirements are those (in each unit) that contain weld metal specimens. HUPCAP A2 will provide data on SA-1101 weld material with a fluence of 3.0E19. SA-1769 is a beltline material in Crystal River-3 and Zion Unit 2. Capsule X contains weld metal tension test, Charpy V-notch, and WOL specimens. Capsules U, W, Y, and Z do not contain weld specimens. Therefore, capsule X will be irradiated to the estimated IS-EOL fluence of 2.8E19 to fulfill regulatory requirements for Turkey Point Unit 3; all remaining capsules will be designated as standbys and irradiated to satisfy regulatory requirements or for dosimetry and do not need to be tested.

Turkey Point Unit 4

Capsules in the Turkey Point Unit 4 RVSP contain SA-1094 weld material which is a surrogate for SA-1101 and SA-1769. The Turkey Point Unit 3 discussion is also applicable for this RVSP material. Capsules V and X contain weld metal tension test, Charpy V-notch, and WOL specimens. Capsules U, W, Y, and Z do not contain weld specimens. Weld data from capsules in Turkey Point Unit 3 and the HUPCAPS can be used to cover the T/4-EOL and IS-EOL data requirements for Turkey Point Unit 4. HUPCAP A2 will provide data on SA-1101 weld material with a fluence of 3.0 E19. Consideration will also be given to maximizing the fluence that can be achieved for capsules V and X. All other capsules in this RVSP will be designated as standbys and irradiated to satisfy regulatory requirements or for dosimetry and do not need to be tested.

Zion Units 1 and 2

The Zion Units 1 and 2 RVSP capsules contain WF-209-1 weld material which is a surrogate for WF-70. The beltline material of concern in these plants is WF-70. WF-70 and WF-209-1 weld material is well characterized in the SUPCAPS, HUPCAPS, and the Crystal River Unit 3 and Oconee Units 2 and 3 RVSPs. Zion-1 capsule Z contains weld metal tension test, Charpy V-notch, and WOL specimens. Capsules X, W, S, and V only contain Charpy V-notch and tension test weld metal specimens. Therefore, it is recommended that Zion-1 capsule Z be irradiated to a fluence of 2.2E19 to fulfill regulatory requirements of the IS-EOL for these plants. All remaining capsules in Zion-1 and the Zion-2 capsules will be designated as standbys and irradiated to satisfy regulatory requirements or for dosimetry and do not need to be tested.

3.8.4. Irradiation Schedule

The irradiation schedule for this integrated surveillance program includes the plant-specific capsules for the B&W- and Westinghouse-designed plants and the SUPCAPS and HUPCAPS. All the irradiations, with the exception of capsule W1 and Westinghouse plant-specific capsules, are performed in the B&W host reactors, Crystal River-3 and Davis-Besse. Capsule W1, an irradiation capsule of Westinghouse-design, is irradiated in Surry Unit 2. The Westinghouse plant-specific capsules are irradiated in their respective plants. The schedules are shown in Tables 3-19 through 3-21.

Table 3-19. Capsule Insertion and Withdrawal Schedule for Crystal River Unit 3

Holder Tube	Location in Holder Tube	Withdraw	Insert	Capsule Status
Installed at Initial Fuel Load				
XW	Top		CR3-B (WC)	
XW	Bottom		CR3-D (WC)	
End of First Fuel Cycle (1A)				
WZ	Top		CR3-LG1 (WC)	
WZ	Bottom		CR3-LG2 (WC)	
ZY	Top		CR3-C (W)	
ZY	Bottom		CR3-A (W)	
YZ	Top		OC2-A (W)	
YZ	Bottom		OC1-A (W)	
YX	Top		OC2-E (W)	
YX	Bottom		OC3-D (W)	
XW	Top	CR3-B (WC)	CR3-E (W)	Tested
WX	Top		OC3-B (W)	
WX	Bottom		CR3-F (WC)	
End of First Fuel Cycle (1B)				
No changes				
End of Second Fuel Cycle				
YZ	Top	OC2-A (W)	OC1-C (W)	Tested
WX	Top	OC3-B (W)	TMI1-C (W)	Tested
End of Third Fuel Cycle				
No changes				
End of Fourth Fuel Cycle				
YZ	Bottom	OC1-A (W)	OC1-B	Tested
WZ	Top	CR3-LG1 (WC)	None	Tested
WZ	Bottom	CR3-LG2 (WC) (WZ now empty)	None	Stored
End of Fifth Fuel Cycle				
WX	Top	TMI1-C (W)	OC3-C (W)	Tested
XW	Bottom	CR3-D (WC)	TMI1-B	Tested
ZY	Top	CR3-C (W)	OC3-F (W)	Tested
WZ	Top	None	OC2-B	
WZ	Bottom	None (WZ no longer empty)	CR3-LG2 (WC)	

Table 3-19. Capsule Insertion and Withdrawal Schedule for Crystal River Unit 3 (Cont'd)

Holder Tube	Location in Holder Tube	Withdraw	Insert	Capsule Status
End of Sixth Fuel Cycle				
YX	Top	OC2-E (W)	TMI2-D*	Tested
WX	Bottom	CR3-F (WC)	TMI1-F	Tested
YZ	Top	OC1-C (W)	TMI2-LG1 (WC)	Tested
YZ	Bottom	OC1-B	TMI2-LG2 (WC)	Stored
End of Seventh Fuel Cycle				
XW	Bottom	TMI1-B	TMI2-D* from YX top	2
YX	Top	TMI2-D* to XW top	A2 (WC)	
YX	Bottom	OC3-D (W)	A4 (WC)	1
WZ	Top	OC2-B	OC3-E (W)	2
End of Eighth Fuel Cycle				
ZY	Bottom	CR3-A (W)	OC1-D	2
XW	Top	CR3-E (W)	None	2
XW	Bottom	TMI2-D	None	2
WX	Top	OC3-C (W)	OC2-F	2
WX	Bottom	TMI1-F (XW now empty)	TMI1-D	2
End of Ninth Fuel Cycle				
YZ	Top	TMI2-LG1 (WC)	OC2-D	1
WZ	Bottom	CR3-LG2 (WC)	TMI2-D*	1
End of Tenth Fuel Cycle				
No changes				
End of Eleventh Fuel Cycle				
WX	Top	OC2-F	None	2
WX	Bottom	TMI1-D (WX now empty)	None	2

Table 3-19. Capsule Insertion and Withdrawal Schedule for Crystal River Unit 3 (Cont'd)

Holder Tube	Location in Holder Tube	Withdraw	Insert	Capsule Status
End of Twelfth Fuel Cycle				
YZ	Top	OC2-D	None	2
YZ	Bottom	TMI2-LG2 (WC)	None	1
WZ	Top	OC3-E (W)	None	2
WZ	Bottom	TMI2-D* (YZ and WZ now empty)	None	2
End of Thirteenth Fuel Cycle				
ZY	Top	OC3-F (W)	None	2
ZY	Bottom	OC1-D (ZY now empty)	None	2
End of Fourteenth through Sixteenth Fuel Cycle				
No changes				
End of Seventeenth Fuel Cycle				
YX	Top	A2 (WC)	None	1
YX	Bottom	A4 (WC) (All locations now empty)	None	1

(W) - Capsule contains weld metal specimens.

(WC) - Capsule contains weld metal compact fracture toughness specimens.

* - Dummy capsule.

1 - Capsule to be removed, specimens will be tested, dosimetry evaluated, and thermal monitors evaluated.

2 - Capsule to be removed and placed in storage. Dosimetry may be evaluated at this time.

Table 3-20. Capsule Insertion and Withdrawal Schedule for Davis-Besse Unit 1

Holder Tube	Location in Holder Tube	Withdraw	Insert	Capsule Status
Installed at Initial Fuel Load				
WZ	Top		AN1-B	
WZ	Bottom		RS1-B (WC)	
ZY	Top		TE1-B (WC)	
ZY	Bottom		TE1-F (WC)	
YZ	Top		AN1-A (W)	
YZ	Bottom		AN1-C (W)	
YX	Top		RS1-D (WC)	
YX	Bottom		TE1-C (W)	
XW	Top		TE1-D (WC)	
XW	Bottom		RS1-C (W)	
WX	Top		TE1-A (W)	
WX	Bottom		RS1-F (WC)	
End of First Fuel Cycle				
WZ	Top	AN1-B	DB1-LG1 (WC)	Tested
WZ	Bottom	RS1-B (WC)	RS1-E (W)	Tested
ZY	Bottom	TE1-F (WC)	DB1-LG2 (WC)	Tested
End of Second Fuel Cycle				
YX	Top	RS1-D (WC)	RS1-A (W)	Tested
End of Third Fuel Cycle				
YZ	Top	AN1-A (W)	AN1-D	Tested
ZY	Top	TE1-B (WC)	TE1-E (W)	Tested
End of Fourth Fuel Cycle				
YX	Top	RS1-A (W)	AN1-F	Stored
WZ	Top	DB1-LG1 (WC)	RS1-F from WX bottom	Tested
WX	Top	TE1-A (W)	None	Tested
WX	Bottom	RS1-F to WZ top	None	
	(WX now empty)			
End of Fifth Fuel Cycle				
WZ	Top	RS1-F (WC)	None	Tested
WZ	Bottom	RS1-E (W)	None	Stored
YZ	Top	AN1-D to XW bottom	None	
YZ	Bottom	AN1-C (W)	None	Tested
XW	Bottom	RS1-C (W)	AN1-D from YZ top	Stored
	(YZ and WZ now empty)			

Table 3-20. Capsule Insertion and Withdrawal Schedule for Davis-Besse Unit 1 (Cont'd)

Holder Tube	Location in Holder Tube	Withdraw	Insert	Capsule Status
End of Sixth Fuel Cycle				
XW	Top	TE1-D (WC)	None	1
XW	Bottom	AN1-D	None	2
YX	Bottom	TE1-C (W)	A5	2
YZ	Top		A3 (WC)	
YZ	Bottom		A1 (WC)	
WZ	Top		L2 (WC)	
WZ	Bottom (XW now empty)		L1 (WC)	
End of Seventh Fuel Cycle				
YX	Top	AN1-F	Dummy	2
End of Eighth through Tenth Fuel Cycle				
No changes				
End of Eleventh Fuel Cycle				
ZY	Top	TE1-E (W)	None	2
ZY	Bottom (ZY now empty)	DB1-LG2 (WC)	None	1
End of Twelfth Fuel Cycle				
YX	Top	Dummy	None	
YX	Bottom	A5 (WC)	None	1
YZ	Top	A3 (WC)	L2 (WC) from WZ top	1
WZ	Top	L2 (WC) to YZ top*	None	
WZ	Bottom (YX and WZ now empty)	L1 (WC)	None	1
End of Thirteenth through Fourteenth Fuel Cycle				
No changes				

Table 3-20. Capsule Insertion and Withdrawal Schedule for Davis-Besse Unit 1 (Cont'd)

Holder Tube	Location in Holder Tube	Withdraw	Insert	Capsule Status
End of Fifteenth Fuel Cycle				
YZ	Top	L2 (WC)	Dummy	1
End of Sixteenth Fuel Cycle				
No changes				
End of Seventeenth Fuel Cycle				
YZ	Top	Dummy	None	
YZ	Bottom	A1 (WC)	None	1
	(All locations now empty)			

(W) - Capsule contains weld metal specimens.

(WC) - Capsule contains weld metal compact fracture toughness specimens.

* - L2 to be annealed before reinsertion.

1 - Capsule to be removed, specimens will be tested, dosimetry evaluated, and thermal monitors evaluated.

2 - Capsule to be removed and placed in storage. Dosimetry may be evaluated at this time.

Table 3-21. Capsule Insertion and Withdrawal Schedule for the Westinghouse Plant-Specific RVSPs

Nuclear Plant	Capsule Location*	Capsule Identification	Withdraw	Insert	Capsule Status
R. E. Ginna	13°	V (WC)	EOC-1		Tested
	13°	R (WC)	EOC-4		Tested
	23°	T (WC)	EOC-7		Tested
	23°	P (WC)	EOC-26		1
	33°	S (WC)			2,5
	33°	N (WC)	EOC-20		3
	13°	N (WC)		EOC-20	1,5
Point Beach Unit 1	13°	V (WC)	EOC-1		Tested
	13°	R (WC)	EOC-5		Tested
	23°	T (WC)	EOC-11		Tested
	23°	P (W)	EOC-22		2
	33°	S (W)	EOC-3		Tested
	33°	N (W)			2,5
Point Beach Unit 2	13°	V (WC)	EOC-1		Tested
	13°	R (WC)	EOC-5		Tested
	23°	T (W)	EOC-3		Tested
	23°	P (W)	EOC-23		2
	33°	S (WC)	EOC-16		2
	33°	N (W)			2,5
Surry Unit 1	15°	T (WC)	EOC-1		Tested
	15°	V (WC)	EOC-8		Tested
	25°	S			2,5
	25°	X (WC)			2,5
	25°	Z (WC)			2,5
	35°	W	EOC-4		Tested**
	35°	Y	EOC-20		3
	15°	Y		EOC-20	2,5
	45°	U	EOC-20		3
	15°	U		EOC-20	2,5

Table 3-21. Capsule Insertion and Withdrawal Schedule for the Westinghouse Plant-Specific RVSPs (Cont'd)

Nuclear Plant	Capsule Location*	Capsule Identification	Withdraw	Insert	Capsule Status
Surry Unit 2	15^o	X (W)	EOC-1		Tested
	15^o	V (W)	EOC-8		Tested
	25^o	Y (WC)			2,5
	25^o	W (W)	EOC-4		Tested**
	25^o	U (W)			2,5
	35^o	Z (WC)	EOC-20		3
	15^o	Z (WC)		EOC-20	1,5
	35^o	T (W)	EOC-20		3
	15^o	T (W)		EOC-20	2,5
	45^o	S (W)			4
	15^o	W1 (WC)***	EOC-17	EOC-10	1
Turkey Point Unit 3	0^o	T (WC)	EOC-1		Tested
	10^o	S	EOC-4		Tested
	20^o	V (WC)			Tested
	30^o	U			4
	30^o	Y			4
	40^o	W			4
	40^o	X (WC)	EOC-12		3,6
	0^o	X (WC)	EOC-24	EOC-12	1
	40^o	Z			4
Turkey Point Unit 4	0^o	T (WC)	EOC-1		Tested
	10^o	S	EOC-3		Tested
	20^o	V (WC)	EOC-12		3,6
	10^o	V (WC)	EOC-17	EOC-12	1
	30^o	U			4
	30^o	Y			4
	40^o	W			4
	40^o	X (WC)	EOC-12		3,6
	0^o	X (WC)		EOC-12	1,5
	40^o	Z			4

Table 3-21. Capsule Insertion and Withdrawal Schedule for the Westinghouse Plant-Specific RVSPs (Cont'd)

Nuclear Plant	Capsule Location*	Capsule Identification	Withdraw	Insert	Capsule Status
Zion Unit 1	4^o	S (W)			2,5
	4^o	V (W)			2,5
	4^o	W (W)	EOC-11		3
	40^o	W (W)		EOC-11	2,5
	4^o	Z (WC)	EOC-11		3
	40^o	Z (WC)	EOC-22	EOC-11	1
	40^o	T (W)	EOC-1		Tested
	40^o	U (W)	EOC-4		Tested
	40^o	X (W)	EOC-6		Tested
	40^o	Y (WC)	EOC-10		Tested
Zion Unit 2	4^o	S (W)			2,5
	4^o	V (W)			2,5
	4^o	W (W)			2,5
	4^o	Z (WC)	EOC-11		3
	40^o	Z (WC)		EOC-11	2,5
	40^o	T (W)	EOC-4		Tested
	40^o	U (W)	EOC-1		Tested
	40^o	X (W)			2,5
	40^o	Y (WC)	EOC-10		Tested

W - Capsule contains weld metal specimens

WC - Capsule contains weld metal WOL specimens

* - Capsule locations are relative symmetrical positions and not absolute, e.g. 0^o is equivalent to 90^o, 180^o, or 270^o.

** - Only the dosimetry was evaluated.

*** - HUPCAP

1 - Capsule to be removed, specimens will be tested, dosimetry evaluated, and thermal monitors evaluated.

Table 3-21. Capsule Insertion and Withdrawal Schedule for the Westinghouse Plant-Specific RVSPs (Cont'd)

2 - Capsule to be removed and placed in storage. Dosimetry may be evaluated at this time.

3 - Capsule recommended to be reinserted in higher lead factor location.

4 - Capsule to be maintained in location to EOL.

5 - Standby capsule to be removed at 1-2 times the vessel EOL fluence.

6 - Current cycle for transfer and reinsertion is estimated.

4. BASIS FOR INTEGRATED PROGRAM CONCEPT

The Master Integrated Reactor Vessel Surveillance Program (MIRVP) is an extension of the B&W Owners Group IRVSP to encompass all operating PWR reactor vessels in the USA containing Linde 80 weld seams. The MIRVP represents one phase of a multiphase program of the B&W Owners Group Reactor Vessel Integrity (RVI) Program.[98] The principal objective of the RVI Program is to assure the continued licensability of all eleven participants and their seventeen reactor vessels. The MIRVP provides the data required of Appendix G to 10CFR50 to accomplish this objective. Details of the MIRVP are provided in previous sections to this report and the appendices. This section describes the manner in which relevant technical and regulatory issues are being or will be addressed by the B&W Owners Group MIRVP and RVI Program.

As previously detailed, the power reactor portion of the MIRVP combines the existing RVSPs already in place and additional capsules that have been added, through shared resources, to expand the data base for Linde 80 weld material. The term "integrated" in this case refers primarily to the concept of unified data sharing among all participants. The RVSPs in place prior to instituting this concept remain virtually intact as originally provided, to be in compliance with the appropriate regulations. The MIRVP provides considerably more relevant data to each participant on the Linde 80 class of weld metal. It also provides the opportunity to concentrate the testing efforts on the key RVSP specimens and delaying the testing of a few low priority capsules as defined in Section 3.

The power reactor portion of the MIRVP comprises two principal parts. The first is the continuation of the plant-specific surveillance programs that monitor the irradiation damage to selected materials, as appropriate. The capsules contain samples of weld metal, plate or forging material, and heat-affected zone (HAZ) material from the vessel beltline. This part of the program will continue to monitor the long-term effects of neutron irradiation on the reactor materials and will contribute to the plant-specific materials analysis. The second part of the program consists of a series of supplementary capsules to study the effects of irradiation on a number of Linde 80 reactor vessel weld metals. These capsules contain specimens primarily for obtaining fracture toughness properties of individual weld metals, and are located in the same irradiation holder tubes as the regular plant-specific surveillance capsules.

For an integrated RVSP to be acceptable to the NRC, a number of criteria, as provided by 10CFR50, Appendix H, must be met. Paragraph II.C of Appendix H states the following:

A. An integrated surveillance program may be considered for a set of reactors that have similar design and operating features.

B. The representative materials chosen for surveillance from each reactor in the set may be irradiated in one or more of the reactors, but there must be an adequate dosimetry program for each reactor.

C. No reduction in the requirements for number of materials to be irradiated, specimen types, or number of specimens per reactor is permitted, but the amount of testing may be reduced if the initial results agree with predictions.

D. Integrated surveillance programs must be approved by the Director, Office of Nuclear Reactor Regulation, on a case-by-case basis. Criteria for approval include the following considerations.

E. The design and operating features of the reactors in the set must be sufficiently similar to permit accurate comparison of the predicted amount of radiation damage as a function of total power output.

F. There must be adequate arrangement for data sharing between plants.

G. There must be a contingency plan to assure that the surveillance program for each reactor will not be jeopardized by operation at reduced power level or by an extended outage of another reactor from which data are expected.

H. There must be substantial advantages to be gained, such as reduced power outages or reduced exposure to radiation, as a direct result of not requiring surveillance capsules in all reactors in the set.

There are no exceptions taken to the above criteria and considerations, all having been satisfied by this B&W Owners Group MIRVP. Each of the above criteria and technical considerations are discussed below.

A. Seventeen reactor vessels are included in this integrated program. Eight are B&W 177-FA design having very similar operating and design features and nine are of Westinghouse design. All 17 reactors are of the same basic design concept: pressurized water reactor, operating at 550F and 2250 psi nominal inlet temperature and pressure, and with low enrichment fuel (approximately 2-4% enrichment).

Operating and design features of demonstrated relevancy to neutron radiation damage to reactor vessel materials include the following:

1. neutron energy spectrum

2. irradiation temperature

3. fluence rate

4. gamma heating

<u>The relative neutron energy spectrum</u> is a function of the geometry and materials of the reactor internals components. As shown in Tables 3-2 and 3-6, the materials of the reactors are the same, but the dimensions of the internals vary and will produce some variation in neutron spectra. Differences in neutron spectra, however, are not unique to integrated programs. A surveillance program for a single reactor must contend with the variation in spectrum through the reactor vessel wall and the difference in spectrum between the vessel and the surveillance capsule. In the integrated program, the difference in spectra between the Westinghouse and B&W reactors is no larger than that already encountered in non-integrated programs. The difference in spectra could be ignored if a perfect damage exposure index existed. Since this is not the case, it is desirable to hold the difference in spectra to a minimum and correlate damage in a specimen with different points in the reactor vessel wall using the best available damage function.

The effect of spectra variations will continue to be evaluated in the integrated program just as it would also continue to be evaluated in a non-integrated program.

As an example of the evaluation to be considered, the neutron energy spectrum at the reactor vessel for a typical Westinghouse system design is compared with a typical B&W system design and is shown in Figure 4-1.

Irradiation temperature is controlled by the reactor vessel inlet temperature. Referring to Figure 3-17, it is seen that the cold leg (inlet temperature) for the B&W and Westinghouse reactors are within approximately 10F of each other during full power operation and approximately 25F when operating at partial powers of 70% and above; ± 25F is stated in Regulatory Guide 1.99, Revision 2,[99] Section B, Paragraph 4, as an acceptable range for application of the Regulatory Guide.

Time of operation at partial powers less than 70% averaged over time would not be expected to have a significant effect. It is recognized that these differences in temperature exist among the various reactors in the MIRVP and, therefore, surveillance data will periodically be evaluated for the possible influence of operating temperature on irradiation damage. In addition, other programs will be monitored for potential data related to this matter.

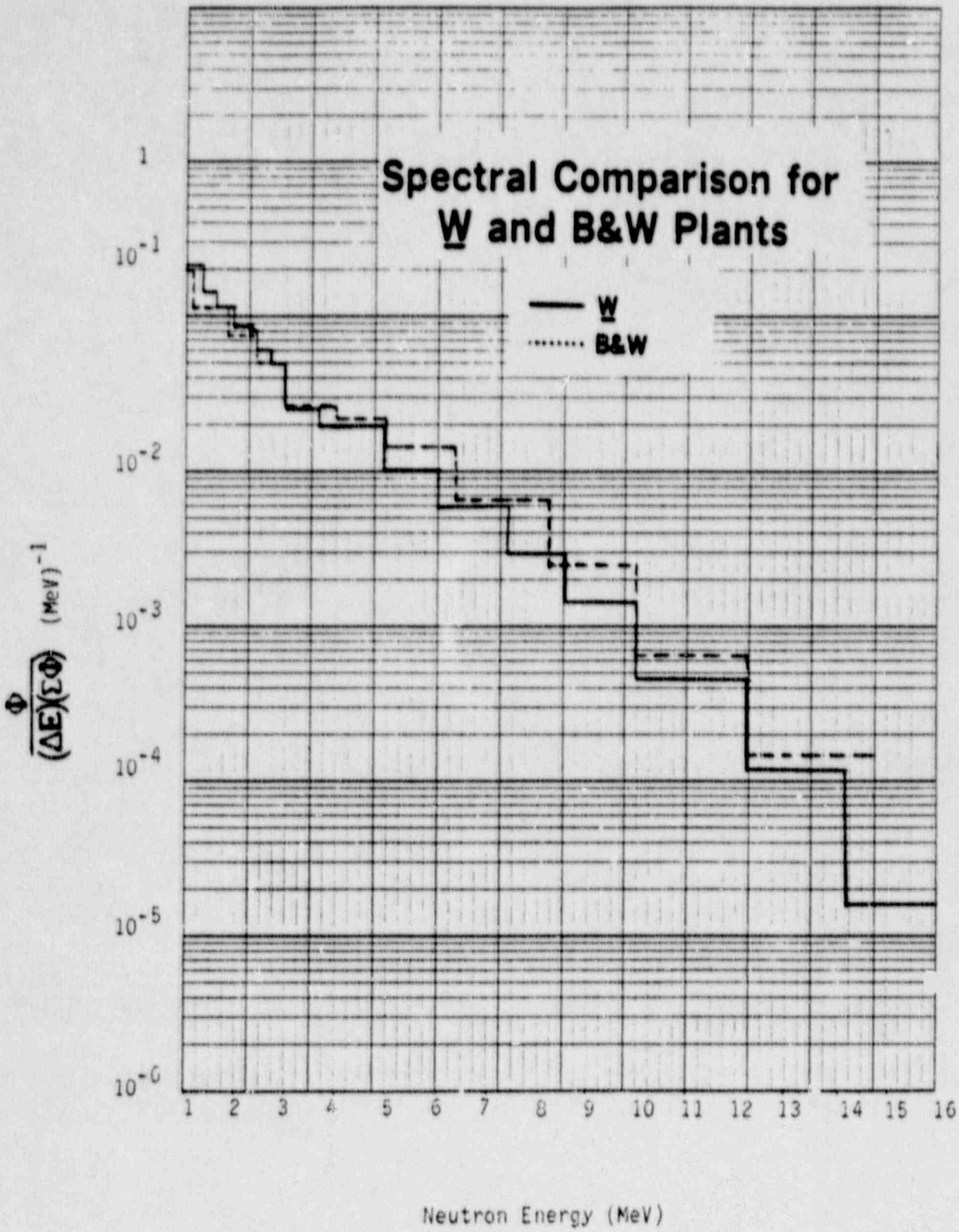
Fluence rate (neutron flux) for all these reactor designs is less than one order of magnitude (E10).

Gamma heating affects the 1/4T environment of all these reactor vessels similarly, since gamma heating is a function of the neutron energy spectrum, which, as discussed above, is similar for these reactors.

Furthermore, ongoing studies of the effects of neutron irradiation on the properties of these vessel materials will be evaluated to develop interrelationships of these variables. In particular, as part of this program, capsule W1 is designed specifically to benchmark data from the two reactor designs.

B. The B&W Owners Group is engaged in a program to provide the 177-FA reactor vessels with cavity dosimeters. A development and benchmarking effort is in progress to be followed by the necessary modifications for the acceptance of removable dosimeters. This is particularly important for "guest" 177-FA reactor vessels. The dosimeters available in the RVSP capsules at Crystal River-3 and Davis-Besse (host reactors) and the Westinghouse-design reactors will continue to provide the required

Figure 4-1 Spectral Comparison for Westinghouse and B&W Plants



information. In addition, several utilities with Westinghouse-design reactors have completed or are conducting cavity dosimetry programs.

C. All of the irradiation capsules originally prepared for all of the reactors in the program are scheduled to be irradiated. However, as stated in this report, it is planned that some of the capsules will not be tested, it being our judgement that they will not provide enough relevant information to be worth the effort. It should be noted, though, that 6 additional capsules are being irradiated (SUPCAPS) and 8 additional capsules are being fabricated (HUPCAPS). These capsules contain test material of the greatest pertinence, improving the quality of the program beyond that of the original capsules.

D. This report is submitted to the NRC to obtain the necessary integrated program approval.

E. The similarity of the 17 reactors with regard to design and operating features as they affect data utilization with regard to neutron radiation damage is discussed in (A) above. Recognizing that at least small differences exist, a benchmarking concept is included in the MIRVP plan to address concerns related to plant-to-plant differences. Archive reactor vessel weld that have previously been characterized in the B&W IRVSP will be irradiated in a Westinghouse plant (HUPCAP-W1). Also a study of the correlation monitor material from each of the plants will be performed. Some environmental differences that may exist between each of the reactor vessels are temperature, capsule design and azimuthal location, fuel assembly design, plant power level, and radial distance to the location of capsules or reactor vessel internals. Each of these differences will be addressed in the final analyses of the MIRVP data.

F. Through the B&W Owners Group Materials Committee, Reactor Vessel Working Group, to which all the Owners of B&W-fabricated operating PWR reactor vessels subscribe, all RVSP reports and relevant information are distributed to all plant owners. The Owners representatives meet regularly to discuss this program and monitor progress of developmental and benchmarking efforts.

G. Operation of a B&W "host" reactor at reduced power level or experiencing an extended outage is not expected to jeopardize the program since the lead factors are sufficient to provide enough time for the capsule to recover their lead. If an outage is extended beyond the margin provided by the lead factor, it is reasonable to shuffle capsules between reactors to maintain the program. B&W has demonstrated the feasibility of shuffling irradiated capsules. The Westinghouse plant-specific capsules will remain in their original irradiation sites.

H. The B&W 177-FA integrated program was instituted as a result of problems with capsule holder tubes because of flow induced vibration. The capsule holder tubes were removed from all the plants and redesigned tubes were installed in those reactors that had not as yet achieved criticality. To have installed new holder tubes in the then operating reactors would have subjected personnel to substantial radiation exposure. The integrated program was extended to include additional capsules to irradiate weld materials that are actually in the beltlines of B&W-fabricated vessels and to provide irradiated specimens for fracture toughness testing. Owners of Westinghouse-designed, B&W-fabricated reactor vessels recently joined the program to benefit from this information. Additional irradiation capsules are being fabricated to provide extended fluence data, which may also be useful for life extension.

5. CERTIFICATION

This report is an accurate description of the master integrated reactor vessel surveillance program prepared in accordance with the requirements of 10CFR50, Appendixes G and H, and revised to include PWR vessels not of B&W design.

S. Fyfitch 10/11/89
S. Fyfitch, Engineer IV Date
Materials and Structural Analysis Unit

This report has been reviewed and is an accurate description of the revised master integrated reactor vessel surveillance program.

K. E. Moore 10/13/89
K. E. Moore, Lead Engineer Date
Reactor Vessel Integrity Program

Verification of independent review.

A. D. McKim 10/13/89
A. D. McKim, Manager Date
Materials and Structural Analysis Unit

This report has been approved for release.

D. L. Howell 10/13/89
D. L. Howell Date
Program Manager

APPENDIX A

Description and Properties of
MIRVP Materials

This appendix compiles the plant-specific reactor vessel surveillance program (RVSP) materials data for each of the seventeen reactor vessels included in the B&W Owners Group MIRVP. The sources of these data are indicated in each table. However, the weld metal chemical compositions are of particular importance and their basis is explained here.

Basis for Weld Metal Chemical Composition

The sources of the listed chemical compositions include the B&W Owners Group weld metal chemical composition characterization (BAW-1500)(96) and the data provided in plant-specific RVSPs where chemical analyses were performed on actual RVSP specimens. All of these data have been evaluated and will be reported in a revision to BAW-1803, "Correlation for Predicting the Effects of Neutron Radiation on Linde 80 Submerged Arc Welds."(100) The most representative chemical composition was established for the particular weld metal subjected to irradiation and testing in each RVSP. This approach is essential in the ongoing testing and evaluations being performed on the Linde 80 class of materials. The general approach used in establishing each chemical composition is:

Category	Basis
Multiple analyses performed on RVSP test blocks.	Mean value from results of multiple analyses.
Analysis of RVSP test specimens. Multiple analyses performed on welds made with the same filler wire heat; not the specific weld in the RVSP.	Preference given to actual analysis performed on RVSP specimens. The copper (Cu) and nickel (Ni) values were also compared to the results obtained from multiple analyses performed on welds made with the same filler wire heat. The RVSP specimen analysis was considered credible if the Cu and Ni values were within the total population. This was generally the case. The exceptions involved Cu concentrations which appeared to be too low and were adjusted accordingly.

Table A-1. Oconee Unit 1 Description and Properties of Reactor Vessel Surveillance Program Materials[a]

Material ID	C	Mn	P	S	Si	Ni	Cr	Mo	Cu	T_{NDT},[e] F	RT_{NDT},[e] F	C_vUSE, ft-lb	UTS,[e] ksi	YS,[e] ksi
	Chemical Composition, %									Impact Properties				
Base metal A	0.21	1.42	0.015	0.015	0.23	0.50	0.17	0.49	0.10	0 0	20 20	141[f] 108[f]	86.1 86.5	64.3 L 65.1 T
Base metal B	0.20	1.40	0.012	0.017	0.20	0.63	0.13	0.50	0.11	20 20	20 20	118[e] 115[e]	89.6 89.5	68.7 L 68.3 T
Weld metal	0.08[b]	1.47[b]	0.016[b]	0.015[b]	0.54[b]	0.59[b]	0.07[b]	0.40[b]	0.32[b]	-50[f]	-10[f]	64[b]	80.5	63.3

Material ID	Heat No.	Spec No.	Supplier	Heat Treatment[c]: Austenitizing	Tempering	Stress Relief
Base metal A	C3265-1	SA 302 Gr B	Lukens	1600-1650F for 9.75 h, brine quench	1200-1220F for 9.5 h, brine quench	1100-1150F for 31 or 40 h, furnace-cooled to 600F
Base metal B	C2800-2	SA 302 Gr B	Lukens	1600-1650F for 9.5 h, brine quench	1200-1225F for 9.5 h, brine quench	1100-1150F for 31 or 40 h, furnace-cooled to 600F
Weld metal	WF-112	N/A[d]	N/A	N/A	N/A	1100-1150F for 31 or 40 h, furnace-cooled

(a) BAW-1820.[22]

(b) BAW-1543, Revision 1.[100]

(c) Stress relief conditions for base metals are those taken from mill certifications. The weld metal stress relief conditions are those actually given to the surveillance weld metal.

(d) N/A - Not Applicable.

(e) BAW-1421, Revision 1.[26]

(f) BAW-2050.[24]

Table A-2. Oconee Unit 2 Description and Properties of Reactor Vessel Surveillance Program Materials[a]

Material ID	Chemical Composition, %									Impact Properties				
	C	Mn	P	S	Si	Ni	Cr	Mn	Cu	T_{NDT}, F[e]	RT_{NDT}, F[e]	C_vUSE, ft-lb	UTS, ksi[e]	YS, ksi[e]
Base metal A	0.24	0.63	0.006	0.012	0.25	0.75	0.36	0.52	0.04	20 20	20 20	152[f] 133[f]	89.2 89.6	68.0 L 68.7 T
Base metal B	0.21	0.62	0.010	0.010	0.23	0.30	0.39	0.58	0.02	-10 -10	-10 -10	160[e] 138[e]	89.9 87.8	69.5 L 67.1 T
Weld metal	0.11[b]	1.55[b]	0.022[b]	0.010[b]	0.65[b]	0.58[b]	0.09[b]	0.39[b]	0.36[b]	-20[f]	4[f]	67[b]	95.2	81.4

Material ID	Heat No.	Spec No.	Supplier	Heat Treatment[c]		
				Austenitizing	Tempering	Stress Relief
Base metal A	3P-2359; AAW 163	SA 508 Cl.2	Ladish	1640F ± 20F held at color 4 h, cold water quenched at 1590F ± 20F held at color 4h, cold water	1260F ± 20F held at color 10 h, cold water quenched	1125F ± 25F held at color 27 or 33 h, furnace-cooled to below 600F
Base metal B	4P-1885; AWG 164	SA 508 Cl.2	Ladish	Same as above	Same as above	Same as above
Weld metal	WF-209-1	N/A[d]	N/A	N/A	N/A	1100-1150F for 27 or 33 h, furnace-cooled

[a] BAW-1820.[22]

[b] BAW-1803, Revision 1.[100]

[c] Stress relief conditions for base metals are those taken from mill certifications. The weld metal stress relief conditions are those actually given to the surveillance weld metal.

[d] N/A - Not Applicable.

[e] BAW-1437.[28]

[f] BAW-2051.[29]

Table A-3. Oconee Unit 3 Description and Properties of Reactor Vessel Surveillance Program Materials[a]

Material ID	Chemical Composition, % C	Mn	P	S	Si	Ni	Cr	Mo	Cu	Impact Properties T_{NDT},[e] F	RT_{NDT},[e] F	C_vUSE, ft-lb	UTS,[e] ksi	YS,[e] ksi
Base metal A	0.24	0.72	0.014	0.012	0.21	0.76	0.34	0.62	0.02	20	20	180[e]	84.0	59.1 L
										20	20	148[f]	85.4	63.1 T
Base metal B	0.21	0.58	0.011	0.015	0.24	0.73	0.30	0.60	0.01	20	20	160[e]	84.6	59.6 L
										20	20	112[f]	83.1	58.2 T
Weld metal	0.08[b]	1.63[b]	0.017[b]	0.012[b]	0.61[b]	0.58[b]	0.10[b]	0.39[b]	0.30[b]	-20	25	66[b]	90.5	75.0

Material ID	Heat No.	Spec No.	Supplier	Heat Treatment[c] Austenitizing	Tempering	Stress Relief
Base metalA	522194; ANK191	SA 508 Cl.2	Ladish	1640F ± 20F held at color 4 h, cold water quenched 1590F ± 20F held at color for 4 h, cold water quenched	1250F ± 20F held at color for 10 h, cold water quenched	1125F ± 25F held at color 30 h, furnace-cooled to below 600F
Base metal B	522314; AWS 192	SA 508 Cl.2	Ladish	Same as above	Same as above	Same as above
Weld metal	WF-209-1	N/A[d]	N/A	N/A	N/A	1100-1150F for 30 h, furnace-cooled

[a] BAW-1820.[22]

[b] BAW-1803, Revision 1.[100]

[c] Stress relief conditions for base metals are those taken from mill certifications. The weld metal stress relief conditions are those actually given to the surveillance weld metal.

[d] N/A - Not Applicable.

[e] BAW-1438.[30]

[f] BAW-1697.[31]

Table A-4. Three Mile Island Unit 1 Description and Properties of Reactor Vessel Surveillance Program Materials[a]

Material ID	Chemical Composition, %									Impact Properties				
	C	Mn	P	S	Si	Ni	Cr	Mo	Cu	T_{NDT},[e] F	RT_{NDT},[e] F	C_vUSE, ft-lb	UTS,[e] ksi	YS,[e] ksi
Base metal A	0.24	1.36	0.010	0.017	0.23	0.57	0.19	0.51	0.09	10	10	131[f]	94.0	71.1 L
										10	30	98[f]	92.2	68.4 T
Base metal B	0.21	1.24	0.010	0.016	0.27	0.55	0.12	0.47	0.12	-10	-10	182[e]	83.7	60.4 L
										-10	20	112[e]	82.4	59.4 T
Weld metal	0.09[b]	1.62[b]	0.013[b]	0.015[b]	0.46[b]	0.66[b]	0.10[b]	0.40[b]	0.33[b]	-20	-20[f]	81[b]	86.2	69.2

Material ID	Heat No.	Spec No.	Supplier	Heat Treatment[c]: Austenitizing	Tempering	Stress Relief
Base metal A	C2789-2	SA 302 Gr B	Lukens	1600-1650F for 9.5 h, brine quench 1200-1225F for 9.5 h, brine quench 1600-1650F for 9.5 h, brine quench 1600-1650F for 9.5 h, brine quench 1510-1535F for 5 h, brine quench 1200-1225F for 5 h, brine quench		1100-1150F for 22.5 or 27.5 h, furnace-cooled
Base metal B	C3307-1	SA 302 Gr B	Lukens	1600-1650F for 9.5 h, brine quench 1200-1225F for 9.5 h, brine quench 1225-1250F for 9.5 h, brine quench		Same as above
Weld metal	WF-25	N/A[d]	N/A	N/A	N/A	1100-1150F for 22.5 or 27.5 h, furnace-cooled

[a] BAW-1820.[22]

[b] BAW-1803, Revision 1.[100]

[c] Stress relief conditions for base metals are those taken from mill certifications. The weld metal stress relief conditions are those actually given to the surveillance weld metal.

[d] N/A - Not Applicable.

[e] BAW-1439.[33]

[f] BAW-1901.[32]

Table A-5. Crystal River Unit 3 Description and Properties of Reactor Vessel Surveillance Program Materials[(a)]

Material ID	Chemical Composition, %									Impact Properties				
	C	Mn	P	S	Si	Ni	Cr	Mo	Cu	T_{NDT},[(f)] F	RT_{NDT},[(f)] F	C_vUSE, ft-lb	UTS,[(f)] ksi	YS,[(f)] ksi
Base metal A	0.23	1.30	0.008	0.016	0.22	0.54	0.11	0.55	0.20	-10 -10	-10 1	124[(g)] 94[(h)]	91.9 92.3	69.8 L 69.4 T
Base metal B	0.23	1.30	0.008	0.016	0.22	0.54	0.11	0.55	0.20	---	---	---	----	----
Weld metal A	0.08[(c)]	1.65[(c)]	0.021[(c)]	0.013[(c)]	1.00[(c)]	0.10[(c)]	0.07[(c)]	0.45[(c)]	0.41[(c)]	---[(b)]	50[(b)]	79[(c)]	93.9[(b)]	77.1[(b)]
Weld metal B	0.10[(i)]	1.57[(i)]	0.018[(i)]	0.009[(i)]	0.54[(i)]	0.60[(i)]	0.094[(i)]	0.43[(i)]	0.35[(i)]	---	---	---	----	----

Material ID	Heat No.	Spec No.	Supplier	Heat Treatment[(d)]		
				Austenitizing	Tempering	Stress Relief
Base metal A	C4344-1 Cl.1	SA 533 Gr B	Lukens	1650-1700F, held 1 h/in. water quenched to 400F	1180F, held 0.5 h/in. air cooled	1100-1150F held 27 h, furnace cooled to below 600F
Base metal B	C4344-2 Cl.1	SA 533 Gr B	Lukens	Same as above	1100F, held 0.5 h/in. air cooled	Same as above
Weld metal A	Atypical	N/A[(e)]	N/A	N/A	N/A	1100-1150F for 27 h, furnace-cooled
Weld metal B	WF-209-1	N/A	N/A	N/A	N/A	1100-1150F for 27 h, furnace-cooled

(a) BAW-1820.[(22)]

(b) BAW-10144[(101)]

(c) BAW-1803, Revision 1.[(100)]

(d) Stress relief conditions for base metals are those taken from mill certifications. The weld metal stress relief conditions are those actually given to the surveillance weld metal.

(e) N/A - Not Applicable.

(f) BAW-1679, Revision 1.[(34)]

(g) BAW-1898.[(36)]

(h) BAW-2049.[(39)]

(i) BAW-1500.[(96)]

Table A-6. Arkansas Nuclear One Unit 1 Description and Properties of Reactor Vessel Surveillance Program Materials[a]

Material ID	Chemical Composition, % C	Mn	P	S	Si	Ni	Cr	Mo	Cu	Impact Properties T_{NDT},[e] F	RT_{NDT},[e] F	C_vUSE, ft-lb	UTS,[e] ksi	YS,[e] ksi
Base metal A	0.21	1.32	0.010	0.016	0.20	0.52	0.19	0.57	0.15	10 10	10 30	132[f] 96[f]	94.9 94.6	72.0 L 71.8 T
Base metal B	0.21	1.32	0.010	0.016	0.20	0.52	0.19	0.57	0.15	-20 -20	-20 10	147[g] 107[g]	90.1 90.3	67.7 L 67.8 T
Weld metal	0.09[b]	1.49[b]	0.016[b]	0.016[b]	0.51[b]	0.59[b]	0.06[b]	0.39[b]	0.28[b]	-20	30	73[b]	84.6	67.6

Material ID	Heat No.	Spec No.	Supplier	Heat Treatment[c] Austenitizing	Tempering	Stress Relief
Base metal A	C5114-1 Cl.1	SA 533 Gr B	Lukens	1650-1700F, held 1 h/in. and water quenched to 400F	1200F, held 1 h/in. and air cooled	1100-1150F, held 29 h, and furnace-cooled within a rate of 35F/h to below 600F
Base metal B	C5114-2 Cl.1	SA 533 Gr B	Lukens	Same as above	1200F, held 0.5 h/in. and air cooled	Same as above
Weld metal	WF-193	N/A[d]	N/A	N/A	N/A	1100-1150F for 29 h, furnace-cooled

(a) BAW-1820.[22]
(b) BAW-1803, Revision 1.[100]
(c) Stress relief conditions for base metals are those taken from mill certifications. The weld metal stress relief conditions are those actually given to the surveillance weld metal.
(d) N/A - Not Applicable.
(e) BAW-1440.[43]
(f) BAW-2075.[42]
(g) BAW-1698.[41]

Table A-7. Rancho Seco Unit 1 Description and Properties of Reactor Vessel Surveillance Program Materials[a]

Material ID	Chemical Composition, %									Impact Properties				
	C	Mn	P	S	Si	Ni	Cr	Mo	Cu	T_{NDT}, F[e]	RT_{NDT}, F[e]	C_vUSE, ft-lb	UTS,[e] ksi	YS,[e] ksi
Base metal A	0.20	1.33	0.010	0.015	0.19	0.58	0.20	0.52	0.10	---	---	---	----	----
Base metal B	0.20	1.26	0.013	0.017	0.15	0.60	0.14	0.55	0.12	-10	-10	90[f]	83.8	63.9
Weld metal[b]	0.09	1.49	0.016	0.016	0.51	0.59	0.06	0.39	0.28	-90	-14	68[b]	83.5	67.5

Material ID	Heat No.	Spec No.	Supplier	Heat Treatment[c]		
				Austenitizing	Tempering	Stress Relief
Base metal A	C5070-1	SA 533 Gr B Cl.1	Lukens	1650-1700F, held 1 h/in. and water quenched to 400F	1200F, held 0.5 h/in. and air cooled	28 h at 1100-1150F and furnace-cooled below 600F
Base metal B	C5062-1	SA 533 Gr B Cl.1	Lukens	Same as above	Same as above	Same as above
Weld metal	WF-193	N/A[d]	N/A	N/A	N/A	1100-1150F for 28 h, furnace-cooled

(a) BAW-1820.(22)

(b) BAW-1803, Revision 1.(100)

(c) Stress relief conditions for base metals are those taken from mill certifications. The weld metal stress relief conditions are those actually given to the surveillance weld metal.

(d) N/A - Not Applicable.

(e) BAW-1702.(44)

(f) BAW-2074.(48)

Table A-8. Davis-Besse Unit 1 Description and Properties of Reactor Vessel Surveillance Program Materials[a]

Material ID	Chemical Composition, % C	Mn	P	S	Si	Ni	Cr	Mo	Cu	Impact Properties T_{NDT},[e] F	RT_{NDT},[e] F	C_vUSE, ft-lb	UTS,[e] ksi	YS,[e] ksi
Base metal A	0.22	0.63	0.011	0.011	0.27	0.81	0.32	0.63	0.02	50	50	127[f]	90.7	72.3
Base metal B	0.26	0.68	0.004	0.006	0.30	0.77	0.38	0.64	0.04	20[f]	20[f]	140[f]	----	----
Weld metal[b]	0.09	1.69	0.014	0.013	0.41	0.63	0.15	0.49	0.21	-20	5	72[f]	85.6	70.2

Material ID	Heat No.	Spec No.	Supplier	Heat Treatment[c] Austenitizing	Tempering	Stress Relief
Base metal A	5P4086; BCC241	SA 508 Cl.2	Ladish	1640F ± 10F held at color 4 h, cold water quenched Reaustenitized 1590F ± 10F held at color 4 h, cold water quenched	1240F ± 10F held at color 5 h, air cooled	1125F ± 25F held at color 15.5 h, furnace-cooled below 600F
Base metal B	123X244; AKJ233	SA 508 Cl.2	Ladish	Same as above	1240F ± 10F held at color 6 h, air cooled	Same as above
Weld metal	WF-182-1	N/A[d]	N/A	N/A	N/A	1100-1150F for 15.5 h, furnace-cooled

(a) BAW-1820.[22]

(b) BAW-1803, Revision 1.[100]

(c) Stress relief conditions for base metals are those taken from mill certifications. The weld metal stress relief conditions are those actually given to the surveillance weld metal.

(d) N/A - Not Applicable.

(e) BAW-1701.[52]

(f) BAW-1882.[49]

Table A-9. R.E. Ginna Unit 1 Description and Properties of Reactor Vessel Surveillance Program Materials

	Chemical Composition, %									Impact Properties				
Material ID	C	Mn	P	S	Si	Ni	Cr	Mo	Cu	T_{NDT}, F	RT_{NDT}, F	C_vUSE, ft-lb	UTS, ksi	YS, ksi
Base metal A[a]	0.19	0.67	0.010	0.011	0.20	0.69	0.37	0.57	0.05	---	--	183	83.6	62.7
Base metal B[a]	0.18	0.66	0.010	0.007	0.23	0.69	0.33	0.58	0.07	---	--	140	97.2	78.2
Weld metal[b]	0.08	1.41	0.012	0.016	0.59	0.56	0.09	0.36	0.23	---	--	80	87.4	73.5

				Heat Treatment[c]		
Material ID	Heat No.	Spec No.	Supplier	Austenitizing	Tempering	Stress Relief
Base metal A	125P666	SA508, Cl.2	Beth.	1550F for 9 h, water quenched	1220F for 12 h, air cooled	1100F for 11-1/2 h, furnace-cooled
Base metal B	125S255	SA508, Cl.2	Beth.	1550F for 15-1/2 h, water quenched	1220F for 18 h, air cooled	Same as above
Weld metal	SA-1036	N/A[d]	N/A	N/A	N/A	Same as above

(a) WCAP-7254[54] and WCAP-11026.[102]

(b) BAW-1803, Revision 1.[100]

(c) WCAP-10086.[64]

(d) N/A - Not Applicable.

Table A-10. Point Beach Unit 1 Description and Properties of Reactor Vessel Surveillance Program Materials

Material ID	Chemical Composition, %									Impact Properties				
	C	Mn	P	S	Si	Ni	Cr	Mo	Cu	T_{NDT}, F	RT_{NDT}, F	C_vUSE, ft-lb	UTS, ksi	YS, ksi
Base metal A[a]	0.19	1.42	0.010	0.020	0.25	0.056[b]	----	0.48	0.20[b]	-30[c]	- 2[d]	107	85.1	65.2
Base metal B[a]	0.21	1.37	0.014	0.019	0.25	0.065[b]	----	0.46	0.12[b]	-20[c]	-20[d]	119	76.2	53.6
Weld metal[e]	0.09	1.47	0.019	0.024	0.49	0.57	0.13	0.39	0.22	0[d]	0[d]	65	86.3	69.9

Material ID	Heat No.	Spec No.	Supplier	Heat Treatment[a]		
				Austenitizing	Tempering	Stress Relief
Base metal A	A9811	SA302 Gr.B	Lukens	1650F for 7 h, water quenched	1225F, 7 h, air cooled	1125F for 11-1/4 h, furnace-cooled
Base metal B	C1423	SA302 Gr.B	Lukens	Same as above	Same as above	1125F for 10-1/2 h, furnace-cooled
Weld metal	SA-1263	N/A[f]	N/A	N/A	N/A	1125F for 11-1/4 h, furnace-cooled

(a) WCAP-7513[55] and WCAP-8739.[67]

(b) Docket No. 50-266, response to 10CFR50.61, January 20, 1986.[103]

(c) WCAP-8743.[104]

(d) Estimated by the methods of the U.S. NRC Standard Review Plan, Section 5.3.2, Pressure Temperature Limits.

(e) BAW-1803, Revision 1.[100]

(f) N/A - Not Applicable.

Table A-11. Point Beach Unit 2 Description and Properties of Reactor Vessel Surveillance Program Materials

Material ID	Chemical Composition, %: C	Mn	P	S	Si	Ni	Cr	Mo	Cu	Impact Properties: T_{NDT}, F	RT_{NDT}, F	C_vUSE, ft-lb	UTS, ksi	YS, ksi
Base metal A[a]	0.20	0.65	0.009	0.009	0.24	0.71	0.35	0.59	0.088	40[c]	40[c]	180	80.2	55.4
Base metal B[a]	0.22	0.59	0.010	0.008	0.23	0.70	0.33	0.60	0.051	40[c]	40[c]	145	92.0	70.8
Weld metal[b]	0.08	1.40	0.014	0.013	0.55	0.59	0.07	0.39	0.25	25[d]	27[d]	66	87.0	71.9

Material ID	Heat No.	Spec No.	Supplier	Heat Treatment[a]: Austenitizing	Tempering	Stress Relief
Base metal A	123V500 VA1	A508 Cl.2	Beth.	1550F for 9-1/2 h, water quenched	1200F for 12 h, air cooled	1125F for 12 h, furnace-cooled
Base metal B	122W195 VA1	A508 Cl.2	Beth.	1550F for 8 h, water quenched	Same as above	Same as above
Weld metal	WF-193	N/A[e]	N/A	N/A	N/A	1125F for 11-1/2 h, furnace-cooled

[a] WCAP-7712[56] and WCAP-9331.[71]

[b] BAW-1803, Revision 1.[100]

[c] WCAP-8738.[105]

[d] Estimated by the methods of the U.S. NRC Standard Review Plan, Section 5.3.2, Pressure Temperature Limits.

[e] N/A - Not Applicable.

Table A-12. Surry Unit 1 Description and Properties of Reactor Vessel Surveillance Program Materials

Material ID	C	Mn	P	S	Si	Ni	Cr	Mo	Cu	T_{NDT}, F	RT_{NDT}, F	C_vUSE, ft-lb	UTS, ksi	YS, ksi
	Chemical Composition, %									Impact Properties				
Base metal A[a]	0.23	1.35	0.008	0.015	0.23	0.55	0.069	0.55	0.11	---	--	142	90.5	68.1
Base metal B[a]	0.22	1.33	0.014	0.014	0.23	0.50	0.078	0.55	0.11	---	--	125	93.8	71.8
Weld metal[b]	0.09	1.53	0.013	0.017	0.53	0.68	0.08	0.42	0.35	---	--	70	83.2	69.7

Material ID	Heat No.	Spec No.	Supplier	Heat Treatment[a]: Austenitizing	Heat Treatment[a]: Tempering	Heat Treatment[a]: Stress Relief
Base metal A	C4326-1	SA 533 Gr.B Cl.1	Lukens	1650-1700F, for 9 h, water quenched	1210F for 9 h, air cooled	1125F for 15-1/2 h, furnace-cooled to 600F
Base metal B	C4415-1	SA 533 Gr.B Cl.1	Lukens	Same as above	1200F for 9 h, air cooled	Same as above
Weld metal	SA-1526	N/A[c]	N/A	N/A	N/A	Same as above

(a) WCAP-7723(57) and WCAP-11415.(74)

(b) BAW-1803, Revision 1.(100)

(c) N/A - Not Applicable.

Table A-13. Surry Unit 2 Description and Properties of Reactor Vessel Surveillance Program Materials

Material ID	Chemical Composition, % C	Mn	P	S	Si	Ni	Cr	Mo	Cu	Impact Properties T_{NDT}, F	RT_{NDT}, F	C_vUSE, ft-lb	UTS, ksi	YS, ksi
Base metal A[a]	0.23	1.30	0.012	0.014	0.25	0.54	0.075	0.54	0.11	---	--	125	91.3	68.2
Weld metal[a]	0.09	1.51	0.017	0.016	0.46	0.56	0.10	0.41	0.19	---	--	90	86.5	70.8

Material ID	Heat No.	Spec No.	Supplier	Heat Treatment[a] Austenitizing	Tempering	Stress Relief
Base metal A	C4339-1	SA 533 Gr.B Cl.1	Lukens	1625F for 9 h, brine quenched	1212F for 9 h, brine quenched	1140F, for 15-1/4 h, furnace-cooled
Weld metal A	R3008	N/A[b]	N/A	N/A	N/A	Same as above

(a) WCAP-8085[58] and WCAP-11499.[76]

(b) N/A - Not Applicable.

Table A-14. Turkey Point Unit 3 Description and Properties of Reactor Vessel Surveillance Program Materials

Material ID	Chemical Composition, %									Impact Properties				
	C	Mn	P	S	Si	Ni	Cr	Mo	Cu	T_{NDT}, F	RT_{NDT}, F	C_vUSE, ft-lb	UTS, ksi	YS, ksi
Base metal A[a]	0.20	0.64	0.010	0.010	0.26	0.70	0.395	0.62	0.058	---	40	145	86.2	64.4
Base metal B[a]	0.20	0.615	0.010	0.008	0.155	0.67	0.38	0.585	0.079	---	30	154	82.6	57.4
Weld metal[b]	0.08	1.51	0.020	0.013	0.57	0.60[c]	0.16	0.37	0.26[c]	---	10[c]	65[a]	92.8	76.3

Material ID	Heat No.	Spec No.	Supplier	Heat Treatment[a]		
				Austenitizing	Tempering	Stress Relief
Base metal A	123P461 VA-1	SA508 Cl.2	Beth.	1550F for 13 h, water quenched	1210F for 8 h, air cooled	1125F for 10-1/2 h, furnace-cooled to 600F
Base metal B	123S266 VA-1	SA508 Cl.2	Beth.	Same as above	Same as above	Same as above
Weld metal	SA-1101	N/A[d]	N/A	N/A	N/A	1125F for 10-1/4 h, furnace-cooled to 600F

(a) WCAP-7656[59] and Final Report SWRI Project No. 02-5131.[79]

(b) BAW-1500.[96]

(c) Safety Evaluation Report, Memorandum, S. Varga to J. W. Williams, April 26, 1984[106]

(d) N/A - Not Applicable.

Table A-15. Turkey Point Unit 4 Description and Properties of Reactor Vessel Surveillance Program Materials

Material ID	Chemical Composition, %									Impact Properties				
	C	Mn	P	S	Si	Ni	Cr	Mo	Cu	T_{NDT}, F	RT_{NDT}, F	C_vUSE, ft-lb	UTS, ksi	YS, ksi
Base metal A[a]	0.22	0.67	0.010	0.009	0.20	0.71	0.33	0.56	0.054	---	50	135	90.1	68.6
Base metal B[a]	0.21	0.67	0.011	0.009	0.23	0.70	0.31	0.56	0.056	---	40	132	91.5	70.8
Weld metal[a]	0.10	1.44	0.014	0.011	0.50	0.[illegible]	0.14	0.36	0.26[c]	---	--	66[a]	90.8	70.2

Material ID	Heat No.	Spec No.	Supplier	Heat Treatment[a]		
				Austenitizing	Tempering	Stress Relief
Base metal A	123P481 VA-1	A508 Cl.2	Lukens	1550F for 10-1/2 h, water quenched	1200F for 18 h, air cooled	1125F for 10-1/2 h, furnace-cooled to 600F
Base metal B	122S180 VA-1	A508 Cl.2	Lukens	1550F for 10-1/4 h, water quenched	1210F for 18 h, air cooled	Same as above
Weld metal A	SA-1094	N/A[b]	N/A	N/A	N/A	1125F for 10-1/4 h, furnace-cooled to 600F

(a) WCAP-7660[60] and Final Report SWRI Project No. 02-4221,[83] and Final Report SWRI Project No. 02-5380.[82]

(b) N/A - Not Applicable.

Table A-16. Zion Unit 1 Description and Properties of Reactor Vessel Surveillance Program Materials

Material ID	Chemical Composition, %									Impact Properties				
	C	Mn	P	S	Si	Ni	Cr	Mo	Cu	T_{NDT}, F	RT_{NDT}, F	C_vUSE, ft-lb	UTS, ksi	YS, ksi
Base metal A[a]	0.20	1.30	0.010	0.011	0.20	0.49	---	0.47	0.11	---	--	140	83.8	63.0
Weld metal[b]	0.09	1.51	0.020	0.013	0.68	0.57	0.06	0.39	0.35	---	--	64	89.4	72.7

Material ID	Heat No.	Spec No.	Supplier	Heat Treatment[a]		
				Austenitizing	Tempering	Stress Relief
Base metal A	B7835-1	SA533, Gr.B Cl.1	Lukens	1625F for 9-3/4 h, brine quenched	1212F for 9-3/4 brine quenched	1125F for 25 h, furnace-cooled
Weld metal	WF-209-1	N/A[c]	N/A	N/A	N/A	1125F for 23 hours, furnace-cooled

(a) WCAP-8064[61] and WCAP-9890.[85]

(b) BAW-1803, Revision 1.[100]

(c) N/A - Not Applicable.

Table A-17. Zion Unit 2 Description and Properties of Reactor Vessel Surveillance Program Materials

	Chemical Composition, %									Impact Properties				
Material ID	C	Mn	P	S	Si	Ni	Cr	Mo	Cu	T_{NDT}, F	RT_{NDT}, F	C_vUSE, ft-lb	UTS, ksi	YS, ksi
Base metal A[a]	0.23	1.39	0.010	0.016	0.22	0.53	0.065	0.54	0.12	---	--	128	92.4	69.9
Weld metal[b]	0.08	1.51	0.017	0.013	0.68	0.57	0.06	0.39	0.30	---	--	70	88.8	73.6

				Heat Treatment[a]		
Material ID	Heat No.	Spec No.	Supplier	Austenitizing	Tempering	Stress Relief
Base metal A	C4007-1	SA533 Gr.B Cl.1	Lukens	1600/1650F for 9-3/4 h, brine quenched	1200/1225F for 9-3/4 h, brine quenched	1100/1150F for 30 h, furnace-cooled
Weld metal	WF-209-1	N/A[c]	N/A	N/A	N/A	Same as above

(a)WCAP-8132.(62)

(b)BAW-1803, Revision 1.(100)

(c)N/A - Not Applicable.

APPENDIX B

Description and Properties of the SUPCAP and HUPCAP Materials

Table B-1. Chemical Composition and Unirradiated Mechanical Properties of the SUPCAP and HUPCAP Surveillance Weld Metals

Ident. No.	Chemical Composition, %									Impact Properties				
	C	Mn	P	S	Si	Ni	Cr	Mo	Cu	T_{NDT}, F	RT_{NDT}, F	C_vUSE, ft-lb	UTS, ksi	YS, ksi
WF-70(N)[1]	0.09	1.63	0.018	0.009	0.54	0.59	0.11	0.40	0.42	-50	20	66	85.5	69.0
WF-112	0.08	1.47	0.016	0.015	0.54	0.59	0.07	0.39	0.32	-50	0	65	83.0	66.0
SA-1585	0.08	1.45	0.016	0.016	0.51	0.59	0.09	0.38	0.21	-50	- 8	78	81.0	--
SA-1526	0.09	1.53	0.013	0.017	0.53	0.70	0.08	0.42	0.37	-40	-20	74	88.0	--
WF-25(6)[2]	0.09	1.58	0.015	0.016	0.54	0.67	0.09	0.42	0.35	-10	40	73	80.7	66.5
WF-67	0.08	1.55	0.021	0.016	0.58	0.60	0.10	0.40	0.22	-20	8	70	81.5	64.0
WF-25(9)[3]	0.09	1.55	0.014	0.015	0.55	0.70	0.08	0.41	0.35	-40	10	71	80.7	66.5
SA-1135	0.08	1.45	0.011	0.013	0.49	0.59	0.08	0.38	0.27	--	--	--	81.5	67.0
SA-1101	0.08	1.56	0.019	0.008	0.59	0.54	0.16	0.38	0.18	-70	10	75	89.3	72.8
WF-70(B)[4]	0.09	1.62	0.018	0.011	0.59	0.59	0.10	0.40	0.35	To be determined.				
WF-182-1	0.09	1.69	0.014	0.013	0.41	0.63	0.15	0.40	0.21	-20	5	70	85.6	70.2
SA-1484	0.08	1.56	0.016	0.020	0.47	0.59	0.09	0.39	0.26	--	--	--	--	--
WF-209-1	0.09	1.62	0.018	0.011	0.59	0.59	0.10	0.40	0.35	--	15	64	89.4	72.7
SA-1263	0.09	1.47	0.019	0.024	0.49	0.57	0.13	0.39	0.22	--	--	65	76.2	53.5

1: N = Material from the nozzle drop-out (weld)

2: 6 = Material from the TMI-2 nozzle drop-out (weld)

3: 9 = Material from the OC-3 nozzle drop-out (weld)

4: B = Material from the Midland vessel beltline weld

Table B-2. Description of the SUPCAP and HUPCAP Surveillance Weld Metals

Ident No.	Filler Metal Type	Flux Type	Welding Process	Test Qualification Post-Weld Heat Treatment
WF-70(N)[1]	Mn,Mo,Ni	Linde 80	Sub. arc	48 h at 1100-1150F
WF-112	Mn,Mo,Ni	Linde 80	Sub. arc	48 h at 1100-1150F
SA-1585	Mn,Mo,Ni	Linde 80	Sub. arc	80 h at 1100-1150F
SA-1526	Mn,Mo,Ni	Linde 80	Sub. arc	48 h at 1100-1150F
WF-25(6)	Mn,Mo,Ni	Linde 80	Sub. arc	48 h at 1100-1150F
WF-67	Mn,Mo,Ni	Linde 80	Sub. arc	48 h at 1100-1150F
WF-25(9)[2]	Mn,Mo,Ni	Linde 80	Sub. arc	48 h at 1100-1150F
SA-1135	Mn,Mo,Ni	Linde 80	Sub. arc	Eight 6 h cycles at 1100-1150F
SA-1101	Mn,Mo,Ni	Linde 80	Sub. arc	25 h at 1100-1150F
WF-70(B)[3]	Mn,Mo,Ni	Linde 80	Sub. arc	30 h at 1100-1150F
WF-182-1	Mn,Mo,Ni	Linde 80	Sub. arc	15-1/2 h at 1100-1150F
SA-1484	Mn,Mo,Ni	Linde 80	Sub. arc	48 h at 1100-1150F
WF-209-1	Mn,Mo,Ni	Linde 80	Sub. arc	23 h at 1100-1150F
SA-1263	Mn,Mo,Ni	Linde 80	Sub. arc	11-1/4 h at 1100-1150F

1: N = Material from the nozzle drop-out (weld)

2: 6 = Material from the TMI-2 nozzle drop-out (weld)

3: 9 = Material from the OC-3 nozzle drop-out (weld)

4: B = Material from the Midland vessel beltline weld

APPENDIX C

Description of Surveillance Capsule
Test Specimens -- Plant-Specific, SUPCAP,
and HUPCAP Capsules

This appendix describes the tension test, Charpy V-notch, and compact fracture toughness specimens included in the reactor vessel surveillance capsules utilized in this program.

C.1. Tension Test Specimens

The tension test specimens used in the reactor vessel surveillance capsules conform to the requirements of ASTM E 8-69T.[18] There are three different sizes of tension test specimens among the various capsule designs. In the B&W plant specific and SUPCAP Type R1 designs standard size specimens with a gage length of 1.428 inches are used. The tension test specimens in the SUPCAP Type R2 and HUPCAP designs are smaller and fit in a Charpy specimen envelope. The gage length for the subsize tension test specimen is 0.840 inch. Figures C-1 and C-2 illustrate the standard and subsize size tension test specimens, respectively. The Westinghouse plant-specific capsules contain standard size tension test specimens with a gage length of 1.00 inch and are shown in Figure C-3.

C.2. Charpy V-Notch Specimens

The Charpy V-Notch specimens in the majority of the reactor vessel surveillance capsules conform to the requirements of ASTM E 23-72.[17] Two different sizes of specimens are used by B&W and Westinghouse. Figure C-4 shows the B&W plant-specific and SUPCAP design and Figure C-5 shows the Westinghouse plant-specific design. The HUPCAPs contain Charpy V-notch specimens designed to ASTM E 23-86 which modified the allowable tolerances on the depth of the notch. This design is given in Figure C-6.

C.3. Compact Fracture Toughness Specimens

There are 4 configurations of compact fracture toughness specimens: rectangular, round, slow bend, and wedge-opening-loading (WOL) geometries. The configurations and sizes of these specimens are described in the following sections.

C.3.1. Rectangular Compact Fracture Toughness Specimens

The rectangular compact fracture toughness specimens are modifications of those in ASTM E 399-81[19] and E 813-81.[20] The specimen geometry is illustrated in Figure C-7. As shown in the figure, the specimens were modified for measurement of load versus load line displacement. Five sizes of this type of specimen are included. The specimen sizes (in terms of thickness) are 0.394, 0.417, 0.500, 0.936, and 2.000 inches. The dimensions of these specimens are listed in Table C-1. The 0.500 inch specimen was used in some B&W plant-specific capsules. The 0.394 and 0.500 inch specimens were used in the SUPCAPS. The 0.936 and 2.000 inch specimens were used for unirradiated specimens only. The 0.417 inch specimen will be used in the W1 HUPCAP.

C.3.2. Round Compact Fracture Toughness Specimens

When the SUPCAPs were designed, it was recognized that the round compact fracture specimens would make the most efficient use of the capsule volume. The round compact fracture specimens conform to the requirements for disk-shaped specimens of ASTM E 399-81. Figure C-8 illustrates this specimen, with its corresponding dimensions. These specimens are used in the SUPCAPs and HUPCAPs.

C.3.3. Wedge-Opening-Loading Fracture Toughness Specimens

Fracture toughness specimens of the 1XWOL configuration have been utilized in the Westinghouse plant-specific surveillance capsules for testing in accordance with ASTM E399-70T. Although this configuration was considered a state-of-the-art fracture toughness specimen when these surveillance programs were designed, it is not well suited for more recently developed techniques involving elastic-plastic fracture mechanics. A method for modifying and testing the 1XWOL fracture toughness specimens has been developed to closely conform to the requirements of ASTM E813-87. The specimen geometry, prior to modification, is illustrated in Figure C-9.

C.3.4. Slow-Bend Fracture Toughness Specimens

The HUPCAP capsule of the Westinghouse design, W1, includes precracked Charpy-size slow-bend (three-point) specimens which are described in Reference 107. Figure C-10 illustrates this specimen with its corresponding dimensions.

C.3.5. Side-Grooved Specimens

The 0.936 TRCT specimens for the two SUPCAPs at Crystal River Unit 3 have been side-grooved. The geometry of the side grooves is shown in Figure C-11.

The depth of the grooves is 10% of specimen thickness, with a total reduction of 20%. The angle and radius of the grooves are the same as for the notch of the Charpy specimens.

The decision to side-groove the specimen was made based on the information generated by Shih, et al.[108] In general, the side grooves kept the crack front of the stable crack relatively straight. A large degree of crack tunneling was observed in the testing of non-side-grooved specimens. Shih found that the 25% total side-grooving (12.5% on each side) was sufficient for the tough materials used for his development. Shih tested 12.5, 25, and 50% total side-grooved specimens. The 12.5% side-grooved specimens did not completely suppress the shear tip formation, and the 50% showed higher stable crack growth extension near the tip of the side-grooves than at the center of the specimen. For the materials of this program, the 20% side-grooving was selected because it was believed to be adequate and also minimized the reduction of the net section thickness of the specimen (reducing J measuring capacity). The irradiated welds are not expected to be as ductile as the material used by Shih, et al.,[108] in their studies. Side-grooving is also expected to affect the slope of the J versus Δa R curves because of the straightening of the crack front, which affects the determination of Δa. The J-Δa curves determined with side-grooved specimens are believed to be more representative of the extension of a crack on a thick-walled component. The side-grooves affect neither the determination of J_{IC} nor the slope of the J-Δa curve at very small Δa.

Side-grooving will also be utilized in the modified 1XWOL testing procedure. The depth of the grooves is 10% of specimen thickness, with a total reduction of 20%.

Table C-1. Dimensions of Compact Fracture Toughness Specimens[a]

Specimen ID	Dimensions, in. Load Line to Back Face, W	Thickness, B = W/2	Length, 1.25 W	Width, 1.2 W	Load Line Opening, D	Notch Opening, N
0.394 TCT	0.788	0.394	0.985	0.945	0.100	0.064
0.417 TCT	0.796	0.417	1.042	1.000	0.150	0.064
0.500 TCT	1.000	0.500	1.250	1.200	0.150	0.064
0.936 TCT	1.872	0.936	2.340	2.246	0.150	0.127
2.000 TCT	4.00	2.000	5.000	4.800	0.150	0.127

[a]The round compact fracture toughness specimen is illustrated in Figure C-8.

Figure C-1 Standard Size Tension Test Specimen -- Used in B&W Plant-Specific and SUPCAP Type R1 Capsules

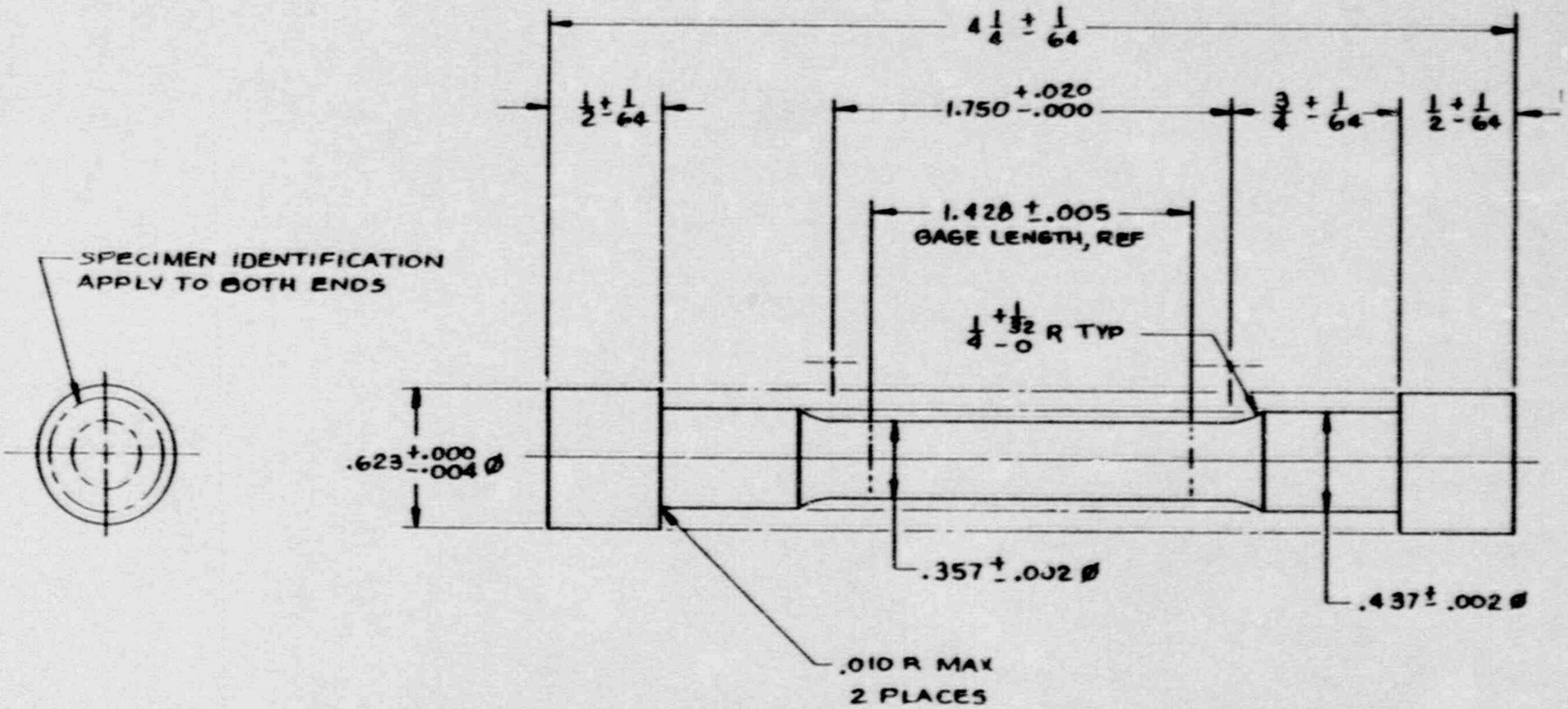


Figure C-2 Subsize Tension Test Specimens -- Used in SUPCAP Type R2 and HUPCAP Capsules

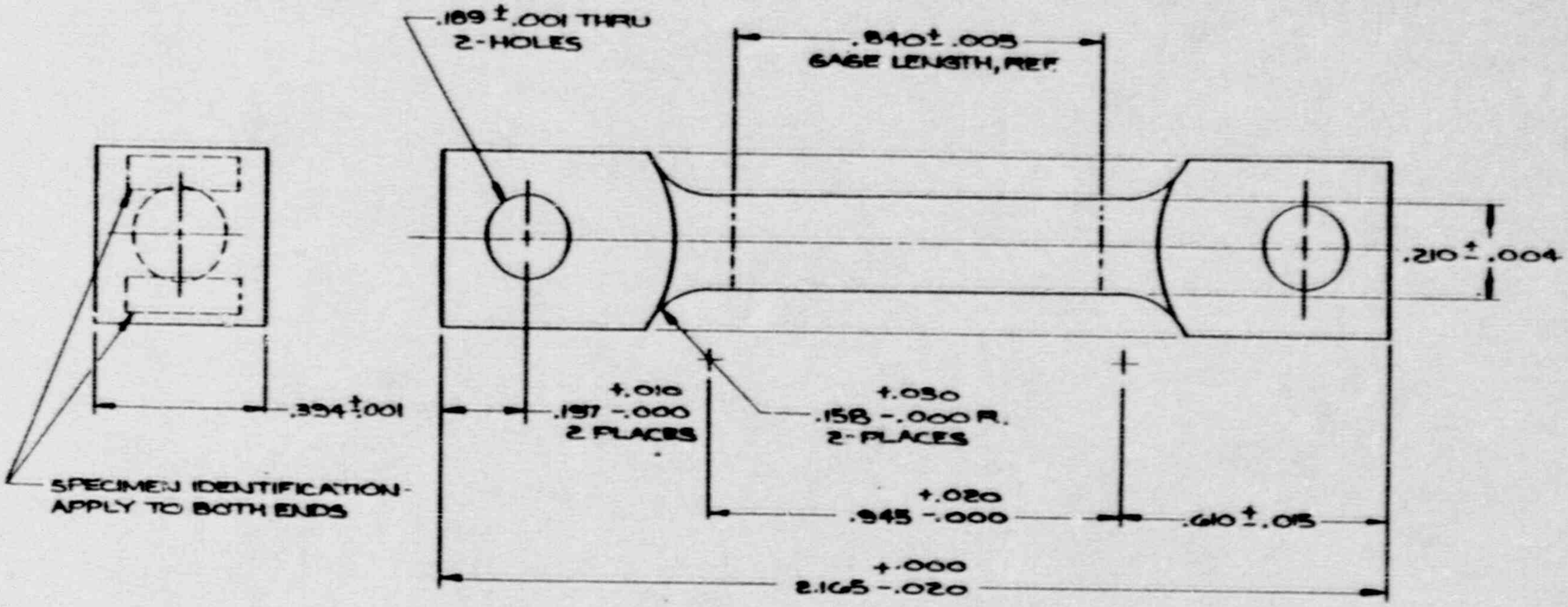
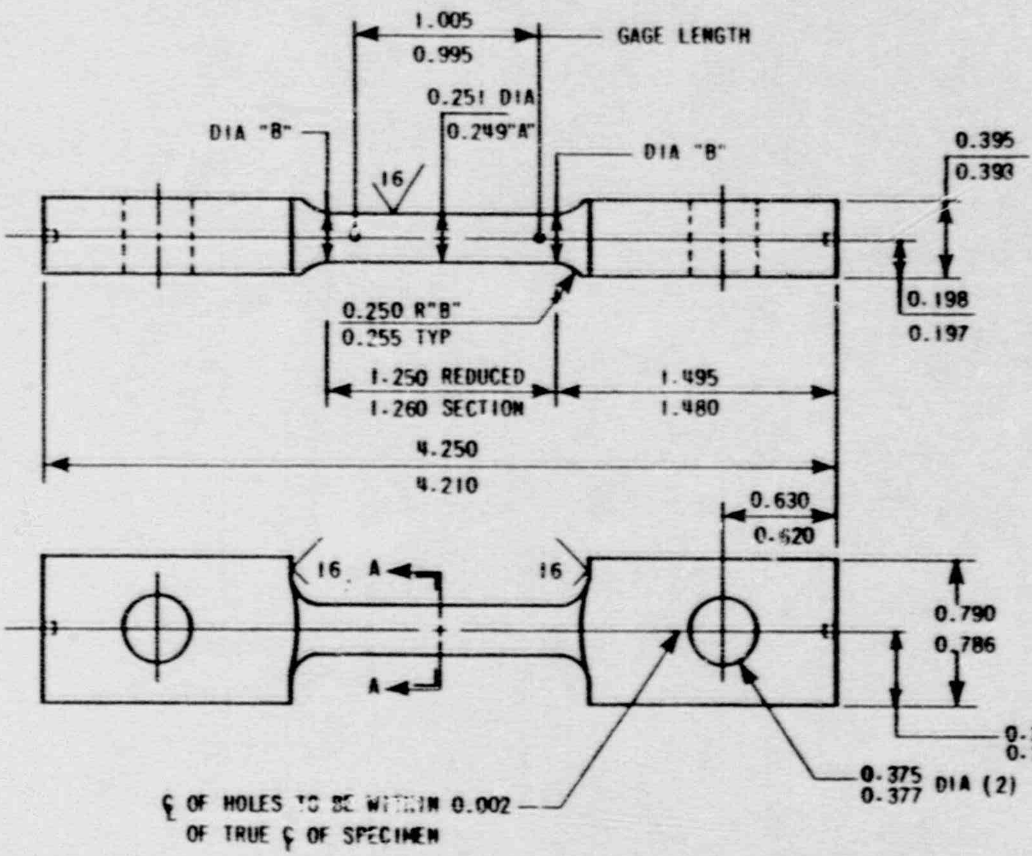


Figure C-3 Tension Test Specimens Used in Westinghouse Plant-Specific Capsules

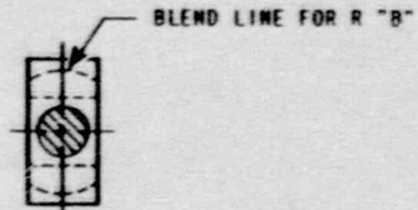


NOTE:

"B" DIA IS TO BE ACTUAL "A" DIA +0.002 TO 0.005 TAPERING TO "A" AT THE CENTER

NOTES:

1. LATHE CENTERS REQUIRED
2. 125 ALL OVER UNLESS OTHERWISE SPECIFIED



SECTION A-A

Figure C-4 Charpy V-Notch Specimens Used in B&W Plant-Specific and SUPCAP Capsules

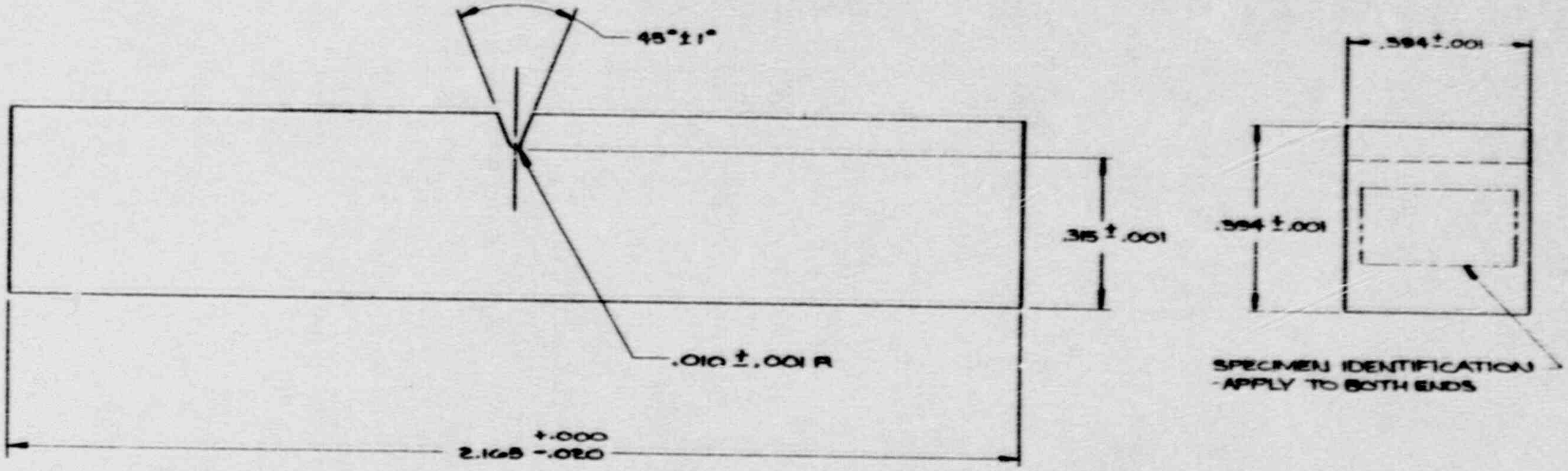


Figure C-5 Charpy V-Notch Specimen Used in Westinghouse Plant-Specific Capsules

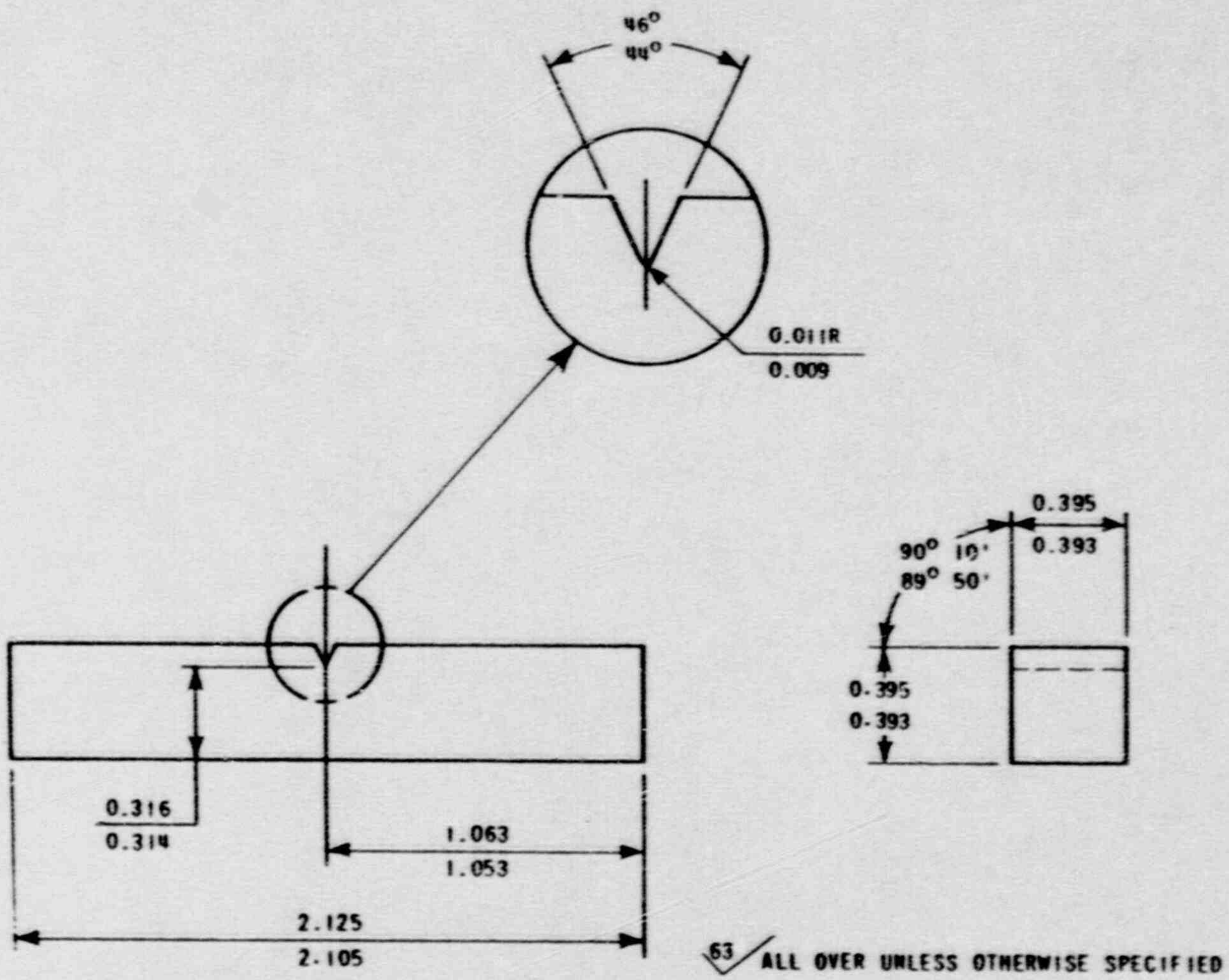


Figure C-6 Charpy V-Notch Specimens Used in HUPCAPS

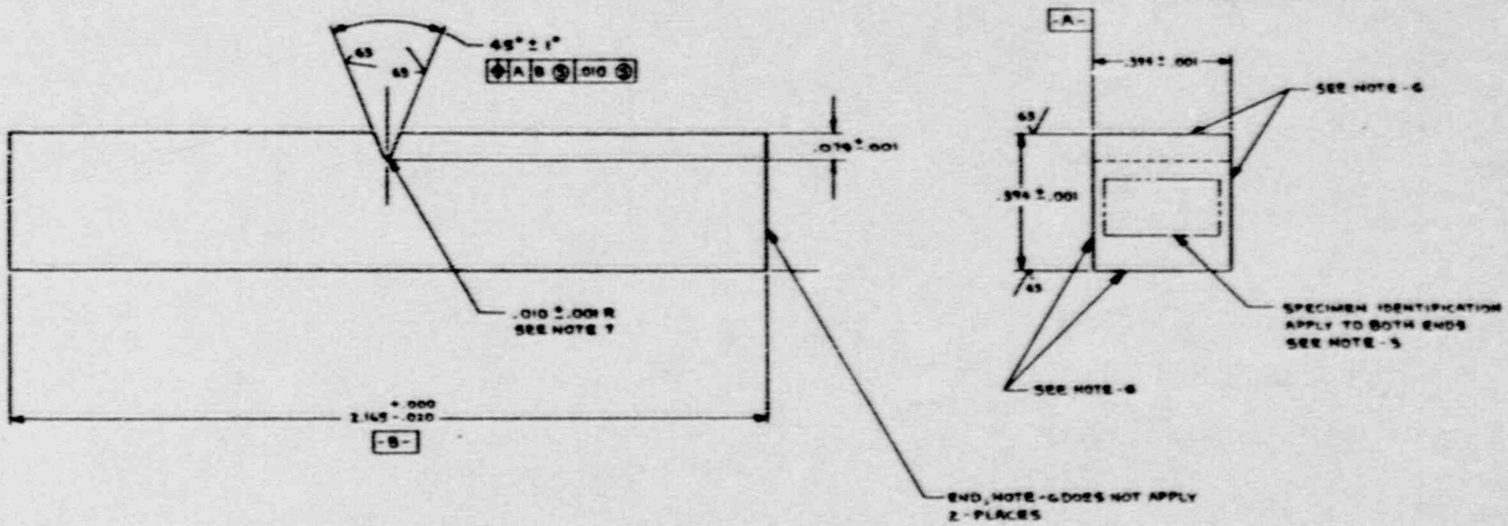


Figure C-7 Rectangular Compact Fracture Toughness Specimens -- Standard Proportions and Modifications for Measurement of Displacement at Load Line

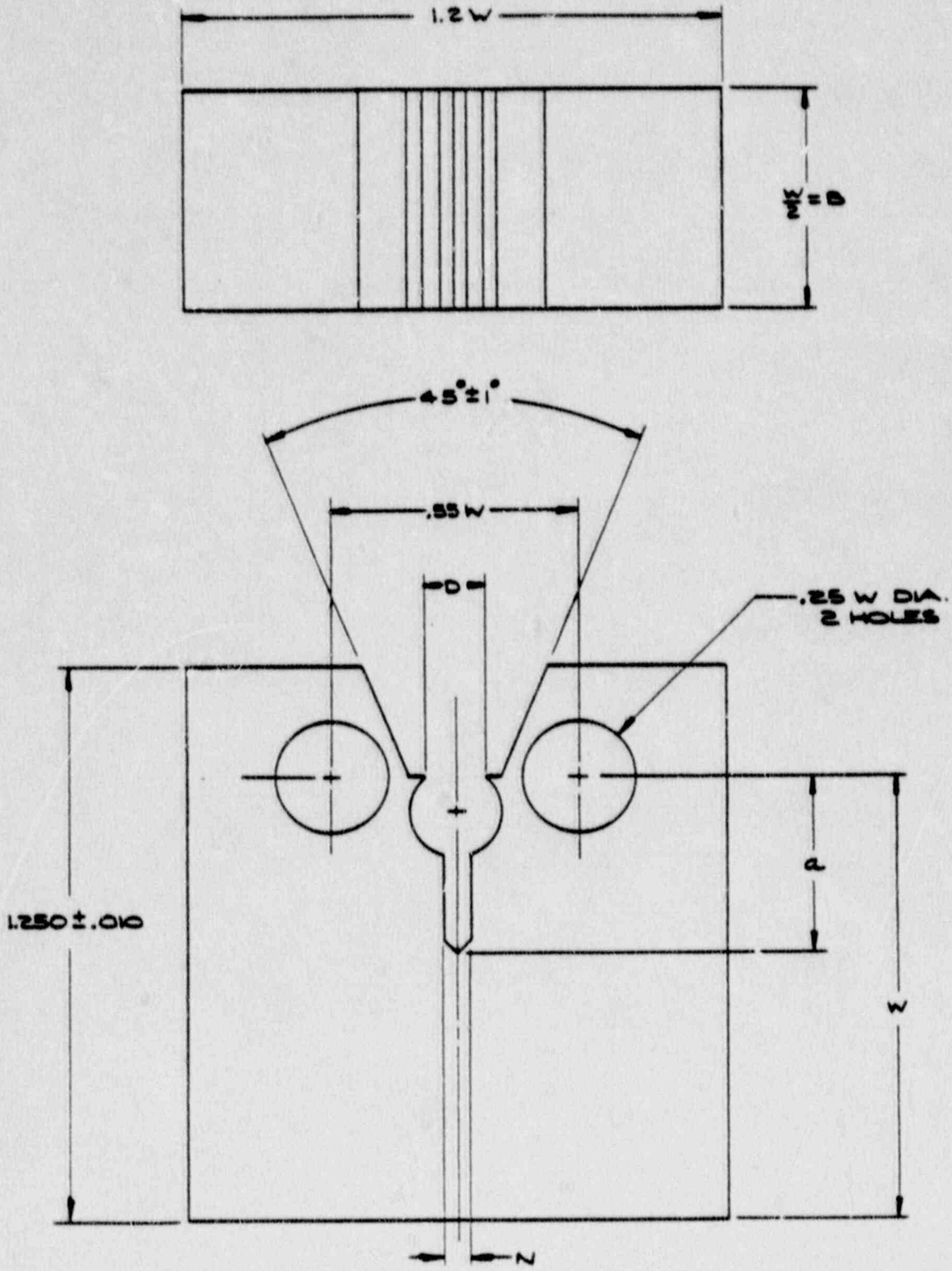


Figure C-8 Round Compact Fracture Toughness Specimens -- Dimensions and Modification for Measurement of Displacement at Load Line

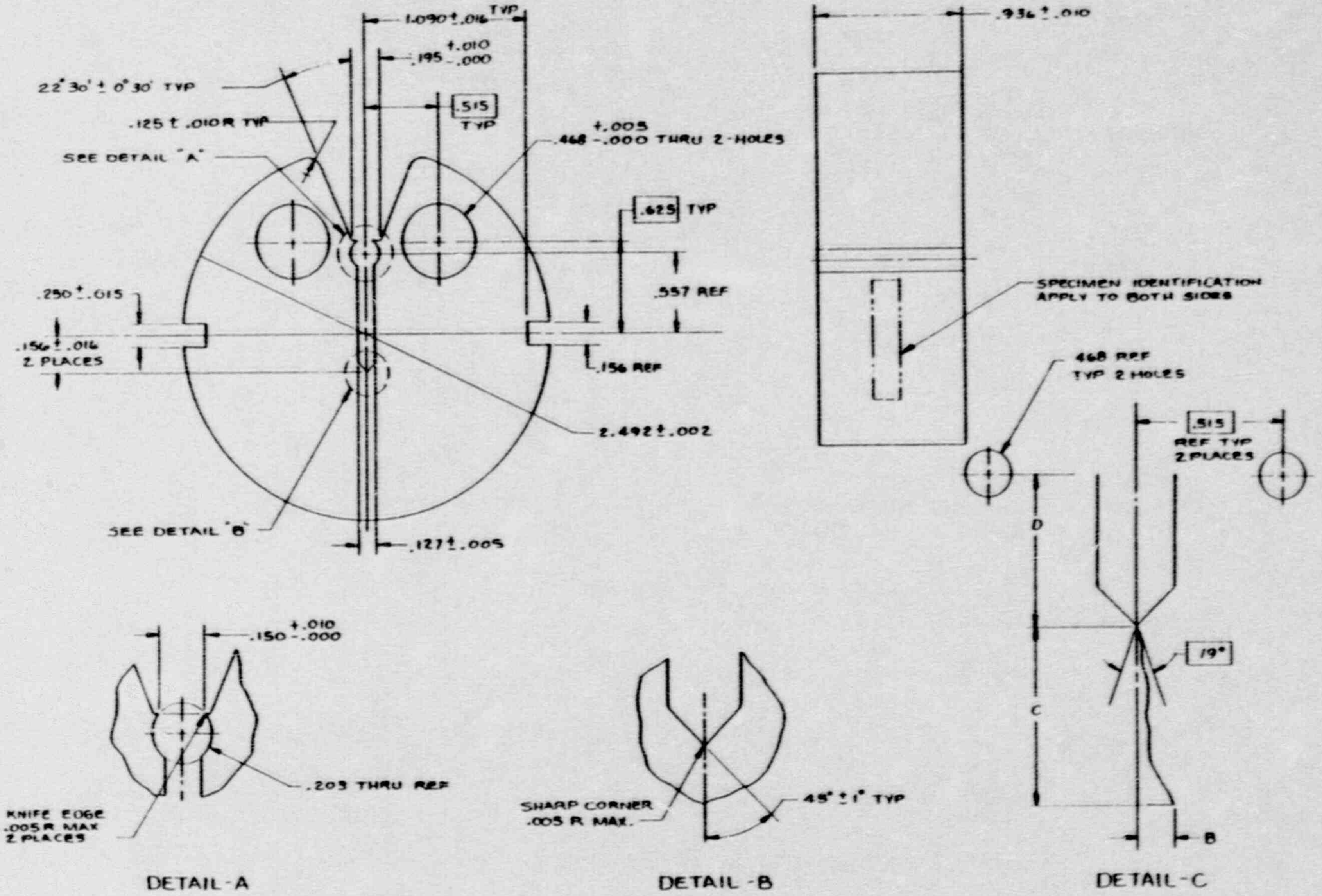
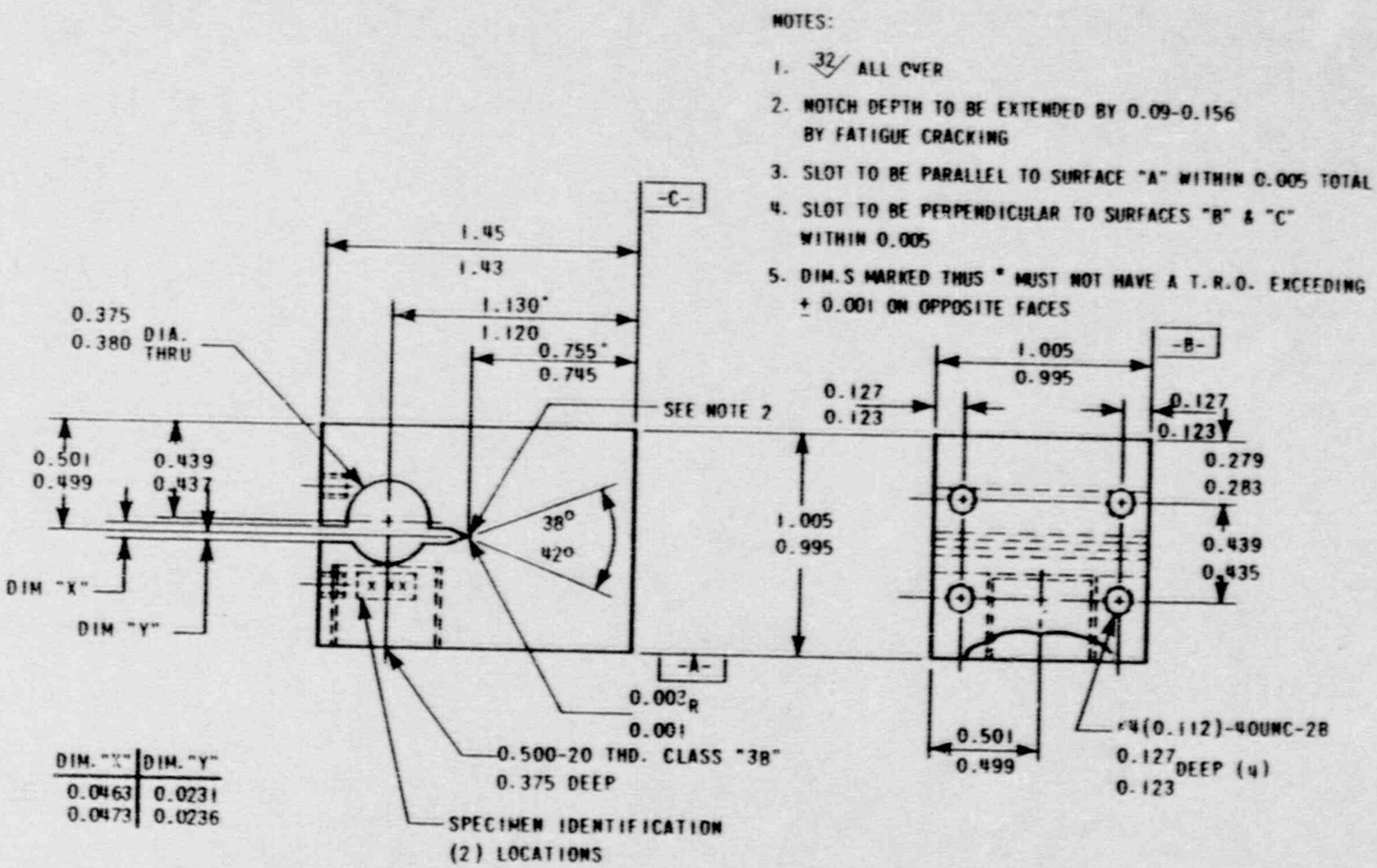


Figure C-9 Wedge-Opening-Loading Specimens



DIM. "X"	DIM. "Y"
0.0463	0.0231
0.0473	0.0236

Figure C-10 Slow Bend (Three Point Loading) Fracture Toughness Specimen

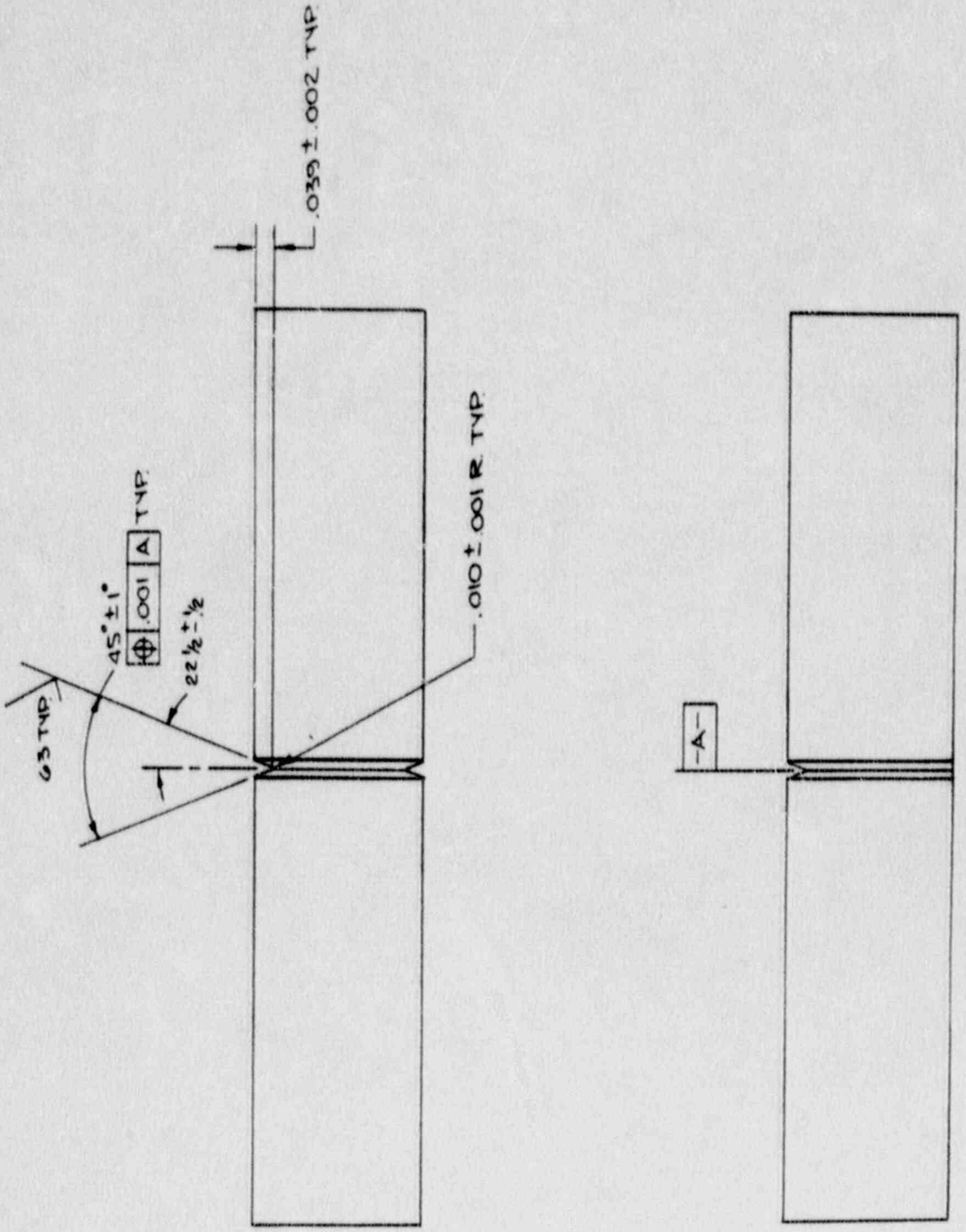
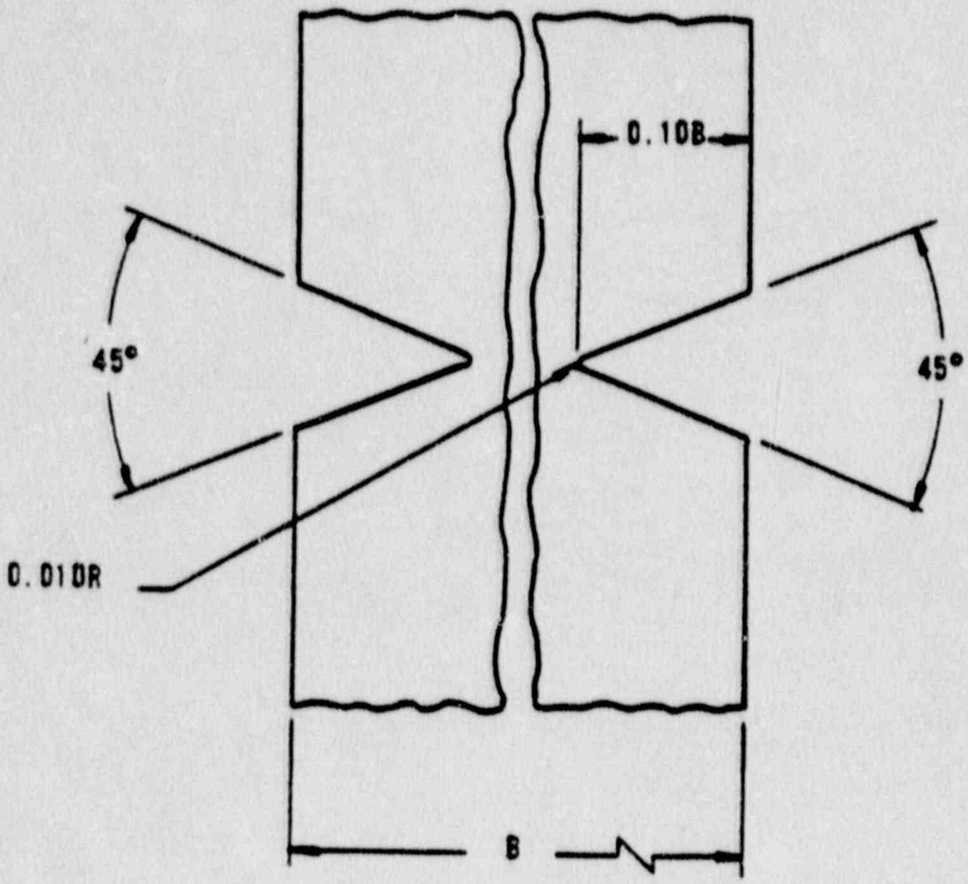


Figure C-11 Geometry of Side Grooves for 0.936 TRCT Specimen



APPENDIX D

MIRVP Plant-Specific Capsule Type Designations and Contents

Table D-1. Materials and Specimens in Surveillance Capsules of Oconee Unit 1

Material Description	No. of Specimens Tension	Charpy
Type I - Capsules OCI-A, -C, -E		
Weld metal, WF-112	4	8
HAZ		
Heat C3265-1, longitudinal	0	8
Beltline base metal, plate		
Heat C3265-1, longitudinal	4	8
transverse	0	4
Correlation, HSST plate 02	0	8
Total per capsule	8	36
Type II - Capsules OCI-B, -D, -F		
HAZ		
Heat C2800-2, longitudinal	4	10
Beltline base metal, plate		
Heat C2800-2, longitudinal	4	10
transverse	0	8
Correlation, HSST plate 02	0	8
Total per capsule	8	36

Table D-2. Materials and Specimens in Surveillance Capsules of Oconee Unit 2

Material Description	No. of Specimens Tension	No. of Specimens Charpy
Type I - Capsules OCII-A, -C, -E		
Weld metal, WF-209-1	4	8
HAZ		
Heat AAW163, longitudinal	0	8
Beltline base metal, forging		
Heat AAW163, longitudinal	4	8
transverse	0	4
Correlation, HSST plate 02	0	8
Total per capsule	8	36
Type II - Capsules OCII-B, -D, -F		
HAZ		
Heat AWG164, longitudinal	4	10
Beltline base metal, forging		
Heat AWG164, longitudinal	4	10
transverse	0	8
Correlation, HSST plate 02	0	8
Total per capsule	8	36

Table D-3. Materials and Specimens in Surveillance Capsules of Oconee Unit 3

Material Description	No. of Specimens Tension	Charpy
Type V - Capsules OCIII-A, -C, -E		
Weld metal, WF-209-1	2	12
HAZ		
Heat A ANK191, longitudinal	0	12
Beltline base metal, forging		
Heat A ANK191, longitudinal	0	9
transverse	2	12
Heat B AWS192, transverse	0	9
Total per capsule	4	54
Type VI - Capsules OCIII-B, -D, -F		
Weld metal WF-209-1	2	12
HAZ		
Heat A ANK191, longitudinal	0	12
Heat B AWS192, longitudinal	0	6
Beltline base metal, forgings		
Heat A ANK191, longitudinal	0	0
transverse	2	12
Heat B AWS192, longitudinal	0	0
transverse	0	6
Correlation, HSST plate 02	0	6
Total per capsule	4	54

Table D-4. Materials and Specimens in Surveillance Capsules of Three Mile Island Unit 1

Material Description	No. of Specimens Tension	Charpy
Type I - Capsules TMI1-A, C, E		
Weld metal, WF-25	4	8
HAZ		
Heat C2789-2, longitudinal	0	8
Beltline base metal, plate		
Heat C2789-2, longitudinal	4	8
transverse	0	4
Correlation, HSST plate 02	0	8
Total per capsule	8	36
Type II - Capsules TMI1-B, -D, - F		
HAZ		
Heat C3307-1, longitudinal	4	10
Beltline base metal, plate		
Heat C3307-1, longitudinal	4	10
transverse	0	8
Correlation, HSST plate 02	0	8
Total per capsule	8	36

Table D-5. Materials and Specimens in Surveillance Capsules of Crystal River Unit 3

Material Description	No. of Specimens Tension	Charpy	0.5 TCT
Type III - Capsules CR3-A, -C, -E			
Weld metal, WF-209-1	2	12	
HAZ			
Heat C4344-1, transverse	0	12	
Heat C4344-2, transverse	0	6	
Beltline base metal, plate			
Heat C4344-1, transverse	2	12	
Heat C4344-2, transverse	0	6	
Correlation, HSST plate 02	0	6	
Total per capsule	4	54	
Type IV - Capsules CR3-B, -D, -F			
Weld metal, WF-209-1	2	12	8
HAZ			
Heat C4344-1, transverse	0	12	0
Beltline base metal, plate			
Heat C4344-1, transverse	2	12	0
Total per capsule	4	36	8

Table D-6. Materials and Specimens in Surveillance Capsules of Arkansas Nuclear One, Unit 1

Material Description	No. of Specimens Tension	Charpy
Type I - Capsules AN1-A, -C, -E		
Weld metal, WF-193	4	8
HAZ		
Heat C5114-1, longitudinal	0	8
Beltline base metal, plate		
Heat C5114-1, longitudinal	4	8
transverse	0	4
Correlation, HSST plate 02	0	8
Total per capsule	8	36
Type II - Capsules AN1-B, -D, -F		
HAZ		
Heat C5114-2, longitudinal	4	10
Beltline base metal, plate		
Heat C5114-2, longitudinal	4	10
transverse	0	8
Correlation, HSST plate 02	0	8
Total per capsule	8	36

Table D-7. Materials and Specimens in Surveillance Capsules of Rancho Seco Unit 1

Material Description	No. of Specimens Tension	Charpy	0.5 TCT
Type III - Capsules RS1-A, -C, -E			
Weld metal, WF-193	2	12	
HAZ			
Heat C5062-1, transverse	0	12	
Heat C5070-1, transverse	0	6	
Beltline base metal, plates			
Heat C5062-1, transverse	2	12	
Heat C5070-1, transverse	0	6	
Correlation, HSST plate 02	0	6	
Total per capsule	4	54	
Type IV - Capsules RS1-B, -D, -F			
Weld metal, WF-193	2	12	8
HAZ			
Heat C5062-1, transverse	0	12	0
Beltline base metal, plate			
Heat C5062-1, transverse	2	12	0
Total per capsule	4	36	8

Table D-8. Materials and Specimens in Surveillance Capsules of Davis-Besse Unit 1

Material Description	No. of Specimens Tension	Charpy	0.5 TCT
Type III - Capsules TE1-A, -C, -E			
Weld metal, WF-182-1	2	12	
HAZ			
Heat BCC241, transverse	0	12	
Heat AKJ233, transverse	0	6	
Beltline base metal, forgings			
Heat BCC241, transverse	2	12	
Heat AKJ233, transverse	0	6	
Correlation, HSST plate 02	0	6	
Total per capsule	4	54	
Type IV - Capsules TE1-B, -D, -F			
Weld metal, WF-182-1	2	12	8
HAZ			
Heat BCC241, transverse	0	12	0
Beltline base metal, forging			
Heat BCC241, transverse	2	12	0
Total per capsule	4	36	8

Table D-9. Materials and Specimens in Surveillance Capsules of R. E. Ginna Unit 1

Material Description	No. of Specimens Tension	Charpy	WOL
Type I - Capsules V, R, T			
Weld metal, SA-1036	3	10	3
HAZ, Heat 125P666	0	10	0
Beltline base metal, forgings			
Heat 125P666, longitudinal	3	10	3
Heat 125S255, longitudinal	3	10	3
Correlation,			
SA 302 Gr. B	0	8	0
Total per capsule	9	48	9
Type II - Capsules N, P, S			
Weld metal, SA-1036	3	12	3
HAZ, Heat 125P666	0	12	0
Beltline base metal, forgings			
Heat 125P666, longitudinal	3	12	3
Heat 125S255, longitudinal	3	12	3
Total per capsule	9	48	9

Table D-10. Materials and Specimens in Surveillance Capsules of Point Beach Unit 1

Material Description	No. of Specimens Tension	Charpy	WOL
Type III - Capsules T & V			
Weld metal, SA-1263	2	8	2
HAZ, Heat A9811	0	8	0
Beltline base metal, plates			
Heat A9811, longitudinal	3	12	3
Heat C1423, longitudinal	4	12	4
Correlation,			
SA 302 Gr. B	0	8	0
Total per capsule	9	48	9
Type IV - Capsules P & S			
Weld metal, SA-1263	0	8	0
HAZ, heat A9811	0	8	0
Beltline base metal, plates			
Heat A9811, longitudinal	5	12	5
Heat C1423, longitudinal	4	12	4
Correlation,			
SA 302, Gr.B	0	8	0
Total per capsule	9	48	9

Table D-10. Materials and Specimens in Surveillance Capsules of Point Beach Unit 1 (Cont'd)

Material Description	No. of Specimens Tension	Charpy	WOL
Type IV - Capsule N			
Weld metal SA-1263	0	8	0
HAZ, heat A9811	0	8	0
Beltline base metal, plates			
Heat A9811, longitudinal	4	12	4
Heat C1423, longitudinal	5	12	5
Correlation,			
SA 302, Gr. B	0	8	0
Total per capsule	9	48	9
Type III - Capsule R			
Weld metal, SA-1263	2	8	2
HAZ, heat A9811	0	8	0
Beltline base metal, plates			
Heat A9811, longitudinal	4	12	4
Heat C1423, longitudinal	3	12	3
Correlation,			
SA-302, Gr. B	0	8	0
Total per capsule	9	48	9

Table D-11. Materials and Specimens in Surveillance Capsules of Point Beach Unit 2

Material Description	No. of Specimens Tension	Charpy	WOL
Type V - Capsules R, S & V			
Weld metal, WF-193	3	8	3
HAZ, Heat 122W195VA1	0	8	0
Beltline base metal, forgings			
Heat 122W195VA1, longitudinal	3	12	3
Heat 123V500VA1, longitudinal	3	12	3
Correlation, HSST plate 02	0	8	0
Total per capsule	9	48	9
Type IV - Capsules N, P, & T			
Weld metal, WF-193	0	8	0
HAZ, heat 122W195VA1	0	8	0
Beltline base metal, forgings			
Heat 122W195VA1	5	12	5
Heat 123V500VA1	4	12	4
Correlation, HSST plate 02	0	8	0
Total per capsule	5	48	9

Table D-12. Materials and Specimens in Surveillance Capsules of Surry Unit 1

Material Description	No. of Specimens Tension	Charpy	WOL
Type VI - Capsules S, U, W, & Y			
Weld metal, SA-1526	0	0	0
HAZ, Heat C4415-1	0	0	0
Beltline base metal, plates			
Heat C4326-1, longitudial	2	10	3
Heat C4415-1, longitudinal	2	10	3
Correlation, HSST plate 02	0	8	0
Total per capsule	4	28	6
Type VII - Capsules T, V, X, & Z			
Weld metal, SA-1526	2	8	2
HAZ, Heat C4415-1, longitudinal	0	8	0
Beltline base metal, plate			
Heat C4415-1, longitudinal	2	8	2
Correlation, HSST plate 02	0	8	0
Total per capsule	4	32	4

Table D-13. Materials and Specimens in Surveillance Capsules of Surry Unit 2

Material Description	No. of Specimens Tension	Charpy	WOL
Type VIII - Capsules X, W, V, & S			
Weld metal, (Rotterdam)	2	8	0
HAZ, Heat C4339-1	0	8	0
Beltline base metal, plates			
Heat C4339-1, longitudinal	0	10	0
transverse	2	10	4
Correlation, HSST plate 02	0	8	0
Total per capsule	4	44	4
Type VIII - Capsules T & U			
Weld metal, (Rotterdam)	2	8	0
HAZ, Heat C4339-1	0	8	0
Beltline base metal, plates			
Heat C4339-1, longitudinal	2	10	4
transverse	0	10	0
Correlation, HSST plate 02	0	8	0
Total per capsule	4	44	4
Type IX - Capsules Y & Z			
Weld metal, (Rotterdam)	2	8	4
HAZ, Heat C4339-1	0	8	0
Beltline base metal, plates			
Heat C4339-1, longitudinal	2	10	0
transverse	0	10	0
Correlation, HSST plate 02	0	8	0
Total per capsule	4	44	4

Table D-14. Materials and Specimens in Surveillance Capsules of Turkey Point Unit 3

Material Description	No. of Specimens Tension	Charpy	WOL
Type VI - Capsules S, U, W, Y, & Z			
Weld metal, SA-1101	0	0	0
HAZ, Heat 123P461VA-1	0	0	0
Beltline base metal, forgings			
Heat, 123P461VA-1, longitudinal	2	10	3
Heat, 123S266VA-1, longitudinal	2	10	3
Correlation, SA 302, Gr.B	0	8	0
Total per capsule	4	28	6
Type VII - Capsules V & X			
Weld metal, SA-1101	2	8	2
HAZ, Heat 123P461VA-1	0	8	0
Beltline base metal, forgings			
Heat 123P461VA-1, longitudinal	0	0	0
Heat 123S266VA-1, longiduainal	2	8	2
Correlation, SA 302, Gr.B	0	8	0
Total per capsule	4	32	4
Type VII - Capsule T			
Weld metal, SA-1101	2	8	2
HAZ, Heat 123P461VA-1	0	8	0
Beltline base metal, forgings			
Heat 123P461VA-1, longitudinal	2	8	2
Heat 123S266VA-1, longitudinal	0	0	0
Correlation, SA 302, Gr.B	0	8	0
Total per capsule	4	32	4

Table D-15. Materials and Specimens in Surveillance Capsules of Turkey Point Unit 4

Material Description	No. of Specimens Tension	Charpy	WOL
Type VI - Capsules S, U, W, Y, & Z			
Weld metal, SA-1094	0	0	0
HAZ, Heat 123P481VA-1	0	0	0
Beltline base metal, forgings			
Heat 123P481VA-1, longitudinal	2	10	3
Heat 122S180VA-1, longitudinal	2	10	3
Correlation, HSST plate 02	0	8	0
Total per capsule	4	28	6
Type VII - Capsules V, X, & T			
Weld metal, SA-1094	2	8	2
HAZ, Heat 123P481VA-1	0	8	0
Beltline base metal, forgings			
Heat 123P481VA-1, longitudinal	0	0	0
Heat 122S180VA-1, longitudinal	2	8	2
Correlation, HSST plate 02	0	8	0
Total per capsule	4	32	4

Table D-16. Materials and Specimens in Surviellance Capsules of Zion Unit 1

Material Description	No. of Specimens Tension	Charpy	WOL
Type VIII - Capsules S, V, W, & X			
Weld metal, WF-209-1	2	8	0
HAZ, Heat B7835-1	0	8	0
Beltline base metal, plates			
Heat B7835-1, longitudinal	0	10	0
transverse	2	10	4
Correlation, HSST plate 02	0	8	0
Total per capsule	4	44	4
Type VIII - Capsules T & U			
Weld metal, WF-209-1	2	8	0
HAZ, Heat B7835-1	0	8	0
Beltline base metal, plates			
Heat B7835-1, longitudinal	2	10	4
transverse	0	10	0
Correlation, HSST plate 02	0	8	0
Total per capsule	4	44	4
Type IX - Capsules Y & Z			
Weld metal, WF-209-1	2	8	4
HAZ, Heat B7835-1	0	8	0
Beltline base metal, plates			
Heat B7835-1, longitudinal	2	10	0
transverse	0	10	0
Correlation, HSST plate 02	0	8	0
Total per capsule	4	44	4

Table D-17. Materials and Specimens in Surviellance Capsules of Zion Unit 2

Material Description	No. of Specimens Tension	Charpy	WOL
Type VIII - Capsules S, V, W, & X			
Weld metal, WF-209-1	2	8	0
HAZ, Heat C4007-1	0	8	0
Beltline base metal, plate			
Heat C4007-1, longitudinal	0	10	0
transverse	2	10	4
Correlation, HSST plate 02	0	8	0
Total per capsule	4	44	4
Type VIII - Capsules T, & U			
Weld metal, WF-209-1	2	8	0
HAZ, Heat C4007-1	0	8	0
Beltline base metal, plate			
Heat C4007-1, longitudinal	2	10	4
transverse	0	10	0
Correlation, HSST plate 02	0	8	0
Total per capsule	4	44	4
Type IX - Capsules Y, & Z			
Weld metal, WF-209-1	2	8	4
HAZ, Heat C4007-1	0	8	0
Beltline base metal, plate			
Heat C4007-1, longitudinal	2	10	0
transverse	0	10	0
Correlation, HSST plate 02	0	8	0
Total per capsule	4	44	4

APPENDIX E

Status of Surveillance Capsules

Table E-1. Summary Status of the B&W Surveillance Capsules

Capsule ID	Weld Metal/ Compacts	Status/ Location	Fluence Target	Fluence Expected/ Received	Estimated Time of Removal	Comments
OC1-F	No/No	Tested	--	5.7E17	--	Reported in BAW-1421, Rev. 1; corrected in BAW-1436
OC1-E	Yes/No	Tested	--	1.5E18	--	Reported in BAW-1436
OC1-B	No/No	Removed	4.4E18	7.0E18	--	Held in storage
OC1-A	Yes/No	Tested	--	9.0E18	--	Reported in BAW-1837
OC1-C	Yes/No	Tested	--	9.9E18	--	Reported in BAW-2059
OC1-D	No/No	Holding	1.0E19	1.1E19	CR3-cycle 13	---
OC2-C	Yes/No	Tested	--	9.4E17	--	Reported in BAW-1437
OC2-A	Yes/No	Tested	--	3.4E18	--	Reported in BAW-1699
OC2-B	No/No	CR3-WZ	5.8E18	5.8E18	CR3-cycle 7	---
OC2-E	Yes/No	Tested	--	1.2E19	--	Reported in BAW-2051
OC2-D	No/No	Holding	9.6E18	1.0E19	CR3-cycle 12	---
OC2-F	No/No	Holding	9.6E18	1.0E19	CR3-cycle 11	---
OC3-A	Yes/No	Tested	--	7.4E17	--	Reported in BAW-1438
OC3-B	Yes/No	Tested	--	3.1E18	--	Reported in BAW-1697
OC3-C	Yes/No	CR3-WX	9.6E18	8.6E18	CR3-cycle 8	---
OC3-D	Yes/No	CR3-YX	1.6E19	1.5E19	CR3-cycle 7	---
OC3-E	Yes/No	Holding	1.6E19	1.6E19	CR3-cycle 12	---
OC3-F	Yes/No	CR3-ZY	1.6E19	1.6E19	CR3-cycle 13	---
TMI1-E	Yes/No	Tested	--	1.0E18	--	Reported in BAW-1439
TMI1-B	No/No	CR3-XW	4.0E18	4.2E18	CR3-cycle 7	---
TMI1-C	Yes/No	Tested	--	8.7E18	--	Reported in BAW-1901
TMI1-A	Yes/No	Removed	8.0E18	1.6E18	--	Held in storage - reported in BAW-2042
TMI1-D	No/No	Holding	8.0E18	1.0E19	CR3-cycle 11	---
TMI1-F	No/No	Holding	8.0E18	7.1E18	CR3-cycle 8	---
CR3-B	Yes/Yes	Tested	--	1.0E18	--	Reported in BAW-1679 and BAW-1718
CR3-C	Yes/No	Tested	--	6.6E18	--	Reported in BAW-1899
CR3-D	Yes/Yes	Tested	--	7.5E18	--	Reported in BAW-1899 and BAW-1914
CR3-F	Yes/Yes	Tested	--	1.1E19	--	Reported in BAW-2049
CR3-A	Yes/No	CR3-ZY	1.1E19	1.1E19	CR3-cycle 8	---
CR3-E	Yes/No	CR3-XW	1.1E19	1.1E19	CR3-cycle 8	---

Table E-1. Summary Status of the B&W Surveillance Capsules (Cont'd)

Capsule ID	Weld Metal/ Compacts	Status/ Location	Fluence Target	Fluence Expected/ Received	Estimated Time of Removal	Comments
AN1-E	Yes/No	Tested	--	7.3E17	--	Reported in BAW-1440
AN1-B	No/No	Tested	--	4.3E18	--	Reported in BAW-1698
AN1-A	Yes/No	Tested	--	1.0E19	--	Reported in BAW-1836
AN1-C	Yes/No	Tested	--	1.5E19	--	Reported in BAW-2075
AN1-D	No/No	DB1-YZ	9.8E18	9.5E18	DB1-cycle 6	---
AN1-F	No/No	DB1-YX	9.8E18	9.9E18	DB1-cycle 7	---
RS1-B	Yes/Yes	Tested	--	4.0E18	--	Reported in BAW-1702 and BAW-1720
RS1-A	Yes/No	Removed	5.3E18	7.2E18	--	Held in storage
RS1-D	Yes/Yes	Tested	--	6.6E18	--	Reported in BAW-1792 and BAW-1793P
RS1-F	Yes/Yes	Tested	--	1.4E19	--	Reported in BAW-2074
RS1-C	Yes/No	Removed	8.9E18	1.1E19	--	Held in storage
RS1-E	Yes/No	Removed	8.9E18	1.3E19	--	Held in storage
TE1-F	Yes/Yes	Tested	--	2.0E18	--	Reported in BAW-1701 and BAW-1719
TE1-B	Yes/Yes	Tested	--	5.9E18	--	Reported in BAW-1834 and BAW-1867
TE1-A	Yes/No	Tested	--	1.3E19	--	Reported in BAW-1882
TE1-D	Yes/Yes	DB1-XW	1.1E19	1.3E19	DB1-cycle 6	---
TE1-C	Yes/No	DB1-YX	1.1E19	1.9E19	DB1-cycle 6	---
TE1-E	Yes/No	DB1-ZY	1.1E19	1.7E19	DB1-cycle 11	---
CR3-LG1	--	Tested	--	6.1E18	--	Reported in BAW-1910P
CR3-LG2	--	CR3-WZ	1.7E19	1.7E19	CR3-cycle 9	---
DB1-LG1	--	Tested	--	8.3E18	--	Reported in BAW-1920P
DB1-LG2	--	DB1-ZY	2.2E19	2.1E19	DB1-cycle 11	---
TMI2-LG1	--	CR3-YZ	8.0E18	9.8E18	CR3-cycle 12	---
TMI2-LG2	--	CR3-YZ	1.7E19	1.9E19	CR3-cycle 9	---
A1	--	DB1-YZ	3.0E19	3.2E19	DB1-cycle 17	---
A2	--	CR3-YX	3.0E19	3.0E19	CR3-cycle 18	---
A3	--	DB1-YZ	1.7E19	1.7E19	DB1-cycle 12	---
A4	--	CR3-YX	3.0E19	3.0E19	CR3-cycle 18	---
A5	--	DB1-YX	1.7E19	1.7E19	DB1-cycle 12	---
L1	--	DB1-WZ	1.7E19	1.7E19	DB1-cycle 12	---
L2	--	DB1-WZ	1.7E19	1.7E19	DB1-cycle 12	---

Table E-2. Summary Status of the Westinghouse Surveillance Capsules

Capsule ID	Weld Metal/ WOL's	Current Location	Fluence Target	Fluence Expected/ Received	Estimated Time of Removal	Comments
REG-N	Yes/Yes	33°*	--	Standby	--	Transfer to new position when possible**
REG-P	Yes/Yes	23°	4.1E19	4.1E19	Cycle 26	---
REG-R	Yes/Yes	Tested	--	1.2E19	--	Reported in WCAP-8421
REG-S	Yes/Yes	33°	--	Standby	--	---
REG-T	Yes/Yes	Tested	--	1.8E19	--	Reported in WCAP-10086
REG-V	Yes/Yes	Tested	--	6.0E18	--	Reported in W report dated 3/73
PB1-N	Yes/Yes	33°	--	Standby	--	---
PB1-P	Yes/No	23°	3.9E19	3.9E19	Cycle 22	---
PB1-R	Yes/Yes	Tested	--	2.2E19	--	Reported in WCAP-9357
PB1-S	Yes/No	Tested	--	7.6E18	--	Reported in WCAP-8739
PB1-T	Yes/Yes	Tested	--	2.1E19	--	Reported in WCAP-10736
PB1-V	Yes/Yes	Tested	--	5.4E18	--	Reported in BCL report dated 6/73
PB2-N	Yes/No	33°	--	Standby	--	---
PB2-P	Yes/No	23°	4.0E19	4.1E19	Cycle 23	---
PB2-R	Yes/Yes	Tested	--	2.4E19	--	Reported in WCAP-9635
PB2-S	Yes/Yes	33°	2.9E19	3.1E19	Cycle 16	---
PB2-T	Yes/No	Tested	--	9.2E18	--	Reported in WCAP-9331
PB2-V	Yes/Yes	Tested	--	6.8E18	---	Reported in BCL report dated 6/75
S1-S	No/No	25°	4.0E19	Standby	--	---
S1-T	Yes/Yes	Tested	--	2.8E18	--	Reported in BCL report dated 6/75, includes WOL
S1-U	No/No	45°	4.0E19	Standby	--	Transfer to new position when possible**
S1-V	Yes/Yes	Tested	--	1.9E19	--	Reported in WCAP-11415
S1-W	No/No	Tested	--	4.0E18	--	Reported in BCL-585-8R, only dosimetry evaluated
S1-X	Yes/Yes	25°	2.8E19	Standby	--	---
S1-Y	No/No	35°*	4.0E19	standby	--	Transfer to new position when possible**
S1-Z	Yes/Yes	25°	3.9E19	standby	--	---

Table E-2. Summary Status of the Westinghouse Surveillance Capsules (Cont'd)

Capsule ID	Weld Metal/ WOL's	Current Location	Fluence Target	Fluence Expected/ Received	Estimated Time of Removal	Comments
S2-S	Yes/No	45°	3.4E19	Standby	--	Transfer to new position when possible**
S2-T	Yes/No	35°	3.4E19	Standby	--	Transfer to new position when possible**
S2-U	Yes/No	25°	3.4E19	Standby	--	---
S2-V	Yes/No	Tested	--	1.9E19	--	Reported in WCAP-11499
S2-W	Yes/No	Tested	--	6.0E18	--	Reported in BCL-585-026, only dosimetry evaluated
S2-X	Yes/No	Tested	--	3.0E18	--	Reported in BCL report dated September, 1975
S2-Y	Yes/Yes	25°F	2.7E19	Standby	--	---
S2-Z	Yes/Yes	35°F	3.4E19	Standby	--	Transfer to new position when possible**
TP3-S	No/No	Tested	--	1.4E19	--	Reported in SWRI-02-5131
TP3-T	Yes/Yes	Tested	--	5.7E18	--	Reported in WCAP-8631
TP3-U	No/No	30°	--	Standby	--	---
TP3-V	Yes/Yes	Tested	--	1.2E19	--	Reported in SWRI-06-8575
TP3-W	No/No	40°	--	Standby	--	---
TP3-X	Yes/Yes	40°	2.8E19	2.8E19	Cycle 24	Transfer to new position when possible**
TP3-Y	No/No	30°	--	Standby	--	---
TP3-Z	No/No	40°	--	Standby	--	---
TP4-S	No/No	Tested	--	1.2E19	--	Reported in SWRI-02-5380
TP4-T	Yes/Yes	Tested	--	6.0E18	--	Reported in SWRI-02-4221
TP4-U	No/No	30°	--	Standby	--	---
TP4-V	Yes/Yes	20°	2.3E19	2.5E19	Cycle 17	Transfer to new position when possible**
TP4-W	No/No	40°	--	Standby	--	---
TP4-X	Yes/Yes	40°	--	Standby	--	Transfer to new position when possible**
TP4-Y	No/No	30°	--	Standby	--	---
TP4-Z	No/No	40°	--	Standby	--	---

Table E-2. Summary Status of the Westinghouse Surveillance Capsules (Cont'd)

Capsule ID	Weld Metal/ WOL's	Current Location	Fluence Target	Fluence Expected/ Received	Estimated Time of Removal	Comments
Z1-S	Yes/No	4^{0}	1.5E19	standby	--	---
Z1-T	Yes/No	Tested	--	2.8E18	--	Reported in BCL-585-4
Z1-U	Yes/No	Tested	--	8.9E18	--	Reported in WCAP-9890
Z1-V	Yes/No	4^{0}	1.5E19	standby	--	---
Z1-W	Yes/No	4^{0}	1.5E19	standby	--	Transfer to new position when possible**
Z1-X	Yes/No	Tested	--	1.4E19	--	Reported in SWRI-06-7484-001
Z1-Y	Yes/Yes	Tested	--	1.8E19	--	Report to be published
Z1-Z	Yes/Yes	4^{0}	2.2E19	2.2E19	Cycle 21	Transfer to new position when possible**
Z2-S	Yes/No	4^{0}	1.5E19	Standby	--	---
Z2-T	Yes/No	Tested	--	9.8E18	--	Reported in SWRI report dated 7/83
Z2-U	Yes/No	Tested	--	2.0E18	--	Reported in BCL-585-4
Z2-V	Yes/No	4^{0}	1.5E19	Standby	--	---
Z2-W	Yes/No	4^{0}	1.5E19	Standby	--	---
Z2-X	Yes/No	4^{0}	2.2E19	Standby	--	Transfer to new posistion when possible**
Z2-Y	Yes/Yes	Tested	--	1.8E19	--	Report to be published
Z2-Z	Yes/Yes	4^{0}	1.5E19	Standby	--	Transfer to new position when possible**
S2-W1	--	15^{0}	1.0E19	1.1E19	Cycle 17	HUPCAP installed in S2 at EOC-10

*All location are shown as relative symmetrical positions and are not always absolute, eg. 0^{0} is equivalent to 90^{0}, 180^{0}, or 270^{0}.

**Fluence values indicated are projected to end of license. Shuffling may be necessary to maintain the lead factor.

Table E-3. Comparison of the Plant-Specific Surveillance Capsules with ASTM E 185 Requirements

Plant	ASTM E 185-82 6 Capsule Program Requirement					
	1.5 EFPY or Fluence > 5E18 RT_{NDT} - 50F	3 EFPY or Fluence Midway First and Third	6 EFPY or 1/4 EOL Fluence	15 EFPY or 1S EOL Fluence	Standby (1-2 Times 1S EOL Fluence)	Standby (1-2 Times 1S EOL Fluence)
Oconee-1[(1)]	F-I/T (5.7E17)	E-I/T (1.5E18)	A-I/T (9.0E18)	C-I/T (9.9E18)	B-I/NT (7.0E18)	D-R
Oconee-2[(1)]	C-I/T (9.4E17)	A-I/T (3.4E18)	B-R	E-I/T (1.2E19)	D-R	F-R
Oconee-3[(1)]	A-I/T (7.4E17)	B-I/T (3.1E18)	C-R	D-R	E-R	F-R
TMI-1[(1)]	E-I/T (1.0E18)	C-I/T (8.7E18)	F-R	D-R	A-I/NT (1.6E18)	B-R
Crystal River-3[(1)]	B-I/T (1.0E18)	C-I/T (6.6E18)	D-I/T (7.5E18)	F-I/T (1.1E19)	A-R	E-R
ANO-1[(1)]	E-I/T (7.3E17)	B-I/T (4.3E18)	A-I/T (1.0E19)	C-I/T (1.7E19)	D-R	F-R
Rancho Seco[(1)]	B-I/T (4.0E18)	A-I/NT (7.2E18)	D-I/T (6.6E18)	F-I/NT (1.4E19)	C-I/NT (1.1E19)	E-I/NT (1.1E19)
Davis-Besse[(1)]	F-I/T (2.0E18)	B-I/T (5.9E18)	A-I/T (1.3E19)	D-R	C-R	E-R
R. E. Ginna[(2)]	V-I/T (6.0E18)	R-I/T (1.2E19)	T-I/T (1.8E19)	P-R	S-R	N-R
Point Beach-1[(2)]	V-I/T (5.4E18)	S-I/T (7.6E18)	T-I/T (2.1E19)	R-I/T (2.2E19)	N-R	P-R
Point Beach-2[(2)]	V-I/T (6.8E18)	T-I/T (9.2E18)	R-I/T (2.4E19)	S-R	P-R	N-R
Surry-1[(2)]	T-I/T (2.8E18)	W-I/T (4.0E18)*	V-I/T (1.9E19)	X-R	S,U-R	Y,Z-R
Surry-2[(2)]	X-I/T (3.0E18)	W-I/T (6.0E18)*	V-I/T (1.9E19)	Y-R	S,T-R	U,Z-R
Turkey Point-3[(3)]	T-I/T (5.7E18)	V-I/T (1.2E19)	--	X-R	U,W-R	Y,Z-R
Turkey Point-4[(3)]	T-I/T (6.0E18)	--	X-R	V-R	U,W-R	Y,Z-R
Zion-1[(2)]	T-I/T (2.8E18)	U-I/T (8.9E18)	X-I/T (1.4E19)	Y-I/T (1.8E19)	W,Z-R	S,V-R
Zion-2[(2)]	U-I/T (2.0E18)	T-I/T (9.8E18)	Y-I/T (1.8E19)	Z-R	S,V-R	W,X-R

Legend: A - X/Y (BE19)

A - Capsule ID

X/Y - I/T (Irradiated/Tested); I/NT (Irradiated/Not Tested); R (In-Reactor)

(BE19) - Fluence at time of capsule withdrawal

* Only dosimetry evaluated.

(1) All capsules in the B&W Integrated Program BAW-1543, Rev. 2.

(2) Plant-specific programs.

(3) All capsules in the integrated program described in Safety Evaluation Report, from J. W. Williams, Jr. to D. G. Eisenhut, dated February 8, 1985.

APPENDIX F

Estimated End-of-Life Reference Temperature and Upper Shelf Energy Information

The estimated end-of-life reference temperature and Charpy upper shelf energy information is provided for plant-to-plant comparisons and to evaluate the surveillance needs for each plant. All specific information and calculational results are presented on Tables F1-F17. The bases for the tabular information are presented below.

Material Identification - All weld numbers and locations were verified by reviewing B&W Mt. Vernon QA records. The materials included conform to the beltline definition of 10CFR50, Appendix G.[2] Those welds in Westinghouse-design vessels that were not fabricated by B&W are so noted.

Chemical Composition - The listed weld metal compositions were obtained from BAW-1500[96] in most cases. The exceptions are: 1) certain welds (referred to as Category 3 welds in BAW-1500) for which additional information was evaluated with the data in BAW-1500. This re-evaluation will be reported in a revision to BAW-1500.[113] 2) The nickel concentration listed for SA-1101 is 0.60 wt. %. The listed nickel value is based on an evaluation performed by the Florida Power and Light Co. and has been the subject of a Safety Evaluation Report[106]. This value agrees closely with the 0.61 wt. % value reported in BAW-1500 and therefore has been adopted in the following tables. The atypical weld composition was obtained from BAW-10144A.[101]

Inside Surface Neutron Fluence - The peak end-of-life IS fluences are listed below. For the B&W-design vessels, the fluences were obtained from plant specific reactor vessel fluence analysis reports compiled in B&W document 51-1174025-00.[109] For the Westinghouse-design vessels, the fluences were obtained from applicable Westinghouse reactor vessel fluence reports or determinations by the Owner; this is footnoted in the tables.

Plant	Date Construction Permit Issued	Date Operating License Issued	License Expiration	Peak EOL (32 EFPY) IS Fluence, n/cm² (E > 1 MeV)
Oconee-1	November 6, 1967	February 6, 1973	November 6, 2007	1.02E19
Oconee-2	November 6, 1967	October 6, 1973	November 6, 2007	9.57E18
Oconee-3	November 6, 1967	July 19, 1974	November 6, 2007	1.56E19
TMI-1	May 18, 1968	April 19, 1974	May 18, 2008	7.98E18
Crystal River-3	September 25, 1968	December 3, 1976	September 25, 2008	1.06E19
ANO-1	December 6, 1968	May 21, 1974	December 6, 2008	9.75E18
Rancho Seco	October 11, 1968	August 16, 1974	October 11, 2008	8.87E18
Davis-Besse	March 24, 1971	April 22, 1977	March 24, 2011	1.10E19
R. E. Ginna	April 25, 1966	September 19, 1969	April 25, 2006	4.10E19
Point Beach-1	July 19, 1967	October 5, 1970	July 19, 2007*	2.45E19
Point Beach-2	July 25, 1968	May 25, 1972	July 25, 2008*	2.55E19
Surry-1	June 25, 1968	May 25, 1972	June 25, 2008	3.96E19**
Surry-2	June 25, 1968	January 29, 1973	June 25, 2008	3.43E19**
Turkey Point-3	April 27, 1967	July 19, 1972	April 27, 2007	2.79E19
Turkey Point-4	April 27, 1967	April 10, 1973	April 27, 2007	2.70E19
Zion-1	December 26, 1968	April 6, 1973	December 26, 2008	1.70E19
Zion-2	December 26, 1968	November 14, 1973	December 26, 2008	1.80E19

*License expiration dates are now October 5, 2010 and March 8, 2013 for Point Beach Units 1 and 2, respectively as License Amendment No. 107 has been approved.

**Surry Unit 1 fluence is calculated at 28.8 EFPY and Surry Unit 2 is calculated at 29.4 EFPY.

The fluences shown for the welds in the B&W-design reactor vessels (Tables F-1 through F-8) were obtained by multiplying the peak IS fluence by the spatial fluence factors given in BAW-1485, Revision 1.[110] For the Westinghouse-design plants (Tables F-9 through F-17), the spatial fluence factors were determined from the plant specific fluence evaluation reports. Fluence attenuation to the quarter-thickness location was performed in accordance with the rules of Regulatory Guide 1.99, Revision 2.[111] Table F-18 shows the fuel management schemes that were utilized in the fluence projection calculations.

Initial RT_{NDT} and Margin

The initial RT_{NDT} values were obtained from BAW-1803[100] and the margins were calculated in accordance with the rules of Regulatory Guide 1.99, Revision 2.

Adjusted RT_{NDT}

In accordance with the rules of Regulatory Guide 1.99, Revision 2, the estimates of end-of-life adjusted RT_{NDT} were obtained by taking the sum of the initial RT_{NDT}, the delta RT_{NDT}, and the margin. The delta RT_{NDT} is calculated by multiplying the fluence factor (a function of the fluence) by the chemistry factor (a function of the copper and nickel contents).

Atypical Weld

In the instance of the atypical weld, the adjusted RT_{NDT} was determined in accordance with an NRC recommendation which is presented in BAW-10144A.

Upper-Shelf Energy

The estimated end-of-life quarter-thickness upper-shelf energy was calculated in accordance with the rules of Regulatory Guide 1.99, Revision 2, and the method of BAW-1803. Initial upper-shelf energy is estimated in accordance with BAW-1803.

Table F-1. Reactor Vessel Weld End-of-Life (32 EFPY) Fracture Toughness Evaluation Data Summary for Oconee Unit 1[a]

Weld Number	Weld Location	Chemical Composition (wt%)[c]		Neutron Fluence (n/cm^2)		Initial Reference Toughness Temperature		Adjusted EOL RT_{NDT} (°F)[b]		Initial USE[d]	Estimated T/4 EOL USE (ft-lb)	
		Cu	Ni	Inside Surface	T/4	RT_{NDT} (°F)[d]	Margin (°F)[b]	Inside Surface	T/4	USE (ft-lb)	RG 1.99, R2	BAW-1803
SA-1135	NB/IS	0.25	0.54	1.6E18	9.6E17	-6	68	149	130	70	53	56
SA-1229	IS/US (Inside 61%)	0.26	0.61	7.8E18	4.7E18	-6	68	231	205	70	46	54
WF-25	IS/US (Outside 39%)	0.35	0.68	--	--	--	--	---	---	--	--	--
SA-1585	US/LS	0.21	0.59	1.0E19	6.0E18	-6	68	224	201	70	49	54
WF-9	LS/Dutchman	0.21	0.59	5.7E16	3.4E16	-6	68	---	---	70	--	--
SA-1073	IS-Longitudinal (both)	0.21	0.64	6.1E18	3.7E18	-6	68	209	185	70	51	63
SA-1493	US-Longitudinal (both)	0.20	0.55	7.4E18	4.4E18	-6	68	201	180	70	51	62
SA-1430	LS-Longitudinal (both)	0.20	0.55	9.1E18	5.5E18	-6	68	210	188	70	50	63
SA-1426	LS-Longitudinal	0.20	0.55	9.1E18	5.5E18	-6	68	210	188	70	50	63

[a] BAW-2050, October 1988.[24]

[b] Regulatory Guide 1.99, Revision 1, May 1988.[111]

[c] BAW-1820, December 1984[22] and BAW-1799, July 1983[112], and BAW-1500, Revision 1.[113]

[d] BAW-1803, Revision 1.[100]

Table F-2. Reactor Vessel Weld End-of-Life (32 EFPY) Fracture Toughness Evaluation Data Summary for Oconee Unit 2[a]

Weld Number	Weld Location	Chemical Composition (wt%)[c] Cu	Ni	Neutron Fluence (n/cm^2) Inside Surface	T/4	Initial Reference Toughness Temperature RT_{NDT} (°F)[d]	Margin (°F)[b]	Adjusted EOL RT_{NDT} (°F)[b] Inside Surface	T/4	Initial[d] USE (ft-lb)	Estimated T/4 EOL-USE (ft-lb) RG 1.99, R2	BAW-1803
WF-154	NB/US	0.31	0.59	7.3E18	4.4E18	-6	68	241	214	70	44	54
WF-25	US/LS	0.35	0.68	9.6E18	5.8E18	-6	68	283	251	70	43	54
WF-112	LS/Dutchman	0.31	0.59	5.4E16	3.2E16	-6	68	76	72	70	--	--

(a) BAW-2051, October 1988.[29]

(b) Regulatory Guide 1.99, Revision 2, May 1988.[111]

(c) BAW-1820, December 1984[22] and BAW-1799, July 1983.[112]

(d) BAW-1803, Revision 1.[100]

Table F-3. Reactor Vessel Weld End-of-Life (32 EFPY) Fracture Toughness Evaluation Data Summary for Oconee Unit 3[a]

Weld Number	Weld Location	Chemical Composition (wt%)[b]		Neutron Fluence (n/cm^2)		Initial Reference Toughness Temperature		Adjusted EOL RT_{NDT} (°F)[a]		Initial USE[c]	Estimated T/4 EOL-USE (ft-lb)	
		Cu	Ni	Inside Surface	T/4	RT_{NDT} (°F)[c]	Margin (°F)[a]	Inside Surface	T/4	(ft-lb)	RG 1.99, R2	BAW-1803
WF-200	NB/US	0.24	0.63	1.29E19	7.1E18	-6	68	249	223	70	45	57
WF-67	US/LS (Inside 75%)	0.24	0.60	1.6E19	9.4E18	-6	68	256	232	70	44	52
WF-70	US/LS (Outside 25%)	0.35	0.59	--	--	--	--	---	---	--	--	--
WF-169-1	LS/Dutchman	0.18	0.63	8.7E16	5.2E16	--	--	---	---	--	--	--

(a) Regulatory Guide 1.99, Revision 2, May 1988.[111]

(b) BAW-1820, December 1984[22] and BAW-1500, September 1978.[96]

(c) BAW-1803, Revision 1.[100]

NOTE: Values presented in this chart are extremely conservative because of high fluences. Fluences will be recalculated at next OC-3 capsule withdrawal and are expected to be reduced by 30%.

Table F-4. Reactor Vessel Weld End-of-Life (32 EFPY) Fracture Toughness Evaluation Data Summary for TMI Unit 1[a]

Weld Number	Weld Location	Chemical Composition (wt%)[c]		Neutron Fluence (n/cm^2)		Initial Reference Toughness Temperature		Adjusted EOL RT_{NDT} (°F)[b]		Initial USE[d] (ft-lb)	Estimated T/4 EOL-USE (ft-lb)	
		Cu	Ni	Inside Surface	T/4	RT_{NDT} (°F)[d]	Margin (°F)[b]	Inside Surface	T/4		RG 1.99, R2	BAW-1803
WF-70	NB/US	0.35	0.59	6.1E18	3.7E18	-6	68	244	214	70	44	53
WF-25	US/LS	0.35	0.68	8.0E18	4.8E18	-6	68	272	240	70	43	54
WF-70	LS/Dutchman (Outside 50%)	0.35	0.59	--	--	--	--	---	---	--	--	--
WF-67	LS/Dutchman (Inside 50%)	0.24	0.60	4.5E16	2.7E16	-6	68	73	69	70	--	--
WF-8	US-Longitudinal (both)	0.20	0.55	8.0E18	4.8E18	-6	68	205	183	70	50	58
SA-1494	LS-Longitudinal (Outside 63%)	0.18	0.63	--	--	--	--	---	---	--	--	--
SA-1526	LS-Longitudinal (Inside 37%)	0.35	0.68	6.7E18	4.0E18	-6	68	260	229	70	44	55
Atypical	US/LS	0.41	0.10	6.1E18	3.7E18	90[e]	28[e]	224	207	79	50	72

(a) BAW-1901, March 1986.[32]

(b) Regulatory Guide 1.99, Revision 2, May 1988.[111]

(c) BAW-1820, December 1984[22] and BAW-1799, July 1983,[112] and BAW-1500, Revision 1.[113]

(d) BAW-1803, Revision 1.[100]

(e) BAW-10144A, February 1980.[101]

Table F-5. Reactor Vessel Weld End-of-Life (32 EFPY) Fracture Toughness Evaluation Data Summary for Crystal River Unit 3[a]

Weld Number	Weld Location	Chemical Composition (wt%)[c] Cu	Ni	Neutron Fluence (n/cm²) Inside Surface	T/4	Initial Reference Toughness Temperature RT_{NDT} (°F)[d]	Margin (°F)[b]	Adjusted EOL RT_{NDT} (°F)[b] Inside Surface	T/4	Initial[d] USE (ft-lb)	Estimated T/4 EOL-USE (ft-lb) RG 1.99, R2	BAW-1803
WF-169-1	NB/US (Outside 60%)	0.18	0.63	--	--	--	--	---	---	--	--	--
SA-1769	NB/US (Inside 40%)	0.26	0.61	8.1E18	4.9E18	-6	68	233	207	70	46	54
WF-70	US/LS	0.35	0.59	1.1E19	6.6E18	-6	68	278	248	70	42	53
WF-154	LS/Dutchman	0.31	0.59	5.9E16	3.5E16	-6	68	77	72	70	--	--
WF-8	US-Longitudinal	0.20	0.55	1.0E19	6.0E18	-6	68	214	192	70	49	58
WF-18	US-Longitudinal	0.20	0.55	1.0E19	6.0E18	-6	68	214	192	70	49	66
SA-1580	LS-Longitudinal (both)	0.20	0.55	8.9E18	5.3E18	-6	68	209	188	70	50	63
Atypical	US/LS	0.41	0.10	1.1E19	6.6E18	90[e]	28[e]	244	227	79	47	71

(a) BAW-2049, September 1988.(39)

(b) Regulatory Guide 1.99, Revision 2, May 1988.(111)

(c) BAW-1820, December 1984(22) and BAW-1500, September 1978,(96) and BAW-1500, Revision 1.(113)

(d) BAW-1803, Revision 1.(100)

(e) BAW-10144A, February 1980.(101)

Table F-5. Reactor Vessel Weld End-of-Life (32 EFPY) Fracture Toughness Evaluation Data Summary for ANO Unit 1[a]

Weld Number	Weld Location	Chemical Composition (wt%)[c]		Neutron Fluence (n/cm^2)		Initial Reference Toughness Temperature		Adjusted EOL RT_{NDT} (°F)[b]		Initial USE[d] (ft-lb)	Estimated 1/4 EOL-USE (ft-lb)	
		Cu	Ni	Inside Surface	1/4	RT_{NDT} (°F)[d]	Margin (°F)[b]	Inside Surface	1/4		RG 1.99, R2	BAW-1803
WF-182-1	NB/US	0.24	0.63	7.4E18	4.4E18	-6	68	225	209	70	48	59
WF-112	US/LS	0.31	0.59	9.8E18	5.9E18	-6	68	258	230	70	43	53
WF-18	US-Longitudinal	0.20	0.55	7.2E18	4.3E18	-6	68	200	179	70	51	66
WF-18	LS-Longitudinal	0.20	0.55	7.6E18	4.6E18	-6	68	202	181	70	51	66
SA-1788	LS/Dutchman	0.25	0.54	5.5E16	3.3E16	--	--	---	---	--	--	--

(a) BAW-2075, March 198[illegible].[47]

(b) Regulatory Guide 1.99, Revision 2, May 1988.[111]

(c) BAW-1820, December 1984[22] and BAW-1799, July 1983[112] and BAW-1500, Revision 1.[113]

(d) BAW-1803, Revision 1.[100]

Table F-7. Reactor Vessel Weld End-of-Life (32 EFPY) Fracture Toughness Evaluation Data Summary for Rancho Seco Unit 1[a]

Weld Number	Weld Location	Chemical Composition (wt%)[c] Cu	Ni	Neutron Fluence (n/cm^2) Inside Surface	T/4	Initial Reference Toughness Temperature RT_{NDT} (°F)[d]	Margin (°F)[b]	Adjusted EOL RT_{NDT} (°F)[b] Inside Surface	T/4	Initial USE[d] (ft-lb)	Estimated T/4 EOL-USE (ft-lb) RG 1.99, R2	BAW-1803
WF-233	NB/US	0.29	0.68	6.7E18	4.0E18	-6	68	243	214	70	45	59
WF-154	US/LS	0.31	0.59	8.9E18	5.3E18	-6	68	252	224	70	43	53
WF-233	LS/Dutchman	0.29	0.68	5.0E16	3.0E16	--	--	---	---	--	--	--
WF-29	US-Longitudinal (both)	0.23	0.63	7.9E18	4.7E18	-6	68	225	200	70	48	56
WF-29	LS-Longitudinal (100%)	0.23	0.63	8.8E18	5.3E18	-6	68	230	205	70	48	56
WF-70	LS-Longitudinal (Inside 73%)	0.35	0.59	8.8E18	5.3E18	-6	68	265	235	70	43	53
WF-29	LS-Longitudinal (Outside 27%)	0.23	0.63	8.8E18	5.3E18	-6	68	230	205	70	48	56
Atypical	LS-Longitudinal (inside 73%)	0.41	0.10	8.8E18	5.3E18	90[e]	28[e]	237	219	79	48	72

(a) BAW-2074, March 1989.[48]

(b) Regulatory Guide 1.99, Revision 2, May 1988.[111]

(c) BAW-1820, December 1984[22] and BAW-1500, September 1978.[96]

(d) BAW-1803, Revision 1.[100]

(e) BAW-10144A, February 1980.[101]

Table F-8. Reactor Vessel Weld End-of-Life (32 EFPY) Fracture Toughness Evaluation Data Summary for Davis-Besse Unit 1[d]

Weld Number	Weld Location	Chemical Composition (wt%)[b] Cu	Ni	Neutron Fluence (n/cm^2)[d] Inside Surface	T/4	Initial Reference Toughness Temperature RT_{NDT} (°F)[c]	Margin (°F)[a]	Adjusted EOL RT_{NDT} (°F)[a] Inside Surface	T/4	Initial[c] USE (ft-lb)	Estimated T/4 EOL-USE (ft-lb) RG 1.99, R2	BAW-1803
WF-232	NB/US (Inside 9%)	0.14	0.69	1.8E18	--	-6	68	143	---	70	57*	36*
WF-233	NS/US (Outside 91%)	0.29	0.68	--	1.1E18	-6	68	---	150	70	52	60
WF-182-1	US/LS	0.24	0.63	1.1E19	6.6E18	-6	68	245	219	81[b]	53	68
WF-232	LS/Dutchman (Inside 12%)	0.14	0.69	6.2E16	--	--	--	---	---	--	--	--
WF-233	LS/Dutchman (Outside 88%)	0.29	0.68	--	3.70E16	--	--	---	---	--	--	--

(a) Regulatory Guide 1.99, Revision 2, May 1988.[111]

(b) BAW-1820, December 1984[22] and BAW-1500, September 1978.[96]

(c) BAW-1803, Revision 1.[100]

(d) B&W Document 32-1150627-01.[114]

*Inside surface

Table F-9. Reactor Vessel Weld End-of-Life (32 EFPY) Fracture Toughness Evaluation Data Summary for R. E. Ginna Unit 1[d]

Weld Number	Weld Location	Chemical Composition (wt%)[a] Cu	Ni	Neutron Fluence (n/cm^2)[c][d] Inside Surface	T/4	Initial Reference Toughness Temperature RT_{NDT} (°F)[e]	Margin (°F)[f]	Adjusted EOL RT_{NDT} (°F)[f] Inside Surface	T/4	Initial USE[e] (ft-lb)	Estimated T/4 EOL-USE (ft-lb) RG 1.99, R2	BAW-1803
SA-1101	NS/LS	0.26[b]	0.60[b]	4.5E18	3.0E18	10[b]	28[b]	194	172	55[b]	45	50
SA-847	IS/LS	0.25	0.54	4.1E19	2.8E19	-6	68	290	275	70	36	51
SA-1779	LS/Dutchman	0.25	0.54	1.6E16	--	-6	68	---	---	70	--	--

(a)BAW-1500, September 1978.[96]

(b)Refer to Turkey Point Unit 3, Table F-14.

(c)Axial flux ratio: WCAP-11026, December 1985.[102]

(d)Peak fluence: Private communication, W. S. Galloway, Jr., RG&E, to A. L. Lowe, Jr., August 3, 1988.[115]

(e)BAW-1803, Revision 1.[100]

(f)Regulatory Guide 1.99, Revision 2, May 1988.[111]

Table F-10. Reactor Vessel Weld End-of-Life (32 EFPY) Fracture Toughness Evaluation Data Summary for Point Beach Unit 1[a]

Weld Number	Weld Location	Chemical Composition (wt%)[a] Cu	Ni	Neutron Fluence (n/cm^2)[b,c] Inside Surface	1/4	Initial Reference Toughness Temperature RT_{NDT} (°F)[d]	Margin (°F)[e]	Adjusted EOL RT_{NDT} (°F)[e] Inside Surface	1/4	Initial USE[d] (ft-lb)	Estimated 1/4 EOL-USE (ft-lb) RG 1.99, R2	BAW-1803
SA-1426	NB/IS	0.20	0.55	2.7E18	1.8E18	-6	68	160	145	70	54	65
SA-812	IS-Longitudinal (Inside 27%)	0.17	0.52	1.8E19	1.2E19	-6	68	222	208	70	48	60
SA-775	IS-Longitudinal (Outside 73%)	0.19	0.63	--	--	--	--	---	---	--	--	
SA-1101	IS/LS	0.26[f]	0.60[f]	2.4E19	1.6E19	10[f]	28[f]	286	264	65[f]	36	48
SA-847	LS-Longitudinal	0.25	0.54	1.6E19	1.1E19	-6	68	251	233	70	42	52
SA-1101	LS/Dutchman	0.26[f]	0.60[f]	9.8E15	--	--	--	---	---	--	--	

(a)BAW-1500, September 1978,[96] and BAW-1500, Revision 1.[113]

(b)Axial and asymmetrical flux ratios: WCAP-10638, December 1984.[116]

(c)Peak fluence: Private communication, M. F. Moylan, WEPCO to C. J. Hudson, B&W, January 16, 1989.[117]

(d)BAW-1803, Revision 1.[100]

(e)Regulatory Guide 1.99, Revision 2, May 1988.[111]

(f)Refer to Turkey Point Unit 3, Table F-14.

Table F-11. Reactor Vessel Weld End-of-Life (32 EFPY) Fracture Toughness Evaluation Data Summary for Point Beach Unit 2[a]

Weld Number	Weld Location	Chemical Composition (wt%)[b]		Neutron Fluence (n/cm^2)[d][e]		Initial Reference Toughness Temperature		Adjusted EOL RT_{NDT} (°F)[g]		Initial[d] USE (ft-lb)	Estimated T/4 EOL-USE (ft-lb)	
		Cu	Ni	Inside Surface	T/4	RT_{NDT} (°F)[f]	Margin (°F)[g]	Inside Surface	T/4		RG 1.99, R2	BAW-1803
CE[a]	NB/IS	0.27[c]	0.90[c]	3.6E18	2.4E18	-56[c]	68	179	156	100[c]	70	--
SA-1484	IS/LS	0.24	0.60	2.6E19	1.8E19	-6	68	279	262	70	40	53
--	LS/Dutchman	--	--	1.0E16	--	--	--	---	---	---	--	--

[a] Weld fabricated at Combustion Engineering, Chattanooga, TN.

[b] BAW-1500, September 1978.[96]

[c] Estimated on basis of reactor vessels fabricated by Combustion Engineering at about the same time.

[d] Axial flux ratios: WCAP-10638, December 1984.[116]

[e] Peak fluence: Private communication, M. F. Moylan, WEPCO to C. J. Hudson, B&W, January 16, 1989.[117]

[f] BAW-1803, January 1983.[100]

[g] Regulatory Guide 1.99, Revision 2, May 1988.[111]

Table F-12. Reactor Vessel Weld End-of-Life (32 EFPY) Fracture Toughness Evaluation Data Summary for Surry Unit 1[a]

Weld Number	Weld Location	Chemical Composition (wt%)[b]		Neutron Fluence (n/cm^2)		Initial Reference Toughness Temperature		Adjusted EOL RT_{NDT} (°F)[g]		Initial USE[e] (ft-lb)	Estimated T/4 EOL-USE (ft-lb)	
		Cu	Ni	Inside Surface	T/4	RT_{NDT} (°F)[e]	Margin (°F)[g]	Inside Surface	T/4		RG 1.99, R2	BAW-1803
J726[a]	NB/IS	0.33[c]	0.10[c]	4.8E18	3.0E18	0[f]	69	190	171	90[c]	59	--
SA-1494	IS-Longitudinal (both)	0.18	0.63	6.4E18	4.0E18	-6	68	201	181	70	52	57
SA-1585	IS/LS (Inside 40%)	0.21	0.59	4.0E19	2.5E19	-6	68	282	265	70	40	52
SA-1650	IS/LS (Outside 60%)	0.21	0.59	--	--	--	--	---	---	--	--	
SA-1494	LS-Longitudinal	0.18	0.63	6.4E18	4.0E18	-6	68	201	181	70	52	57
SA-1526	LS-Longitudinal	0.35	0.68	6.4E18	4.0E18	-6	68	258	229	70	44	55

(a) Weld fabricated by De Rotterdamsche Drogdok, Rotterdam, Netherlands.

(b) BAW-1909, Revision 1, August 1986.[118]

(c) Estimated value.

(d) WCAP-11415, February 1987.[74]

(e) BAW-1803, Revision 1.[100]

(f) SECY 82-465, November 23, 1982.[119]

(g) Regulatory Guide 1.99, Revision 2, May 1988.[111]

Table F-13. Reactor Vessel Weld End-of-Life (32 EFPY) Fracture Toughness Evaluation Data Summary for Surry Unit 2[a]

Weld Number	Weld Location	Chemical Composition (wt%)[b]		Neutron Fluence (n/cm^2)[d]		Initial Reference Toughness Temperature		Adjusted EOL RT_{NDT} (°F)[g]		Initial USE[e] (ft-lb)	Estimated T/4 EOL-USE (ft-lb)	
		Cu	Ni	Inside Surface	T/4	RT_{NDT} (°F)[e]	Margin (°F)[g]	Inside Surface	T/4		RG 1.99, R2	BAW-1803
L737[a]	NB/IS	0.35[c]	0.10[c]	4.1E18	2.6E18	0[f]	69	190	170	90[c]	59	--
SA-1585	IS-Longitudinal	0.21	0.59	7.1E18	4.5E18	-6	68	209	188	70	50	54
SA-1585	IS-Longitudinal (Inside 50%)	0.21	0.59	7.1E18	4.5E18	-6	68	209	188	70	50	54
WF-4	IS-Longitudinal (Outside 50%)	0.20	0.55	--	--	--	--	---	---	--	--	--
R3008[a]	IS/LS	0.19	0.56	3.4E19	2.1E19	0[f]	69	268	251	90[c]	55	--
WF-4	LS-Longitudinal	0.20	0.55	7.1E18	4.5E18	-6	68	200	180	70	51	59
WF-4	LS-Longitudinal (Inside 63%)	0.20	0.55	7.1E18	4.5E18	-6	68	200	180	70	51	59
WF-8	LS-Longitudinal (Outside 37%)	0.20	0.55	--	--	--	--	---	---	--	--	--

[a] Weld fabricated by De Rotterdamische Drogdok, Rotterdam, Netherlands.

[b] BAW-1909, Revision 1, August 1986.[118]

[c] Estimated Value.

[d] WCAP-11499, June 1987.[76]

[e] BAW-1803, Revision 1.[100]

[f] SECY 82-465, November 23, 1982.[119]

[g] Regulatory Guide 1.99, Revision 2, May 1988.[111]

Table F-14. Reactor Vessel Weld End-of-Life (32 EFPY) Fracture Toughness Evaluation Data Summary for Turkey Point Unit 3[a]

Weld Number	Weld Location	Chemical Composition (wt%)[a]		Neutron Fluence (n/cm²)[b][c]		Initial Reference Toughness Temperature		Adjusted EOL RT_{NDT} (°F)[c]		Initial USE[d] (ft-lb)	Estimated T/4 EOL-USE (ft-lb)	
		Cu	Ni	Inside Surface	T/4	RT_{NDT} (°F)[d]	Margin (°F)[e]	Inside Surface	T/4		RG 1.99, R2	BAW-1803
SA-1484	NB/IS	0.24	0.60	3.3E18	2.1E18	-6	68	182	162	70	51	56
SA-1101	IS/LS	0.26[f]	0.60[f]	2.8E19	1.8E19	10[f]	28[f]	293	271	65[g]	35	48
SA-1135	LS/Dutchman	0.25	0.54	2.8E15	--	--	--	---	---	--	--	--

(a) BAW-1500, September 1978.[96]

(b) Axial flux ratios determined from Surry information (both plants being Westinghouse 3-loop PWRs): WCAP-11415, February, 1987.[74]

(c) Peak fluence: Private communication, R. S. Boggs, FP&L to C. J. Hudson, B&W, July 28, 1988.[120]

(d) BAW-1803, Revision 1.[100]

(e) Regulatory Guide 1.99, Revision 2, May 1988.[111]

(f) Safety Evaluation Report, memorandum, S. Varga to J. W. Williams, April 28, 1984.[106]

(g) WCAP-765[illegible].[59]

Table F-15. Reactor Vessel Weld End-of-Life (32 EFPY) Fracture Toughness Evaluation Data Summary for Turkey Point Unit 4[a]

Weld Number	Weld Location	Chemical Composition (wt%)[a]		Neutron Fluence (n/cm^2)[b][c]		Initial Reference Toughness Temperature		Adjusted EOL RT_{NDT} (°F)[e]		Initial USE[d] (ft-lb)	Estimated T/4 EOL-USE (ft-lb)	
		Cu	Ni	Inside Surface	T/4	RT_{NDT} (°F)[d]	Margin (°F)[e]	Inside Surface	T/4		RG 1.99, R2	BAW-1803
WF-70	NB/IS (Outside 33%)	0.35	0.59	--	--	--	--	---	---	--	--	--
WF-67	NB/IS (Inside 67%)	0.24	0.60	3.2E18	2.0E18	-6	68	181	161	70	51	54
SA-1101	IS/LS	0.26[f]	0.60[f]	2.7E19	1.7E19	10[f]	28[f]	292	268	66[g]	36	48
SA-1135	LS/Dutchman	0.25	0.54	2.7E15	--	--	--	---	---	--	--	--

[a] BAW-1500, September 1978.[96]

[b] Axial flux ratios determined from Surry information (both plants being Westinghouse 3-loop PWRs): WCAP-11415, February, 1987.[74]

[c] Peak fluence: Private communication, R. S. Boggs, FP&L to C. J. Hudson, B&W, July 28, 1988.[120]

[d] BAW-1803, Revision 1.[100]

[e] Regulatory Guide 1.99, Revision 2, May 1988.[111]

[f] Safety Evaluation Report, memorandum, S. Varga to J. W. Williams, April 26, 1984.[106]

[g] WCAP-7660.[60]

Table F-16. Reactor Vessel Weld End-of-Life (32 EFPY) Fracture Toughness Evaluation Data Summary for Zion Unit 1[a]

Weld Number	Weld Location	Chemical Composition (wt%)[a] Cu	Ni	Neutron Fluence (n/cm^2)[b] Inside Surface	T/4	Initial Reference Toughness Temperature RT_{NDT} $(^{\circ}F)$[c]	Margin $(^{\circ}F)$[e]	Adjusted EOL RT_{NDT} $(^{\circ}F)$[e] Inside Surface	T/4	Initial USE[c] (ft-lb)	Estimated T/4 EOL-USE (ft-lb) RG 1.99, R2	BAW-1803
WF-154	NB/IS (Inside 82%)	0.31	0.59	3.7E17	2.2E17	-6	68	111	98	70	56	57
SA-1769	NB/IS (Outside 18%)	0.26	0.61	--	--	--	--	---	---	--	--	--
WF-4	IS-Longitudinal	0.20	0.55	8.9E18	5.3E18	-6	68	209	188	70	50	59
WF-4	IS-Longitudinal (Outside 61%)	0.20	0.55	--	--	--	--	---	---	--	--	--
WF-8	IS-Longitudinal (Inside 39%)	0.20	0.55	8.9E18	5.3E18	-6	68	209	188	70	50	58
WF-70	IS/LS	0.35	0.59	1.7E19	1.0E19	-6	68	304	274	70	40	52
WF-8	LS-Longitudinal (both)	0.20	0.55	8.9E18	5.3E18	-6	68	209	188	70	50	58
Atypical	IS/LS	0.41	0.10	1.7E19	1.0E19	9[illegible][d]	28[d]	259	241	79	45	[illegible]

(a) BAW-1500, September 1978[96] and BAW-1500, Revision 1.[113]

(b) WCAP-10962, December 1985.[121]

(c) BAW-1803, Revision 1.[100]

(d) BAW-10144A, February 1980.[101]

(e) Regulatory Guide 1.99, Revision 2, May 1988.[111]

Table F-17. Reactor Vessel Weld End-of-Life (32 EFPY) Fracture Toughness Evaluation Data Summary for Zion Unit 2[a]

Weld Number	Weld Location	Chemical Composition (wt%)[a]		Neutron Fluence (n/cm^2)[b]		Initial Reference Toughness Temperature		Adjusted EOL RT_{NDT} (°F)[e]		Initial USE[c] (ft-lb)	Estimated T/4 EOL-USE (ft-lb)	
		Cu	Ni	Inside Surface	T/4	RT_{NDT} (°F)[c]	Margin (°F)[e]	Inside Surface	T/4		RG 1.99, R2	BAW-1803
WF-200	NB/IS	0.24	0.63	4.0E17	2.4E17	-6	68	108	96	70	58	62
WF-70	IS-Longitudinal*	0.35	0.59	8.4E18	5.0E18	-6	68	262	232	70	43	53
SA-1769	IS/LS	0.26	0.61	1.8E19	1.1E19	-6	68	273	248	70	41	53
WF-29	LS-Longitudinal*	0.23	0.63	8.4E18	5.0E18	-6	68	228	203	70	48	56
Atypical	IS-Longitudinal	0.41	0.10	8.4E18	5.0E18	90[d]	28[d]	289	263	79	49	72

(a) BAW-1500, September 1978.[96]

(b) WCAP-10[illegible] December 1985.[121]

(c) BAW-1803, Revision 1.[100]

(d) BAW-10144A, February 1980.[101]

(e) Regulatory Guide 1.99, Revision 2, May 1988.[111]

*WCAP-10962 shows WF-29 weld material for IS-longitudinal and WF-70 for LS-longitudinal. Review of production records from Mount Vernon plant shows that the welds are properly represented in this table.

Table F-18. Reactor Core Loading Schemes Used for Fluence Estimates

Plant Name	Fuel Management Scheme (Cycles Used)		
	Out-In	Low Leakage	Other
Oconee 1	1-5	6-EOL	---
Oconee 2	1-4	5-EOL	---
Oconee 3	1-5	6-EOL	
TMI-1	1-5	6-7	8-EOL (very low-leakage)
Crystal River 3	1-3	4-EOL	---
ANO-1	1-3	4-EOL	---
Rancho Seco	1-3	4-7	8-EOL (very low-leakage)
Davis-Besse	1-4	5-EOL	---
R. E. Ginna	1-12	13-EOL	---
Point Beach 1	1-7	8-16	17-EOL (low leakage plus modified peripheral assemblies)
Point Beach 2	1-5	6-15	16-EOL (low leakage plus modified peripheral assemblies)
Surry 1	1-7	8-EOL*	---
Surry 2	1-7	8-EOL*	---
Turkey Point 3	1-6	7-8	9-EOL (low leakage plus modified peripheral assemblies)
Turkey Point 4	1-4	5-8	9-EOL (low leakage plus modified peripheral assemblies)
Zion 1	1-6	7-EOL	---
Zion 2	1-5	6-EOL	---

*Includes assumption that core will be uprated from 2441 MWt to 2546 MWt at the beginning of cycle 11.

APPENDIX G

Reactor Vessel Neutron Dosimetry Information

G.1. Description of the Dosimetry Portion of the RVSP

The two principal objectives of the Master Integrated RVSP are (1) to determine the changes in the mechanical properties of RV steel resulting from long-term neutron irradiation, and (2) to monitor the vessel fluence. To accomplish the first objective, it is necessary to know, as accurately as practicable, the neutron fluence that the surveillance specimens were exposed to over their irradiation history. Specimen fluence is determined using a semi-empirical methodology which combines in-capsule dosimetry measurements with flux spectra that is generated by transport code calculations. The results of the fluence analysis are then integrated with the results of the mechanical testing to establish fluence-induced changes in the mechanical properties. The second objective, determination of the vessel fluence, is accomplished by analytical procedures, the results of which have been normalized to capsule fluence.

Vessel fluence must be monitored continuously throughout plant life. This will be accomplished differently in each of three distinct time periods:

> Initial period: Reactor startup to removal of last capsule in reactor vessel.
>
> Intermediate period: Removal of last capsule from the reactor vessel to installation of cavity dosimetry or end of plant life, whichever comes first.
>
> Final period: Installation of cavity dosimetry to the end of plant life. Note that some of the reactors will obtain their cavity dosimetry capability before completion of their RVSP. In such cases, it is possible that those reactors could skip the intermediate period altogether and move directly from in-vessel monitoring to ex-vessel monitoring.

G.1.1. Initial Period (Startup to Last Capsule Removal)

The reactor pressure vessel fluence is determined by the semi-empirical method described below.

The time weighted average power distribution is calculated and is used as input to the DOT-4 neutron transport code, which calculates the saturated activity for each dosimeter and the space dependent fast flux in the reactor vessel. The power history is used to adjust the saturated activities to account for those long half-life isotopes that do not reach saturation. The "final" calculated dosimeter activity is then computed and compared to the measured dosimeter activity. A normalization factor is then generated which is used to correct the DOT-calculated vessel fluence by the measured/calculated ratio at the capsule location.

The calculated dosimeter activities are usually within ±10% of the measured activities. The projected end of life fluence of the reactor pressure vessel is estimated using the future fuel cycle design power distributions and adjoint-mode neutron transport calculations to estimate future cycle fluences. The initial period fluences are considered to be very accurate, with uncertainties conservatively estimated to be in the 10-22% range. The projection to end-of-life fluence is more uncertain due to additional uncertainty associated with future cycle designs, and ranges from 20 to 35%.

G.1.2. Intermediate Period (Capsule Removal to Cavity Dosimetry Installation)

The vessel fluence for this period is analytically determined using the DOT-4 neutron transport code as described below.

The DOT code is used to calculate the reactor pressure vessel fluence for each vessel using the time weighted average power distribution and the power history of the reactor. The DOT-generated fluence is corrected by an experimental-to-calculated ratio which is defined and discussed in BAW 1485.

There is no reason to believe that the uncertainty associated with this method is any different than that estimated for the initial period. The analytical procedure is identical and the power distributions are handled in the same way.

G.1.3. Final Period (Cavity Dosimetry Installation to End of Plant Life)

The final period vessel fluence will be determined using the process described in detail in BAW-1875A[122] which differs from the methodology described in Paragraph G.1.1. in the following ways:

1. Activity measurements will be taken in the cavity rather than in vessel, using "state of the art" dosimetry.

2. Axial flux variations will be explicitly included in the analytical process.

3. Due to improved methodology and due to knowledge gained in the benchmark experiment, the uncertainty in final period calculations

is expected to be less than that estimated for the initial or intermediate periods. The numerical values for final period flux uncertainty will be determined after completion of the benchmark experiment presently underway at Davis-Besse Unit 1.

G.2. Summary

Discussed in Section G.1. there are two principal objectives of the MIRVP: (1) determine fluence induced changes in the mechanical properties of RV steel and (2) monitor the vessel fluence.

The first objective will be accomplished by the completion of each plant-specific RVSP. The second objective is accomplished by the dosimetry program described in Sections G.1.1., G.1.2., and G.1.3., which provides a knowledge of the vessel fluence within a determinable uncertainty range for the past, present, and future. The B&W 177-FA plants have determined that vessel fluences will be obtained through a cavity dosimetry program. The Westinghouse plants are considering the need to monitor vessel fluence through standby capsules; however, if sufficient capsules are not available, installation of cavity dosimetry by the Westinghouse plant owners must be considered. The accomplishment of these two objectives demonstrates that the dosimetry program is adequate and that the B&W Owners Group reactors are in compliance with 10CFR50, Appendix H.

APPENDIX H

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