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THE B&W OWNERS GROUP

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MASTER INTEGRATED REACTOR VESSEL SURVEILLANCE PROGRAM

B&W NUCLEAR SERVICE COMPANY

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MASTER INTEGRATED REACTOR VESSEL SURVEILLANCE PROGRAM

by

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Prepared for

B&W Owners Group Materials Committee
Reactor Vessel Working Group

Commonwealth Edison Company
Duke Power Company
Entergy Operations, Inc.
Florida Power Corporation
Florida Power & Light Company
GPU Nuclear Corporation
Rochester Gas & Electric Corporation
Toledo Edison Company
Virginia Power Company
Wisconsin Electric Power Company

(See section 5 for document signatures)

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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 16 reactor vessels fabricated by Babcock and Wilcox (B&W). These reactor vessels include 7 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, a number of irradiation sites, and, as a contingency, a test reactor irradiation program. As shown in Figure 1-1, this program includes integration of the following irradiation efforts:

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

b. 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 16 reactor vessels

d. existing test reactor irradiation data sources

e. additional test reactor irradiation data sources

The 10 utility participants in the B&WOG Reactor Vessel Integrity Program* have 16 reactor vessels with beltline regions that were fabricated using the automatic submerged arc welding process with 32 combinations of Mn-Mo-Ni filler wire heats and Linde 80 flux lots. 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 analysis to ensure continued compliance with 10CFR50,[2] Appendix G.

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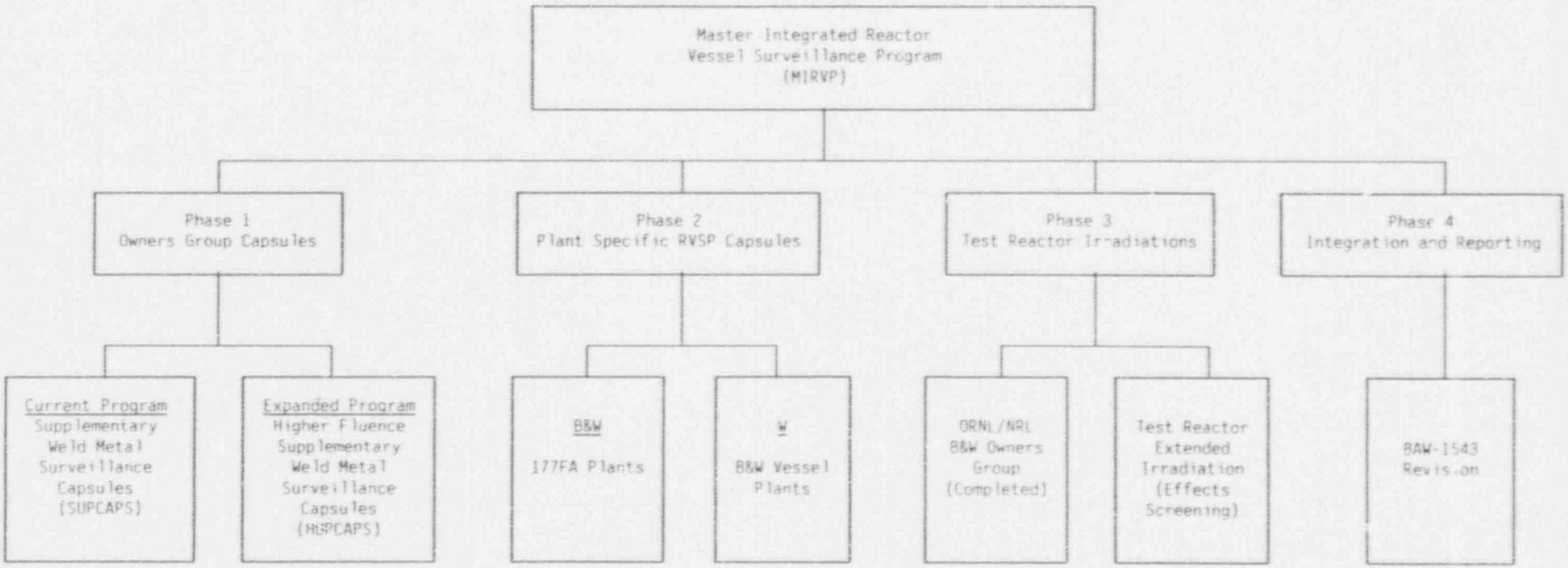
*Entergy Operations, Inc., Commonwealth Edison Company, Duke Power Company, Florida Power Corporation, Florida Power & Light Company, General Public Utilities Nuclear Corporation, Rochester Gas & Electric Corporation, Toledo Edison Company, Virginia Power Company, Wisconsin Electric Power Company.

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



The objectives of the MIRVP 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 or Final Safety Analysis Report.

This report is comprised of three sections: 1) the main report, 2) an appendix section, and 3) a supplement document (BAW-1543, Rev. 4, Supplement). The main report provides a detailed description of each element of the MIRVP and explains their interrelationships. It also 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. The supplement document provides irradiation capsule withdrawal schedules. Current and future plant operations such as extended and unexpected outages and flux reduction measures could cause changes to the withdrawal schedules in the future. The supplement will allow for faster publication of amendments to the schedules.

The supplement to BAW-1543, Rev. 4 is a new addition to the MIRVP documentation. It is one of two major changes between Rev. 3 and Rev. 4. The main report and appendices in Rev. 4 is identical to Rev. 3 except for the absence of withdrawal schedules, which are now contained in the supplement. The second major change between Rev. 3 and Rev. 4 is an updating of each plants' withdrawal schedule. This updating was necessary due to revised fluence calculations for the plants.

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 neutron irradiation damage to the reactor vessel materials could significantly degrade vessel 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 then current state-of-the-art technology for designing a surveillance program.

Research on neutron irradiation damage to reactor vessel steels continued in the 1960s. Specifically, the effects of the principal parameters influencing neutron embrittlement sensitivity were 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 1960s, 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

Table 2-1. Significant Differences Between Revisions of ASTM E 185

ASTM E 185 Revision	Materials Monitored by Program	No. of Capsules	No. of Specimens/ Capsule/Material	No. of Baseline Specimens/Mat's	Specimen Orientation
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 work direction
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 work direction
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 work direction
1979	General guidance for selection controlling materials 1. Controlling base metal 2. Controlling weld metal (same heat 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 work direction
1982	Same as above	5	Same as above	Same as above	Same as above

Table 2-1. Significant Differences Between Revisions of ASTM E 185 (Cont'd)

ASTM E 185 Revision	Index Temp for Measuring T	Capsule Withdrawal Schedule	Dosimetry Requirements	Temperature Monitor Requirements	Special Requirements and Recommendations
1966	Charpy energy fix temp as identified by NDT unirr. drop wt tests (norm. 30 ft-lb)	One at neutron fluence corresponding to EOL; others not specified	Refer to ASTM E 261; 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	Same as above	One corresponding to 30% of design life; one to 100% life; others not specified	Determined per ASTM E 261; 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 shallbe determined
1973 Case B	Measured at 30 ft-lb	First 3 capsules withdrawn at specific times; 4th & 5th capsules standby	Determined per ASTM E 261; Fe & unshielded Co dosimetry 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
1979	T @ 30 ft-lb RT_{NDT} T @ 5 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 E 482 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 & 3 3. Complete chemistry of test materials shall be determined 4. Add'l fracture toughness specimen per ASTM E 535 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	Same as above	Same as above	Same as above	1. Charpy index temps, upper shelf energy determined from average curves

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.

The requirements of 10CFR50, Appendixes G and H, together with ASTM Standard E 185-82, currently provide 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 margin of safety. Fracture mechanics techniques are used to quantitatively define plant operating conditions in terms of pressure-temperature (P-T) limits for heatup and cooldown. 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 locat.on on the crack front of the assumed flaw," and (4) extend Append'x 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 an 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.

(1) 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.

(2) There must be adequate arrangement for data sharing between plants.

(3) 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.

(4) 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 16 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, Appendices 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., end-of-life 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 n/cm^2 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 RVSPs 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-specific RVSPs.

3.2.1. Babcock and Wilcox-Designed Reactor Vessel Surveillance Programs

There are seven 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 seven 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/Entergy Operations, Inc.	Davis-Besse Unit 1/Toledo Edison Company
Three Mile Island Unit 1/GPU Nuclear Corporation	Crystal River Unit 3/Florida Power Corporation

The plant-specific RVSP for each of the seven 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
Davis-Besse Unit 1	BAW-10100A

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

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 $>5 \times 10^{18}$ n/cm^2; highest RT_{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 BAW 177-FA RVSPs

Plant Parameters	Davis-Besse 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	2722	2568	2544	2568	2568	2568	2568
Design overpower, %	112	112	112	112	112	112	112
System pressure (nom), psia	2200	2200	2200	2200	2200	2200	2200
Coolant flow rate, 10^6 lbm/h;	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	556	556	556	556	556	556	556
Avg rise in vessel	46	46	46	46	46	46	46
Avg in vessel	579	579	579	579	579	579	579
No. of fuel assemblies	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)
Core barrel ID/OD, in.	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
Core structural characteristics							
Core equiv diam, in.	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	142.3

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

Plant Parameters	Davis-Besse 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 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
Design temperature, °F	650	650	650	650	650	650	650
Shell ID, in.	171	171	171	171	171	171	171
Shell thickness, in.	8.44	8.44	8.44	8.44	8.44	8.44	8.44
OD across nozzles, in.	261[c]	249	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"
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

[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.

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 numerals) 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 and improve heat transfer. The capsule is helium-filled to improve heat transfer, protect the specimens from oxidation, and provide for leak testing.

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 9°F above the temperature of the entering coolant water. This is well within the ±25°F temperature criterion used for comparison to the 1/4-thickness vessel wall location.

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 opposing 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 ±25°F 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

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

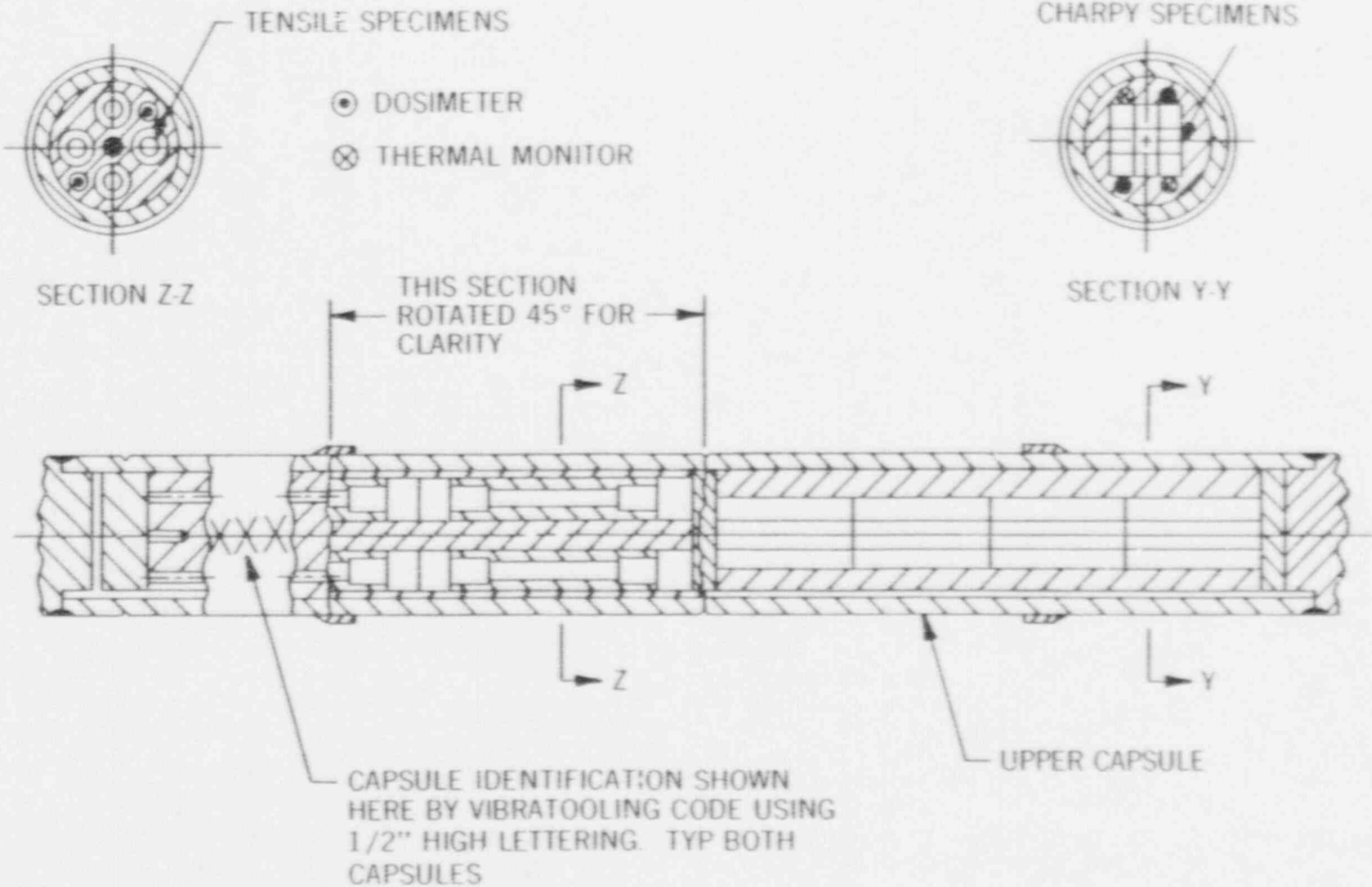


Figure 3-2 Surveillance Capsule Arrangement -- Type IV

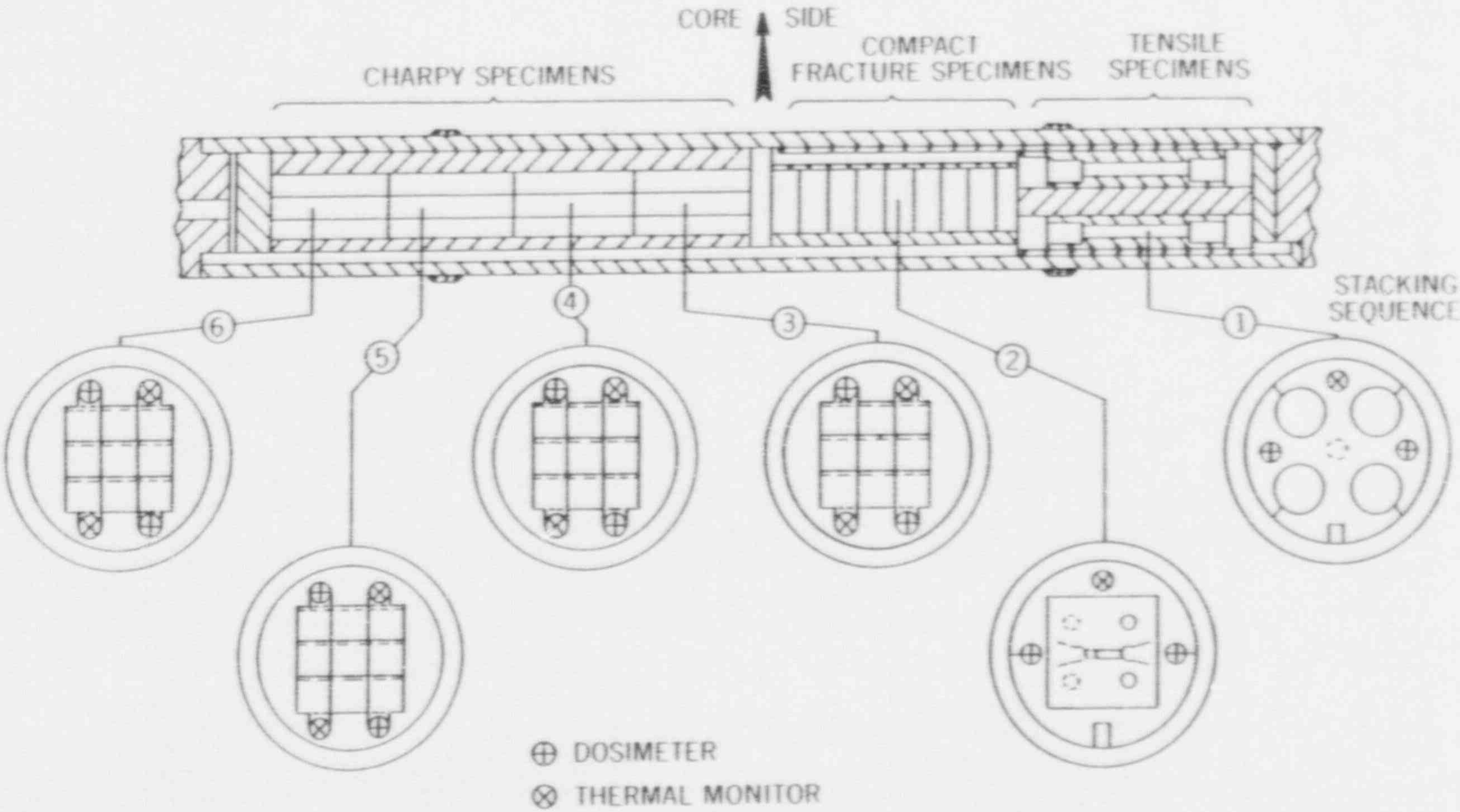


Figure 3-3 Surveillance Capsule Arrangement -- Types III, V, and VI

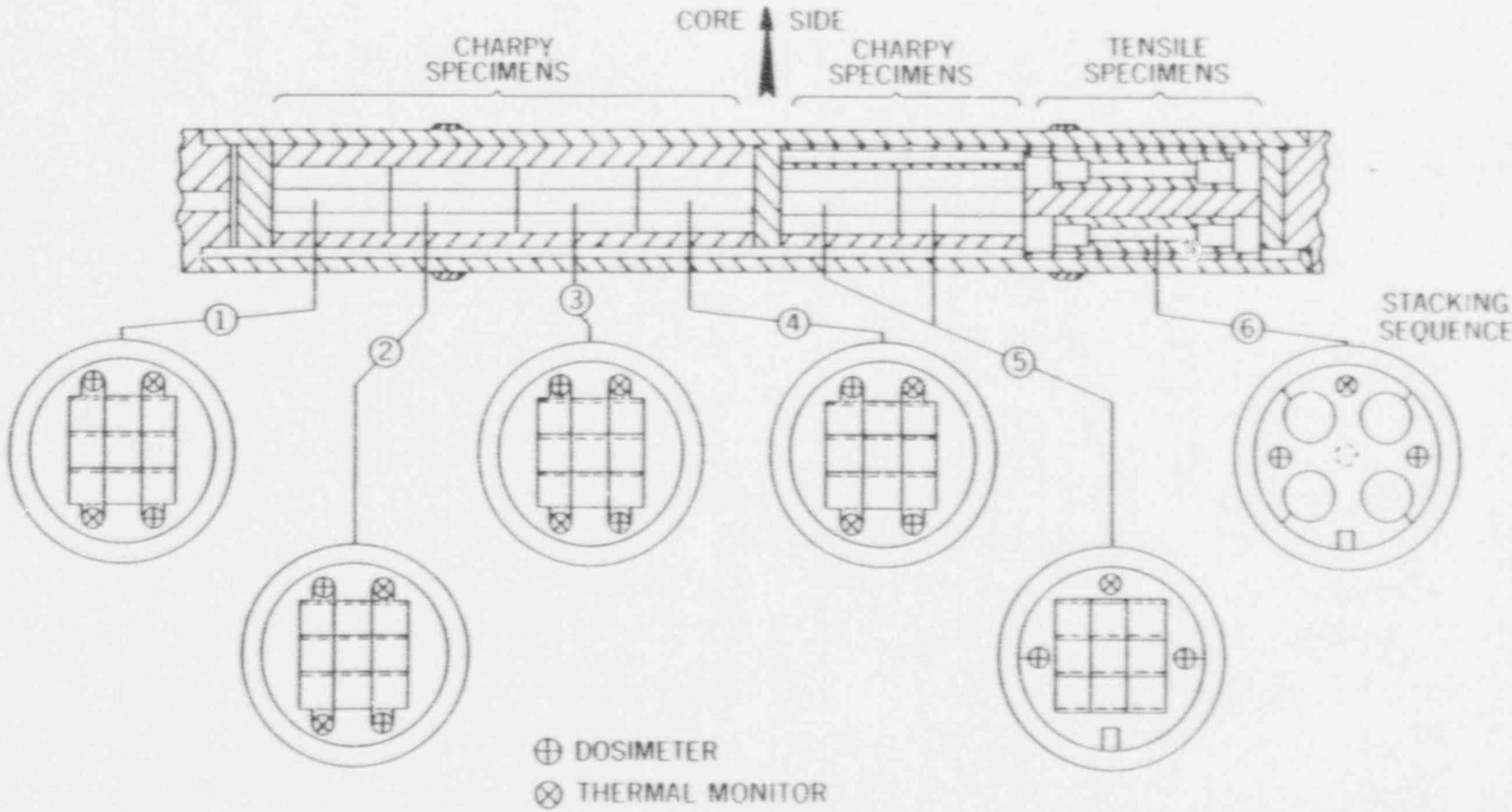


Figure 3-4 Reactor Vessel Arrangement Showing Current Surveillance Capsule Holder Tube Locations

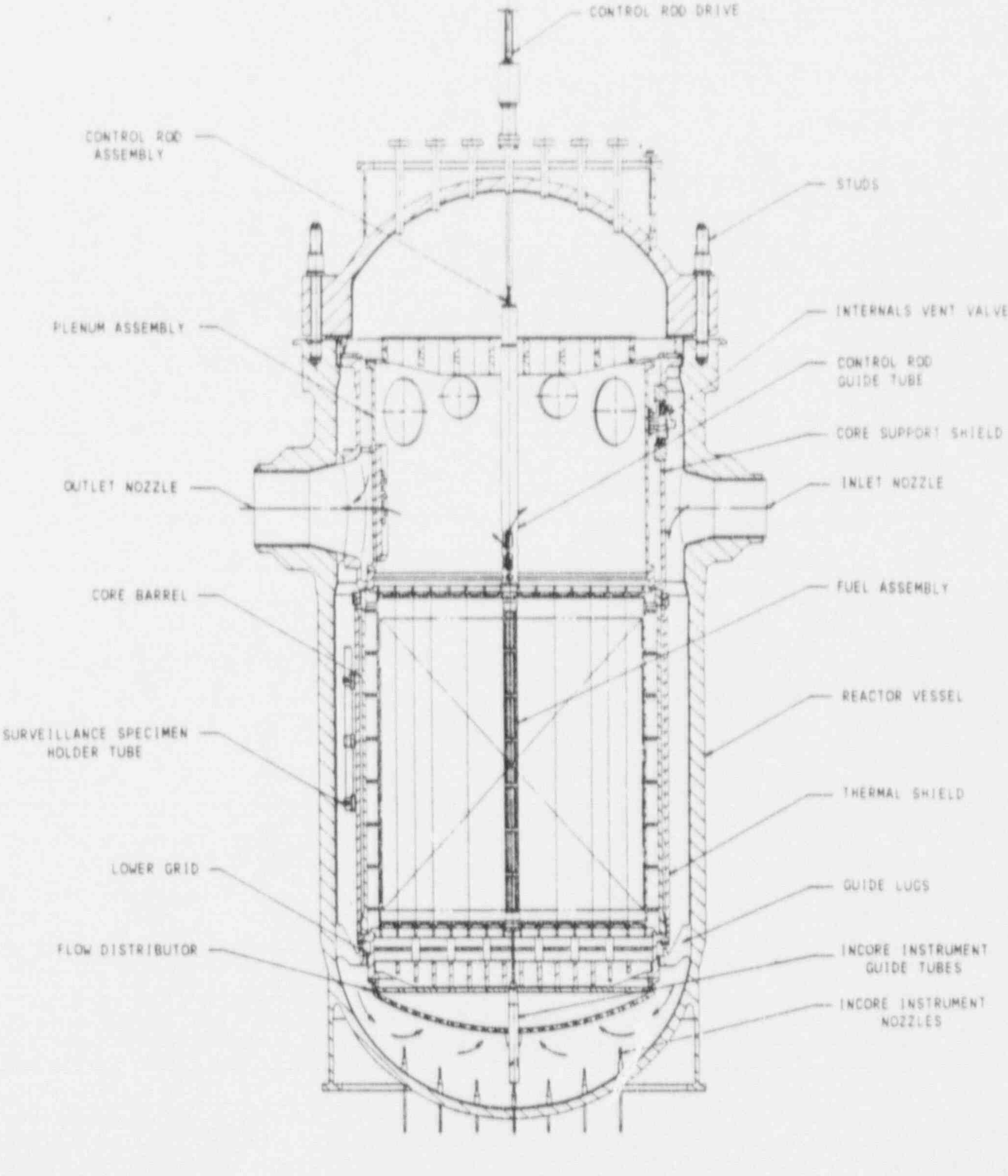
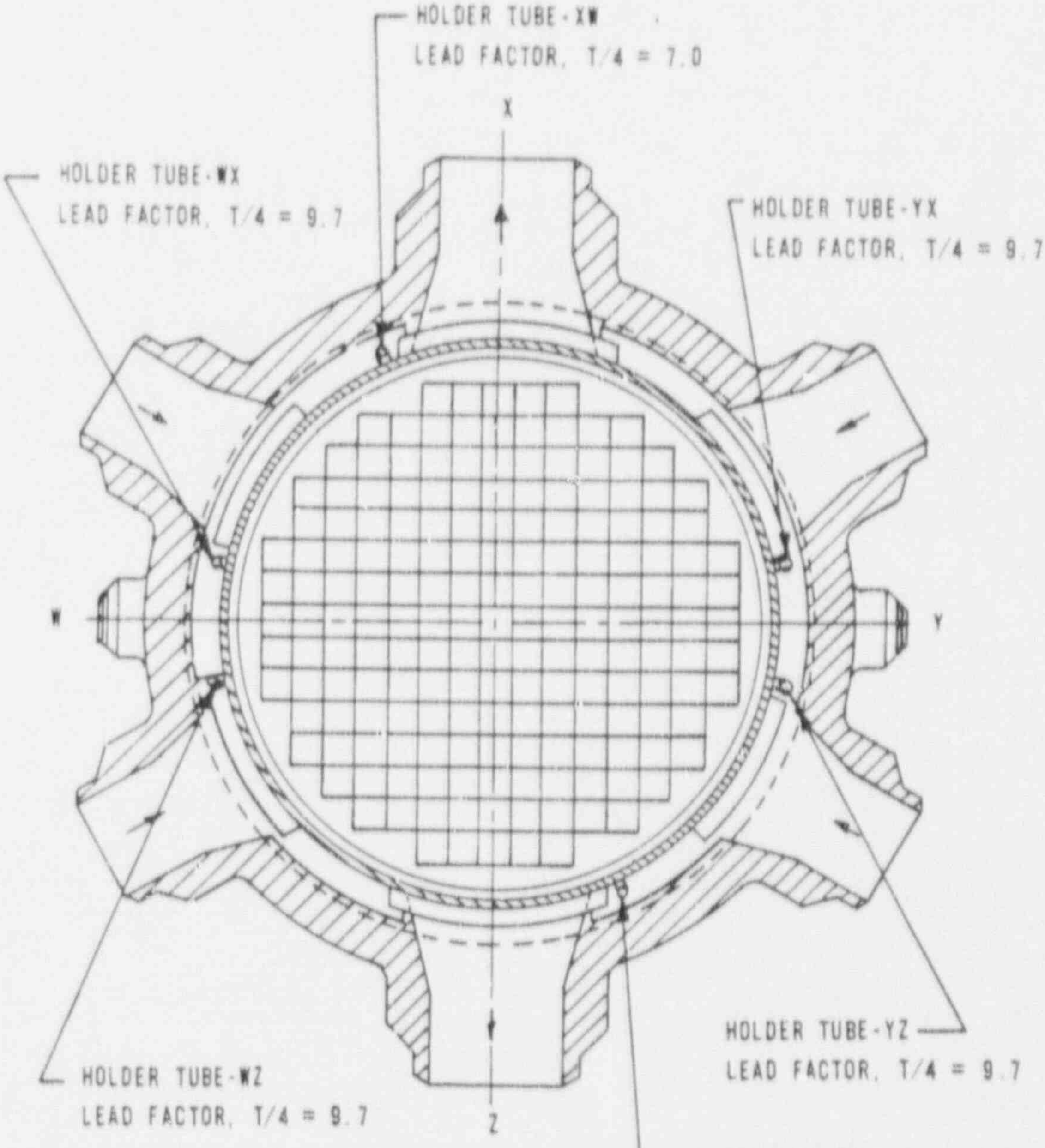
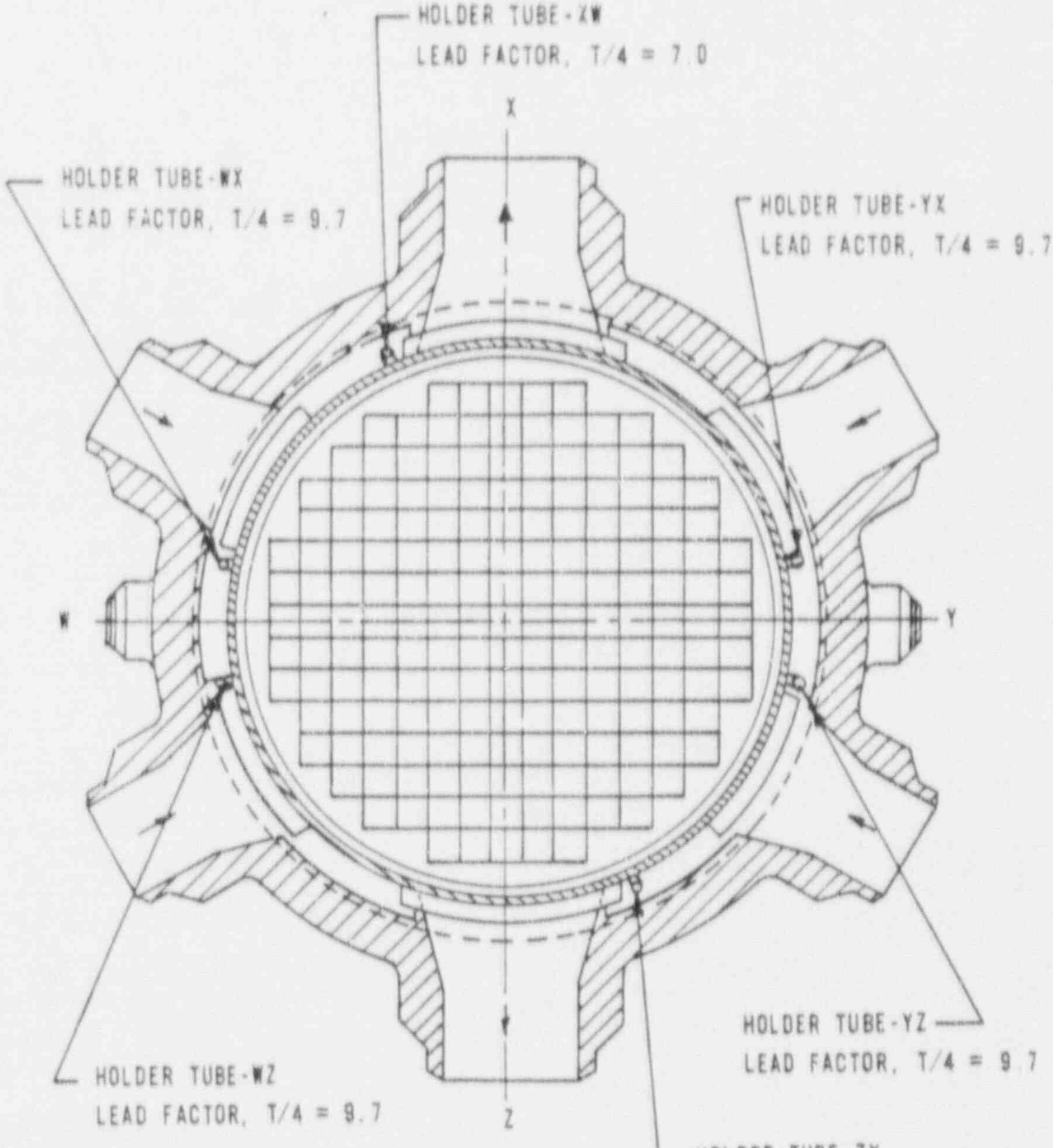


Figure 3-5 Surveillance Capsule Holder Tube Location and Identification for Crystal River Unit 3



Note: Lead factors are for original fuel management and capsule locations are for original placement.

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



Note: Lead factors are for original fuel management and capsule locations are for original placement.

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 25°F of the vessel 1/4T temperature. The coolant temperature serves as the lower bound and is also within 25°F 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.

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.

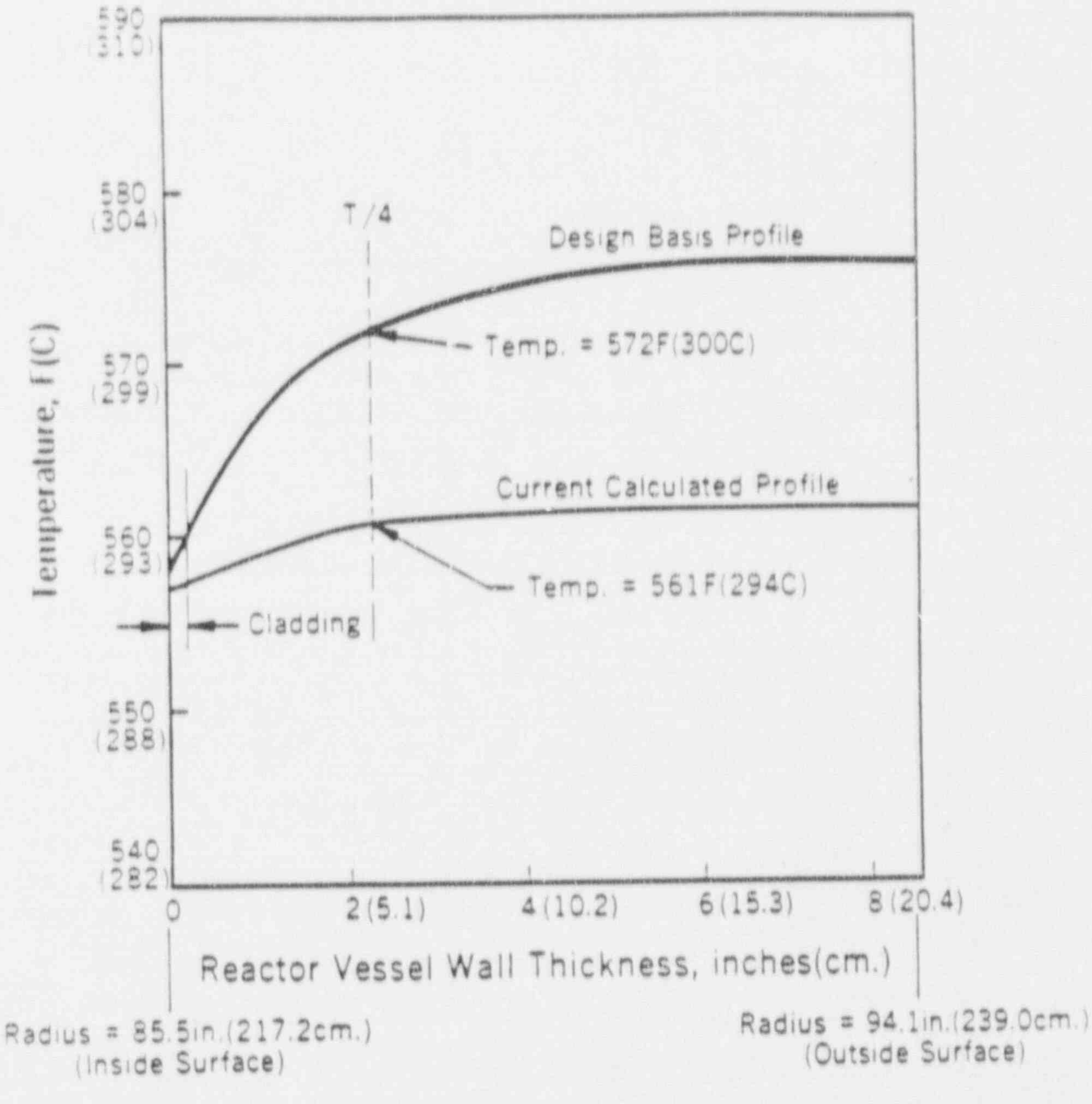
3.2.1.4. Types of Surveillance Programs and Capsules

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 1 of the BAW-1543, Rev. 4 Supplement. 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.

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



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.

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.

*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* 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.

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 3-5. 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.

*Wedge-Opening Loading fracture toughness specimen.

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:

Nuclear Plant/Owner	Applicable Report
Group 1 (2-Loop, 14 x 14 Fuel Array)	
R. E. Ginna/Rochester Gas & Electric Corp.	WCAP-7254[23]
Point Beach Unit 1/Wisconsin Electric Power Co.	WCAP-7513[24]
Point Beach Unit 2/Wisconsin Electric Power Co.	WCAP-7712[25]
Group 2 (3-Loop, 15 x 15 Fuel Array)	
Surry Unit 1/Virginia Power Co.	WCAP-7723[26]
Surry Unit 2/Virginia Power Co.	WCAP-8085[27]
Turkey Point Unit 3/Florida Power & Light Co.	WCAP-7656[28]
Turkey Point Unit 4/Florida Power & Light Co.	WCAP-7660[29]
Group 3 (4-Loop, 15 x 15 Fuel Array)	
Zion Unit 1/Commonwealth Edison Co.	WCAP-8064[30]
Zion Unit 2/Commonwealth Edison Co.	WCAP-8132[31]

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 during fabrication to optimize heat transfer.

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 capsule is approximately 63 inches in height. The capsules 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 E 185 (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 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 Figures 3-8 through 3-11. The types of dosimeters are given in Table 3-6. 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.

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 3-7. 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

Table 3-5. 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	2200	2241/2456 (uprated)	2241/2456 (uprated)	3520/3391 (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	110.7; 271,900	135; 364,500	135; 364,500
Coolant temperatures, F									
Nominal inlet	546	542	542	546	546	543	543	529.4	529.4
Avg rise in vessel	57	56	56	54	55.9	62.8	62.8	60	60
Avg in vessel	573.5	570	570	573	573	574.4	574.4	559.4	559.4
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	UFA	UFA
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	119.7	119.7	119.7	119.7	119.7	132.7	132.7
Core active fuel height, in.	141.4	144	144	14[illegible]	144	144	144	144	144

Table 3-5. 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
Reactor vessel design parameters									
Principal material	A-508 C12	SA-302 GrB	A-508 C12	A-508 C12	A-508 C12	A-533 GrB C11	A-533 GrB C11	A-533 GrB C11	A-533 GrB C11
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

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

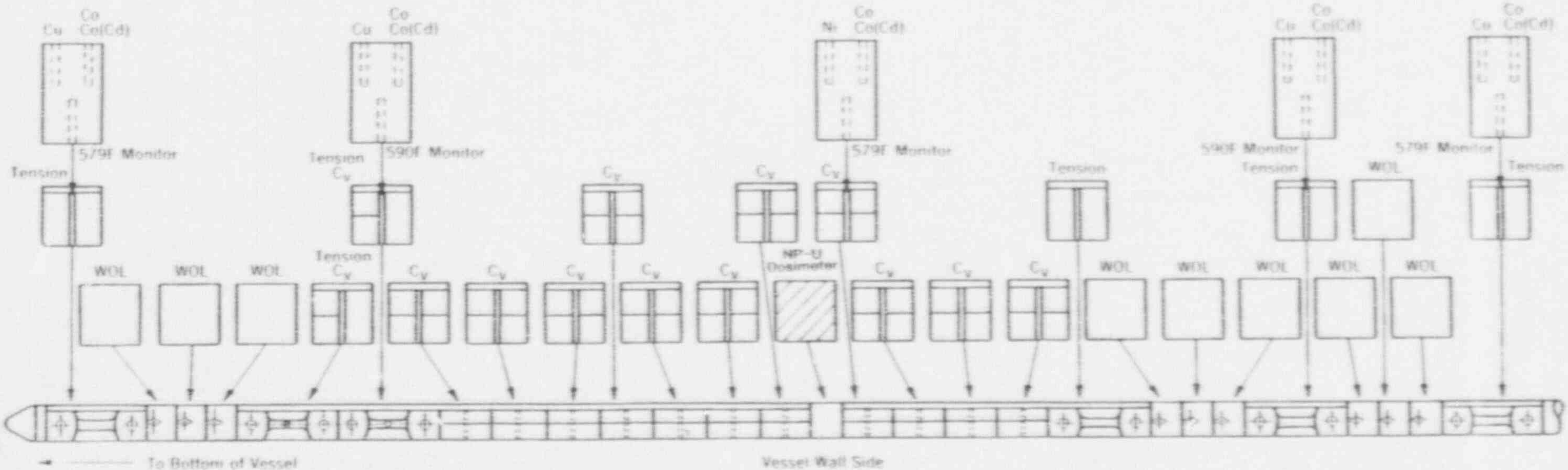


Figure 3-9 Schematic Showing Specimens, Thermal Monitors and Dosimeter Placement and Orientation with Respect to the Core and Vessel Wall for Westinghouse Capsule Type VI

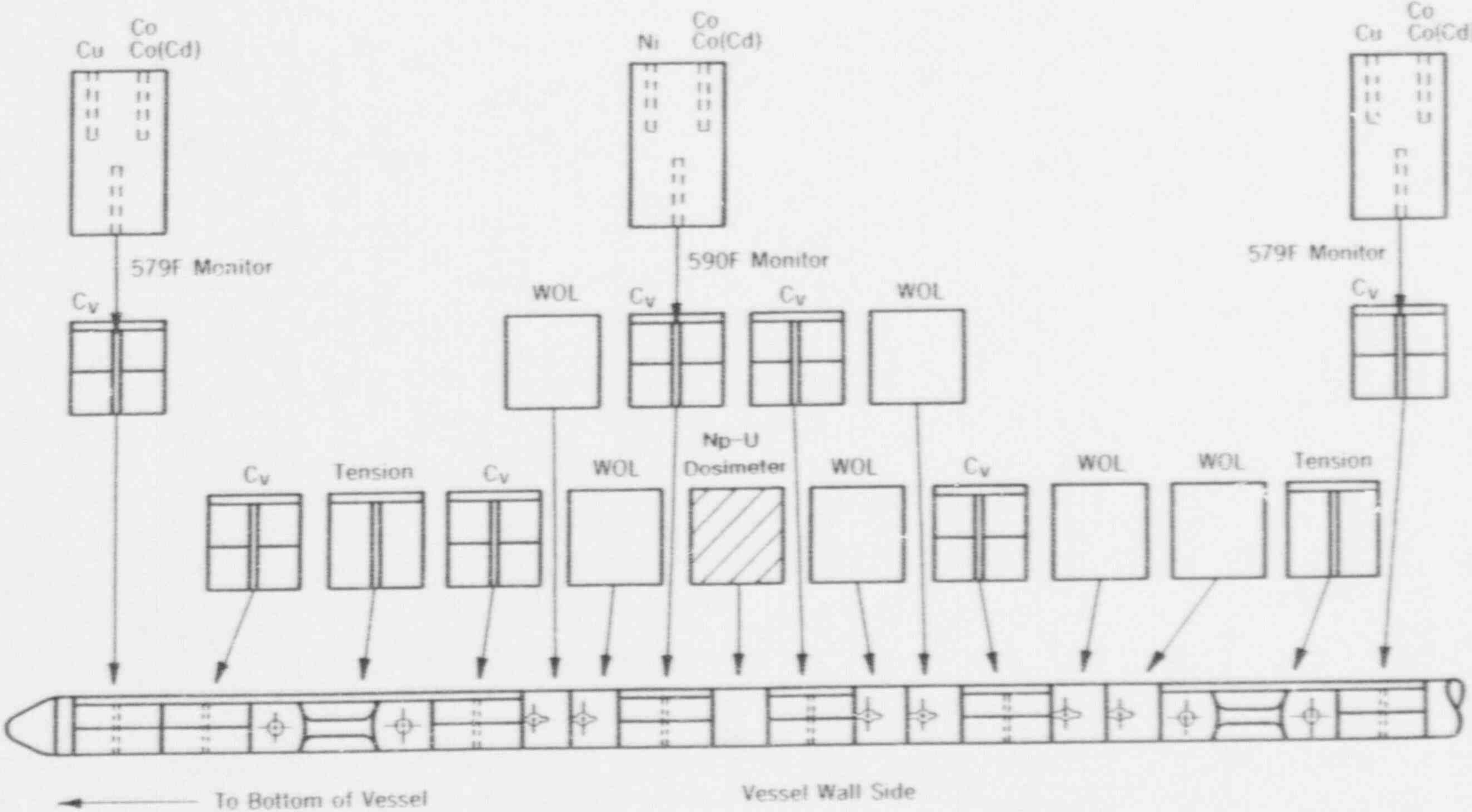


Figure 3-10 Schematic Showing Specimens, Thermal Monitors and Dosimeter Placement and Orientation with Respect to the Core and Vessel Wall for Westinghouse Capsule Type VII

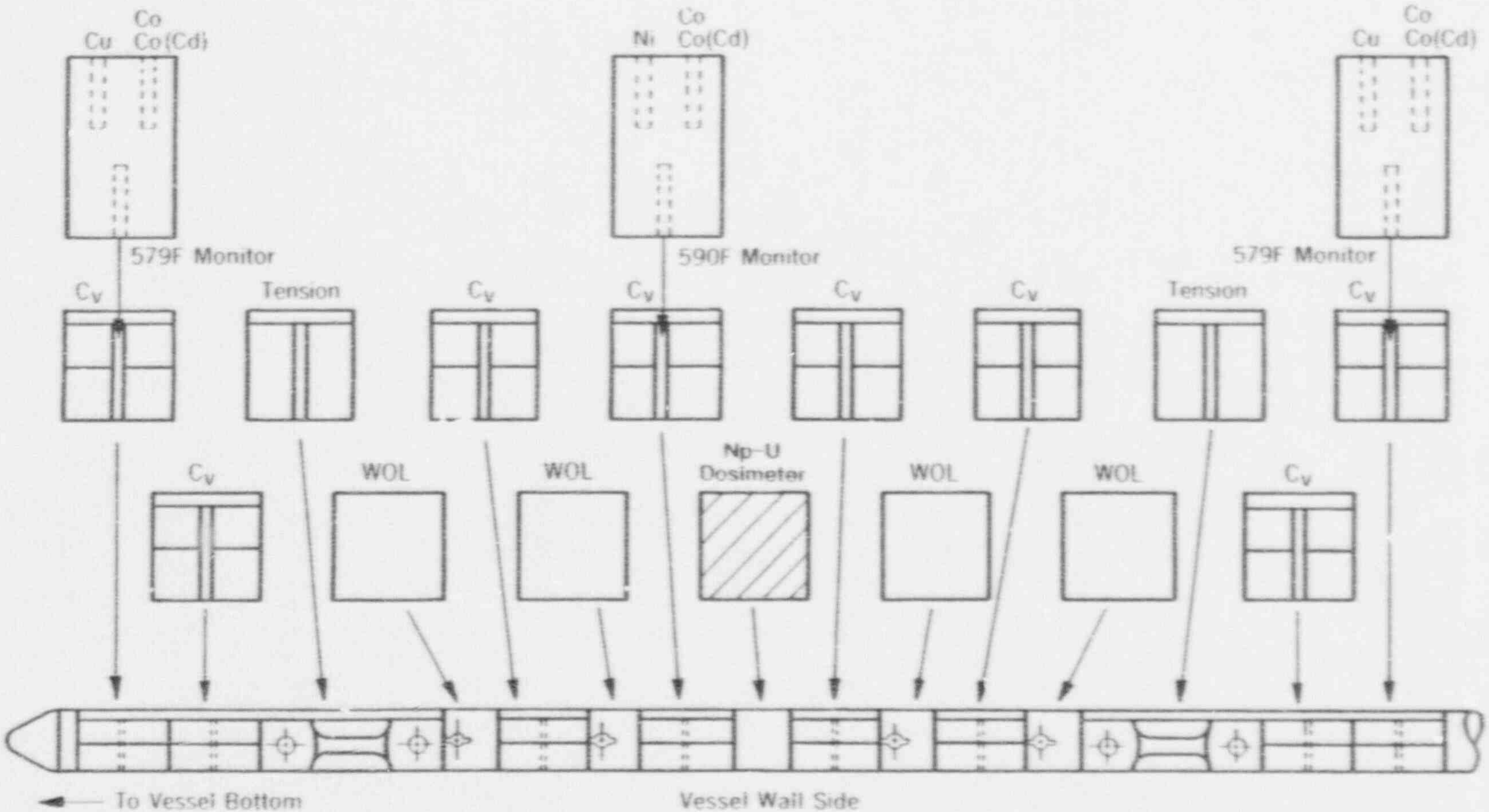


Figure 3-11 Schematic Showing Specimens, Thermal Monitors and Dosimeter Placement and Orientation with Respect to the Core and Vessel Wall for Westinghouse Capsule Types VIII and IX

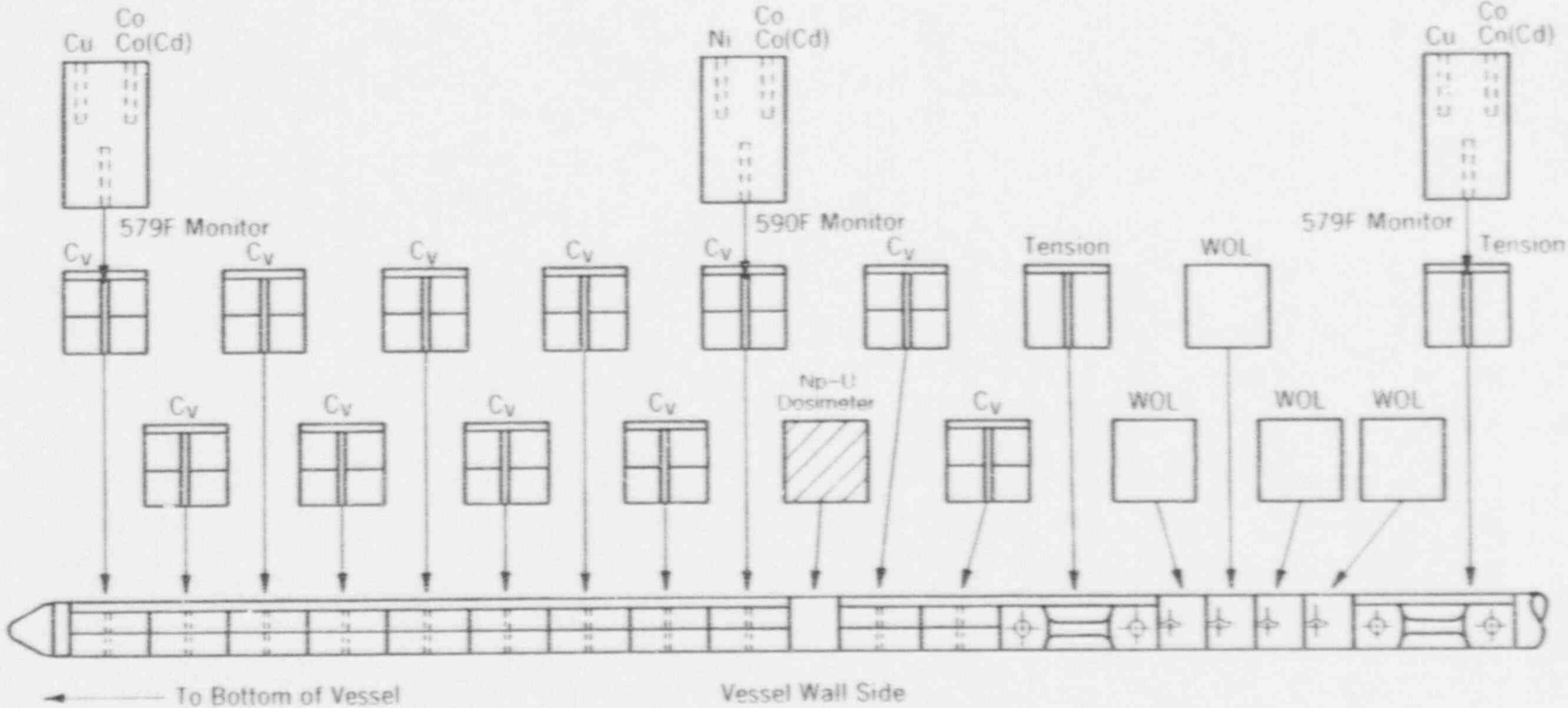
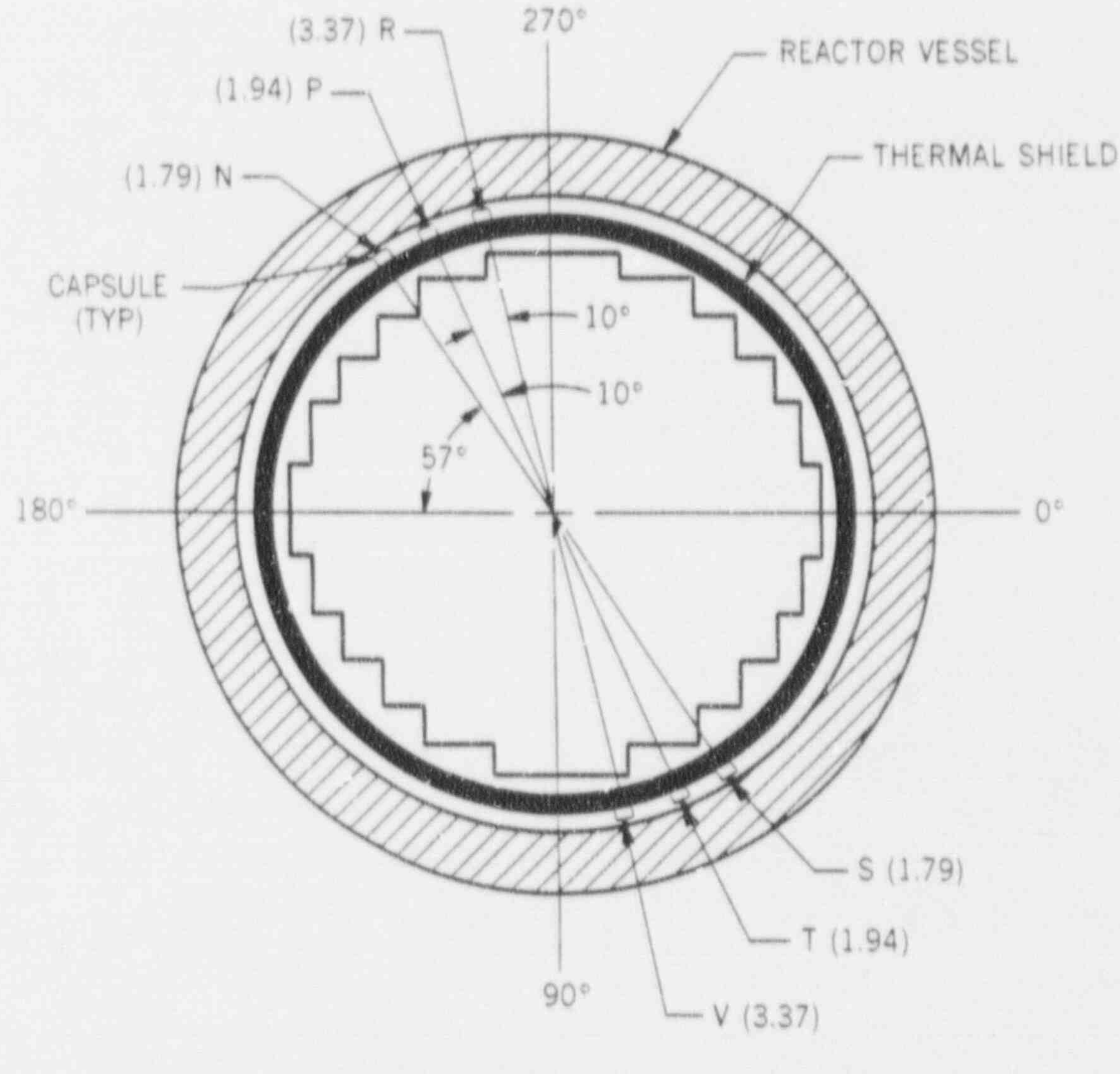
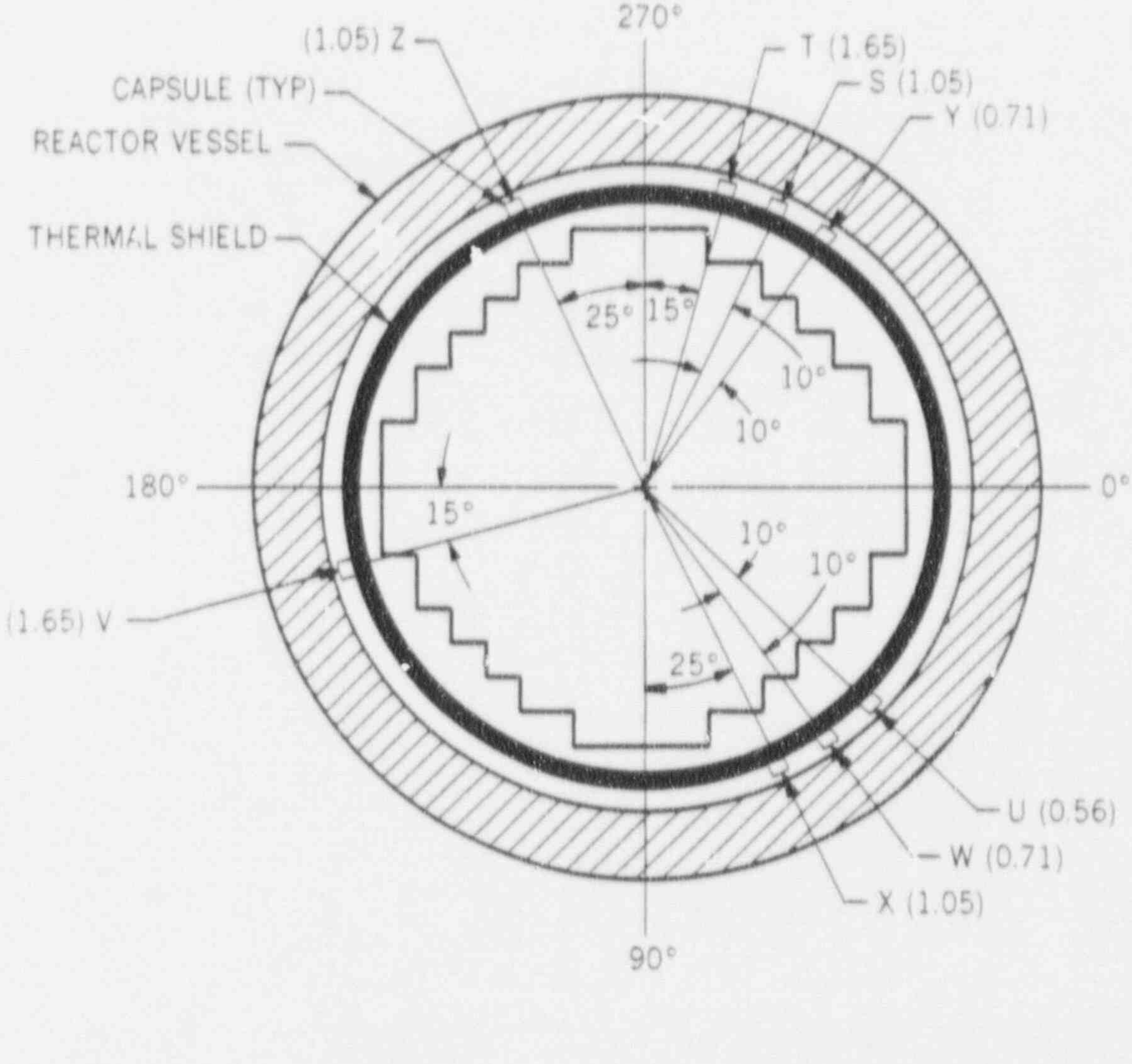


Figure 3-12 Arrangement of Surveillance Capsules in the R. E. Ginna Unit No. 1, and Point Beach Units No. 1 and 2 Reactor Vessels (Lead Factors for the Capsules Shown in Parentheses are for the Original Fuel Management)



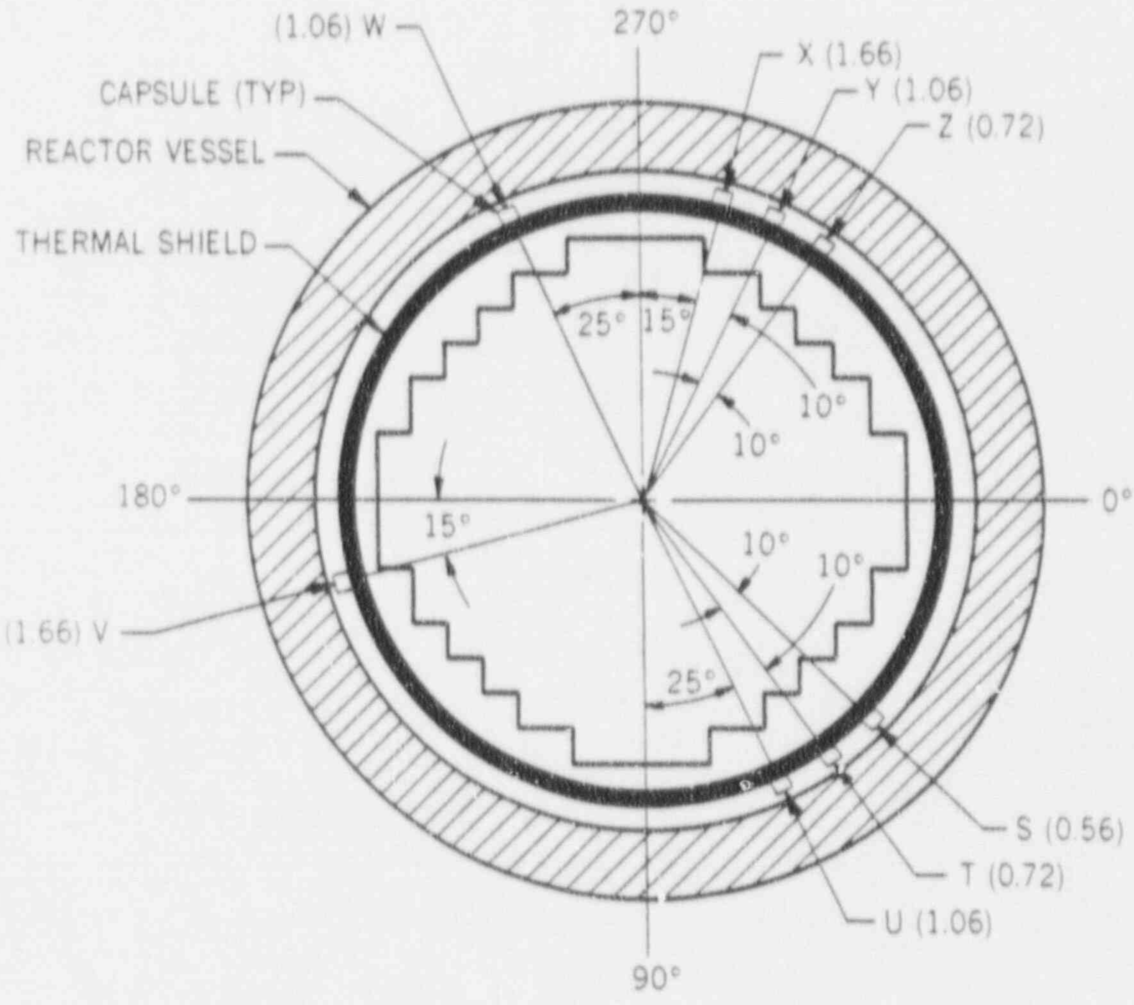
Note: Capsule lead factors shown in parentheses are for the original fuel management and capsule locations are for original placement.

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)



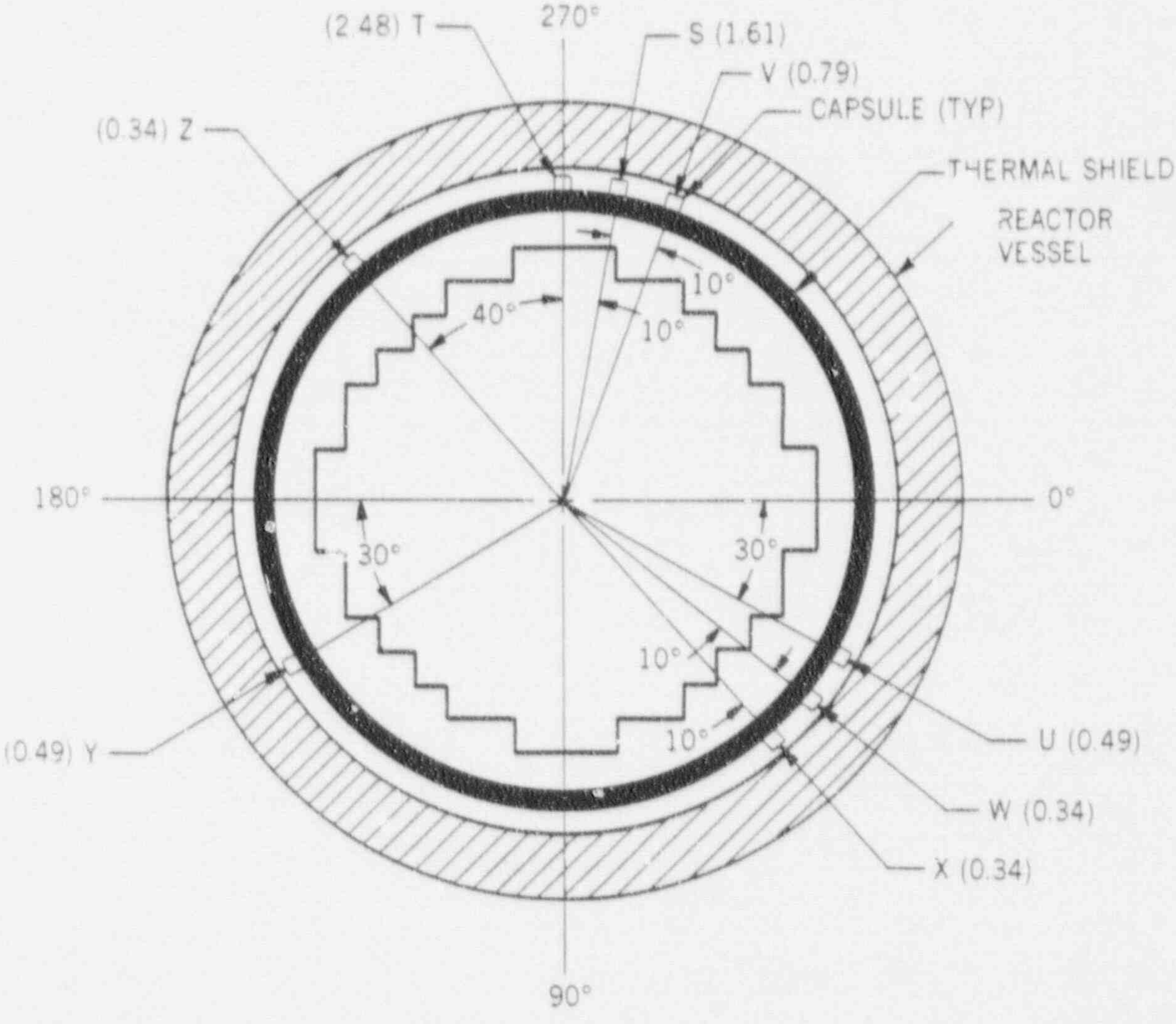
Note: Capsule lead factors shown in parentheses are for the original fuel management and capsule locations are for original placement.

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)



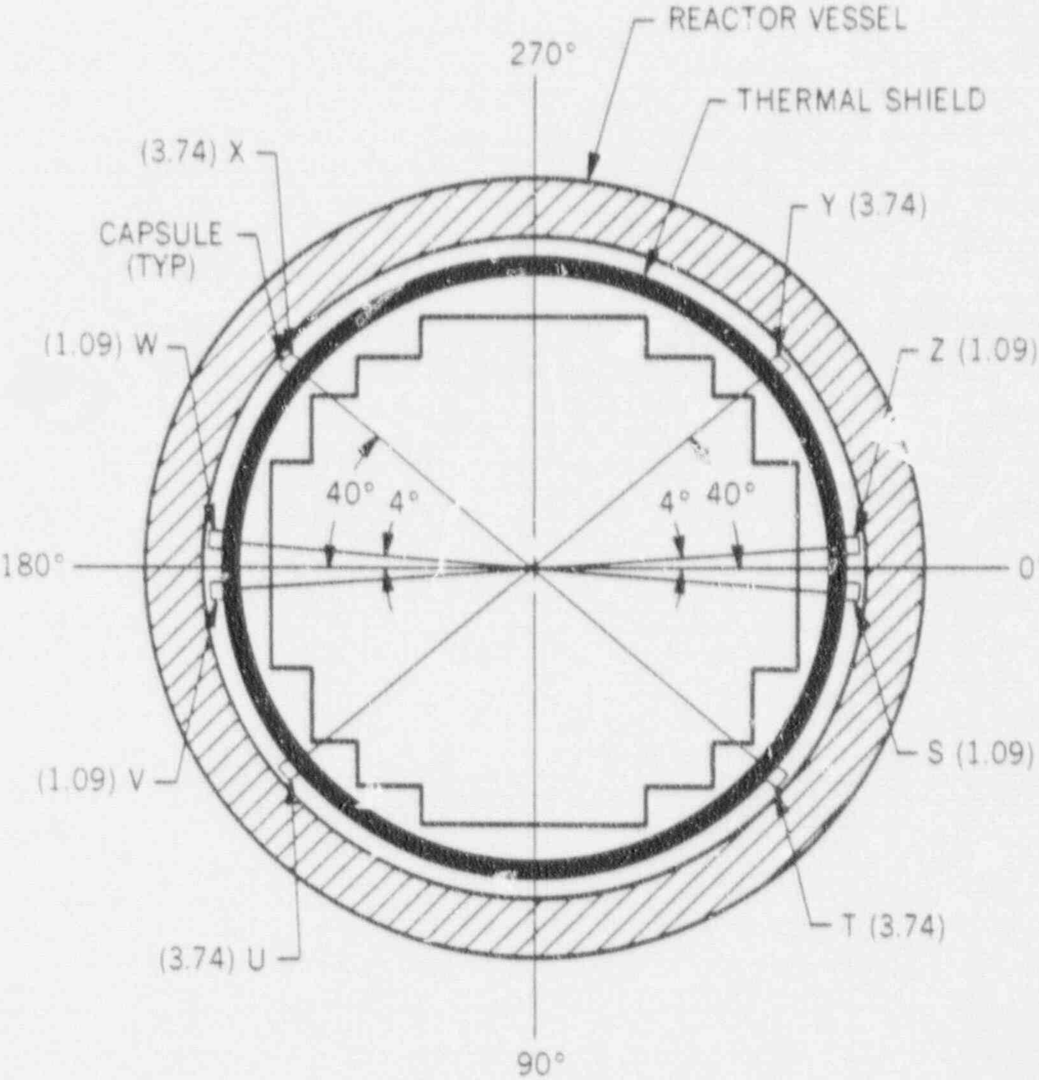
Note: Capsule lead factors shown in parentheses are for the original fuel management and capsule locations are for original placement.

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)



Note: Capsule lead factors shown in parentheses are for the original fuel management and capsule locations are for original placement.

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)



Note: Capsule lead factors shown in parentheses are for the original fuel management and capsule locations are for original placement.

plants. An overview of the plant-specific programs and capsule types is given in Table 2 of the BAW-1543, Rev. 4 Supplement. 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.

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, 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.

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

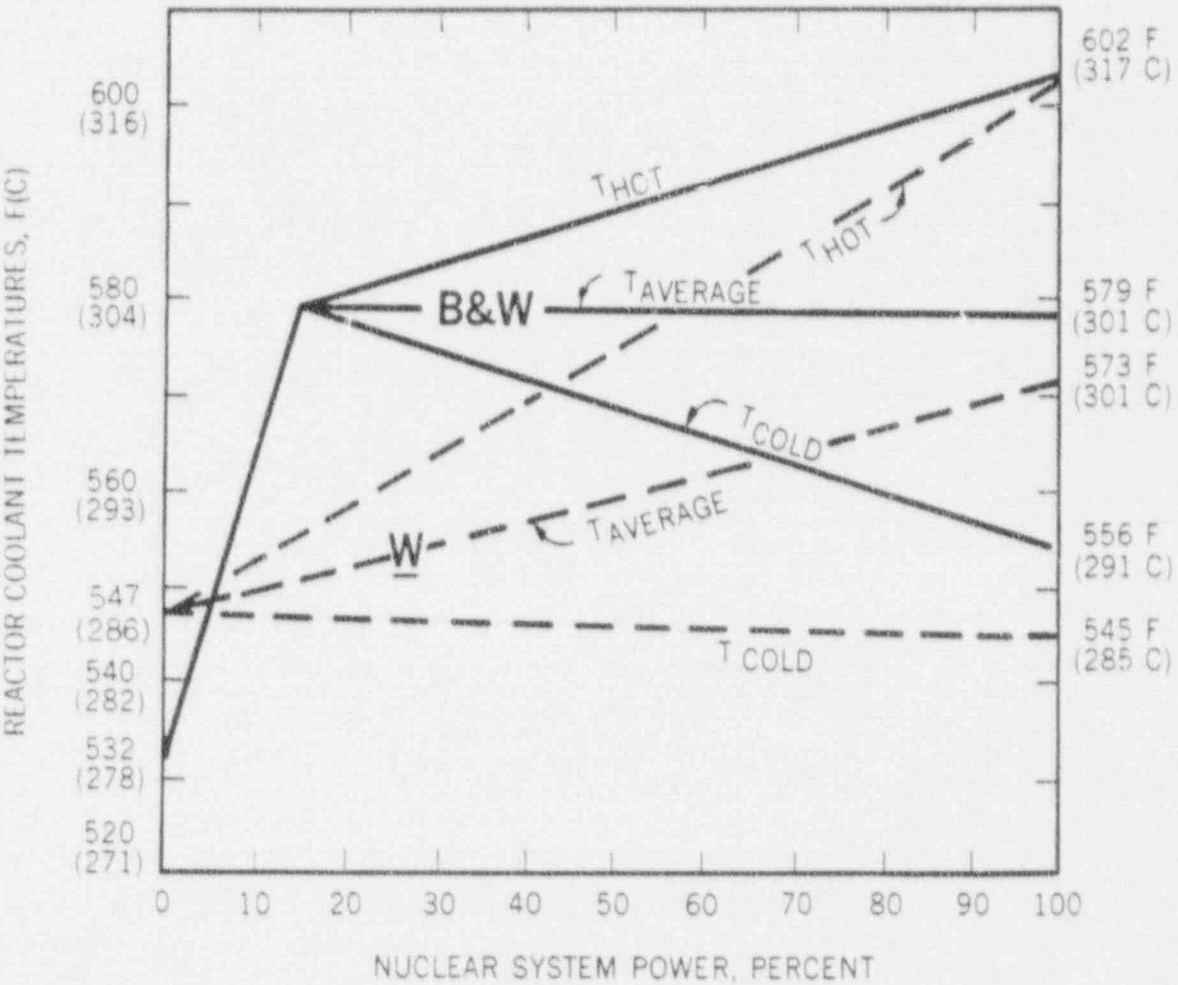


Table 3-6. Westinghouse Plant-Specific Surveillance Capsule Dosimetry

Neutron-Sensitive Element	Shield	Reaction Cross-Section Threshold Energy	Half-Life and 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	71 d ^{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	314 d ^{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-specific RVSP capsules.

Table 3-7. 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.

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 3-8 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 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 (see Figure C-1) could be replaced by subsize Charpy-sized tension test specimens (see Figure C-2). This enabled the inclusion of an 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 3-8. 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 ±25°F 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

Figure 3-18 SUPCAP Capsule, Type R-1

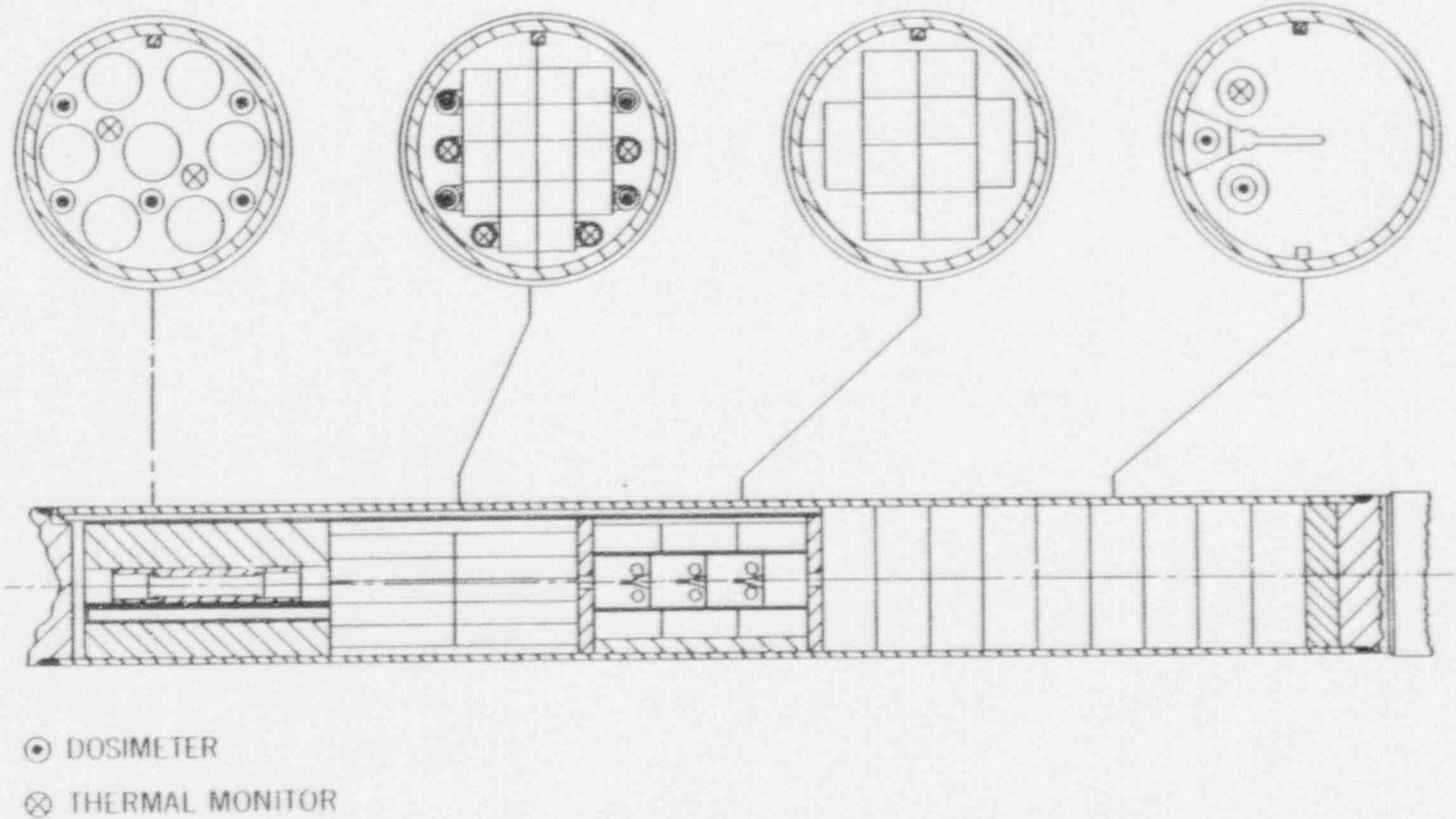
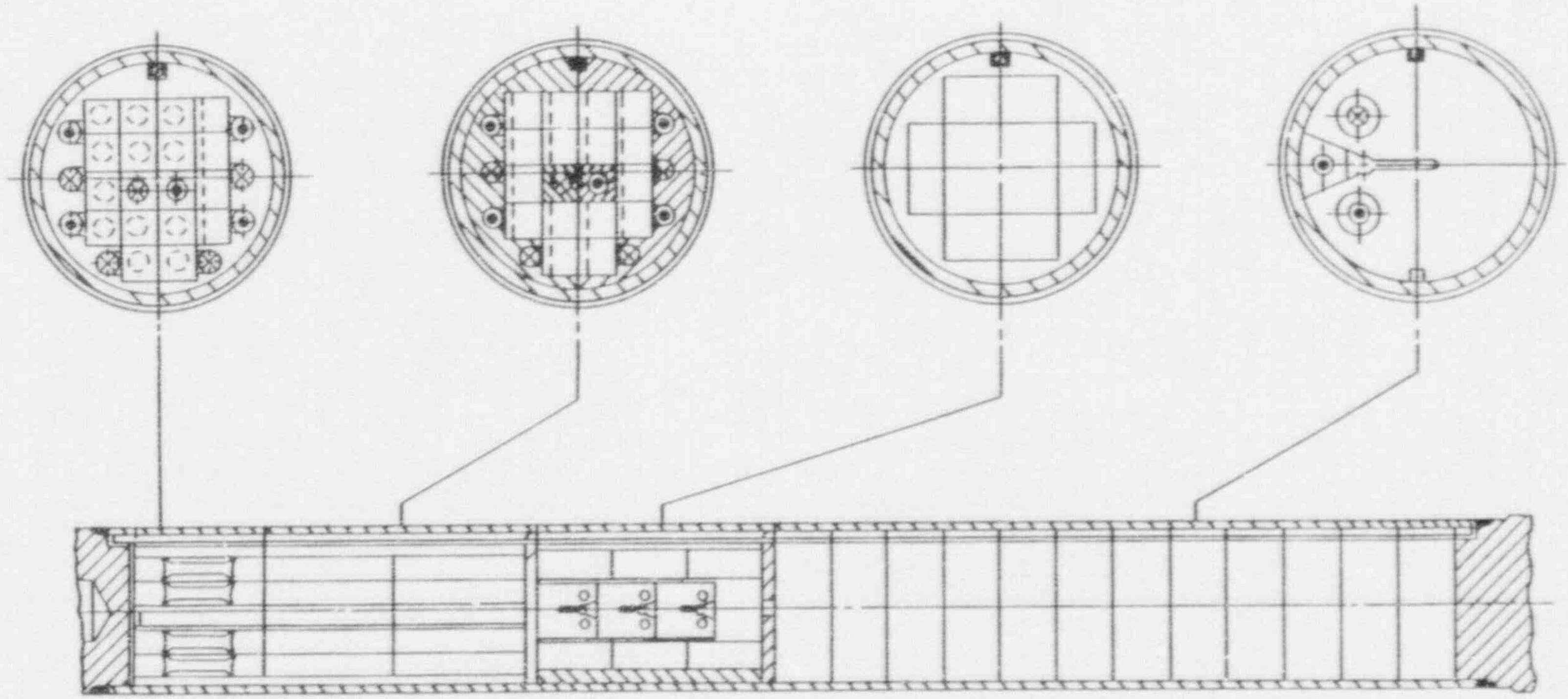


Figure 3-19 SUPCAP Capsule, Type R-2



◉ DOSIMETER

⊗ THERMAL MONITOR

Table 3-8. SUPCAPS -- Material and Specimens Per Capsule

Weld Metal	Amount of Specimens: Tension	Charpy	0.394 TCT	0.500 TCT	0.936 TRCT
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-1585	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.

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 25°F of the vessel 1/4T temperature. The coolant temperature serves as the lower bound and is also within 25°F of the vessel temperature at 1/4T.[21]

3.3.2.2. SUPCAP Dosimetry

Each capsule contains dosimeter tubes, which 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[32] and E 482-82.[33] The neutron dosimeters are distributed throughout the capsule to measure the neutron fluence at various locations.

Table 3-9 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 621°F. By determining which monitors have melted, the peak temperature at various locations within the capsule is determined. Table 3-10 lists the thermal monitors and their 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 3-11.

Table 3-9. Supplementary Weld Metal Surveillance Capsule Dosimetry

Neutron-Sensitive Element	Shield	Reaction Cross-Section Threshold Energy	Half-Life and Product Isotope
Long Tube (TMI2-LG1,-LG2)[34]			
^{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	71 d ^{58}Co
^{54}Fe	Gd	2.5 MeV	314 d ^{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)[34]			
^{237}Np	Gd	0.5 MeV	Appropriate fission products
^{238}U	Gd	1.1 MeV	Appropriate fission products
^{54}Fe	Gd	2.5 MeV	314 d ^{54}Mn
Short Tube Type DA (CR3-LG1,-LG2; DB1-LG1,-LG2)[35,36]			
^{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	71 d ^{58}Co
^{54}Fe	None	2.5 MeV	314 d ^{54}Mn
^{59}Co	None	Thermal	5.3 yr ^{60}Co
Short Tube Type DB (CR3-LG1,-LG2; DB1-LG1,-LG2)[35,36]			
^{237}Np	Gd	0.5 MeV	Appropriate fission products
^{238}U	Gd	1.1 MeV	Appropriate fission products
^{58}Ni	Gd	2.3 MeV	71 d ^{58}Co
^{54}Fe	Gd	2.5 MeV	314 d ^{54}Mn
^{63}Cu	Gd	6.1 MeV	5.3 yr ^{60}Co

Table 3-9. Supplementary Weld Metal Surveillance Capsule Dosimetry (Cont'd)

Neutron-Sensitive Element	Shield	Reaction Cross-Section Threshold Energy	Half-Life and Product Isotope
Long Tube Type DC (CR3-LG1, -LG2; DB1-LG1, -LG2)[35,36]			
^{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	71 d ^{58}Co
^{54}Fe	Gd	2.5 MeV	314 d ^{54}Mn
^{63}Cu	Gd	6.1 MeV	5.3 yr ^{60}Co
^{59}Co	None	Thermal	5.3 yr ^{60}Co

Table 3-10. SUPCAP Thermal Monitors

	Composition:	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
Capsule	Thermal Monitor					
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 3-11. 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 3-12. 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

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 3-12.

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 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. Both capsules will be irradiated and annealed. The L1 capsule will then be tested while the L2 capsule will be reirradiated and then tested. Benchmarking 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 3-13 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 E 844-86.[37] The neutron dosimeters are distributed throughout the capsule to measure the neutron fluence at various locations. Table 3-14 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 contain 6 sets of flux wires (HUPCAP Capsule A5 contains 7 sets of flux wires) and one full-diameter steel block containing radially spaced dosimeter wires to measure the neutron flux gradient thro 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 contains 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 3-13. HUPCAPS -- Material and Specimens Per Capsule

Capsule	Weld Metal	Specimens: Tension Test	Charpy	0.936 TRCT	0.417 1CT	3-Point Bend	WOL
A1	WF-25(a)*	4	12	5	-	-	-
	WF-67	4	12	5	-	-	-
	WF-70(b)**	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(c)***	4	12	5	-	-	-
	WF-182-1	4	12	5	-	-	-
	SA-1484	4	12	5	-	-	-
A4	WF-25(a)	4	12	5	-	-	-
	WF-67	4	12	5	-	-	-
	WF-70(b)	4	12	5	-	-	-
L1	WF-25(a)	4	12	5	-	-	-
	WF-67	4	12	5	-	-	-
	WF-70(b)	4	12	5	-	-	-
L2	WF-25(a)	4	12	5	-	-	-
	WF-67	4	12	5	-	-	-
	WF-70(b)	4	12	5	-	-	-
W1	WF-70(b)	3	10	-	4	3	-
	SA-1526	3	10	-	4	3	-
	SA-1585	3	10	-	4	3	-
A5	WF-209-1(d)****	-	32	-	-	-	2
	SA-1094(d)	-	8	-	-	-	2
	SA-1101(d)	-	23	-	-	-	2
	SA-1263(d)	-	23	-	-	-	2
	SA-1526(d)	-	10	-	-	-	-

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

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

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

****d - Irradiated material.

Table 3-14. Higher Fluence Supplementary Weld Metal Surveillance Capsule Dosimetry

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

Table 3-14. Higher Fluence Supplementary Weld Metal Surveillance Capsule Dosimetry (Cont'd)

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

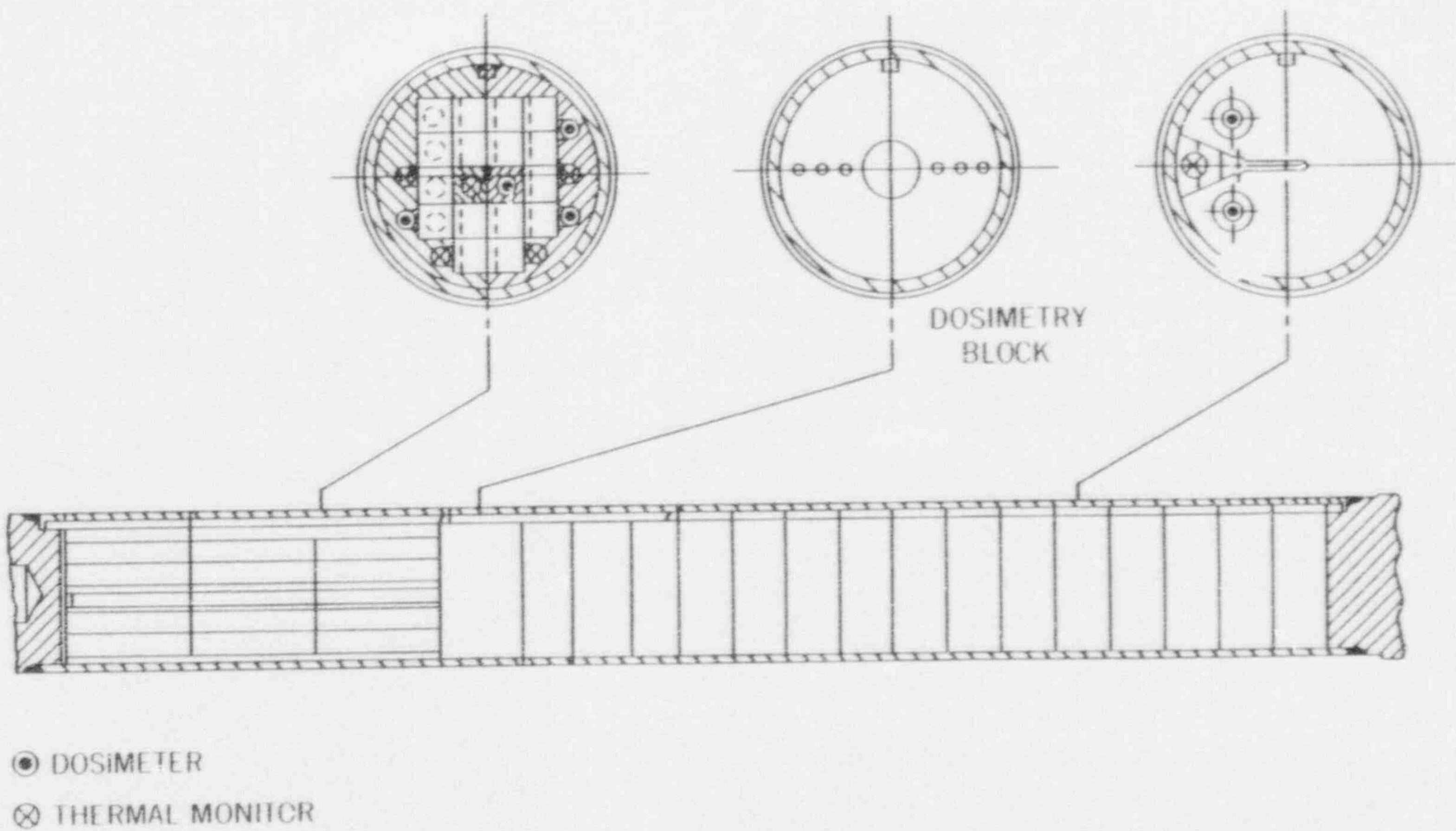
Table 3-14. Higher Fluence Supplementary Weld Metal Surveillance Capsule Dosimetry (Cont'd)

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



*L2 does not contain thermal monitors since it has been annealed.

Figure 3-20b HUPCAP Capsule A5

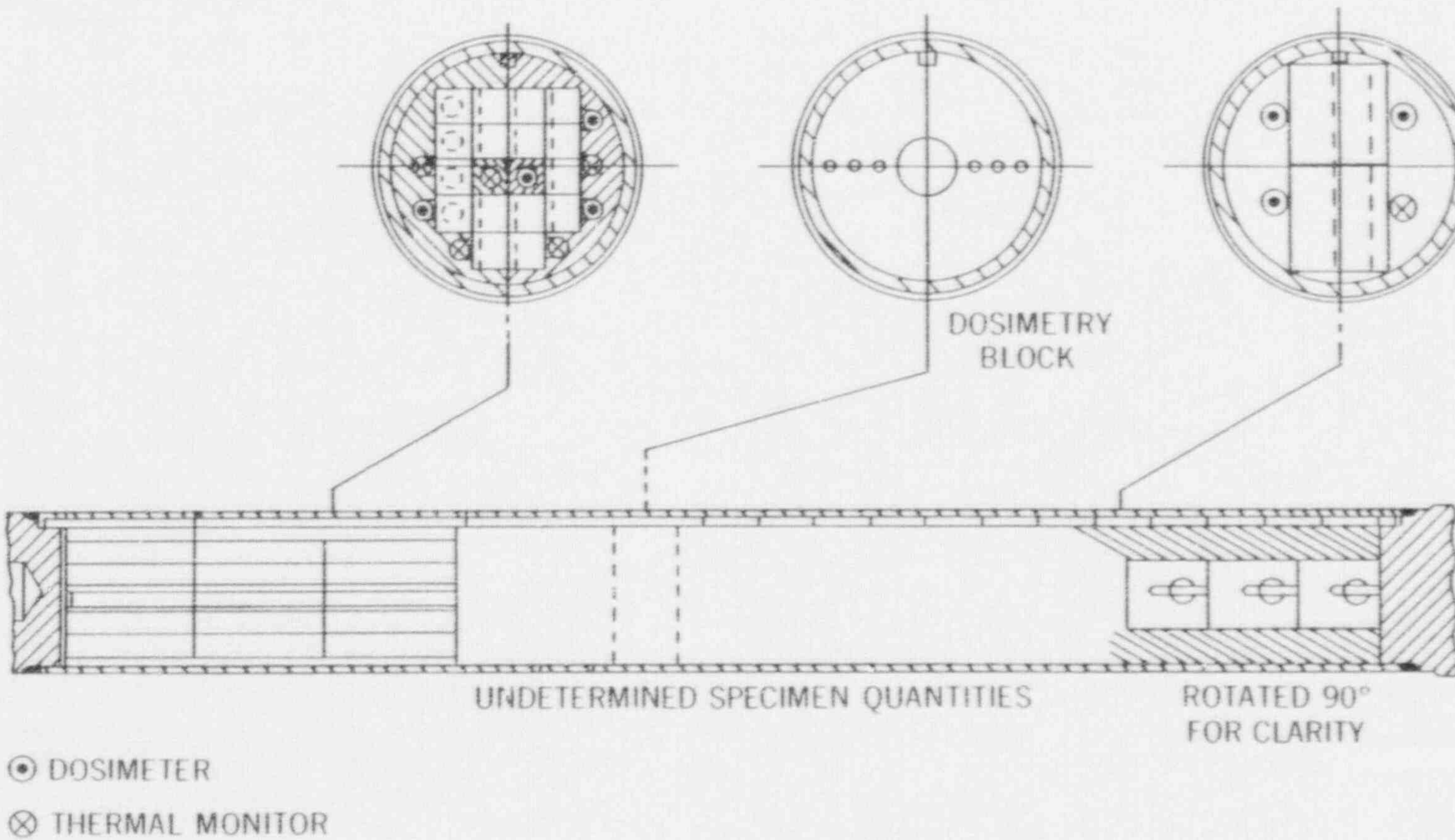
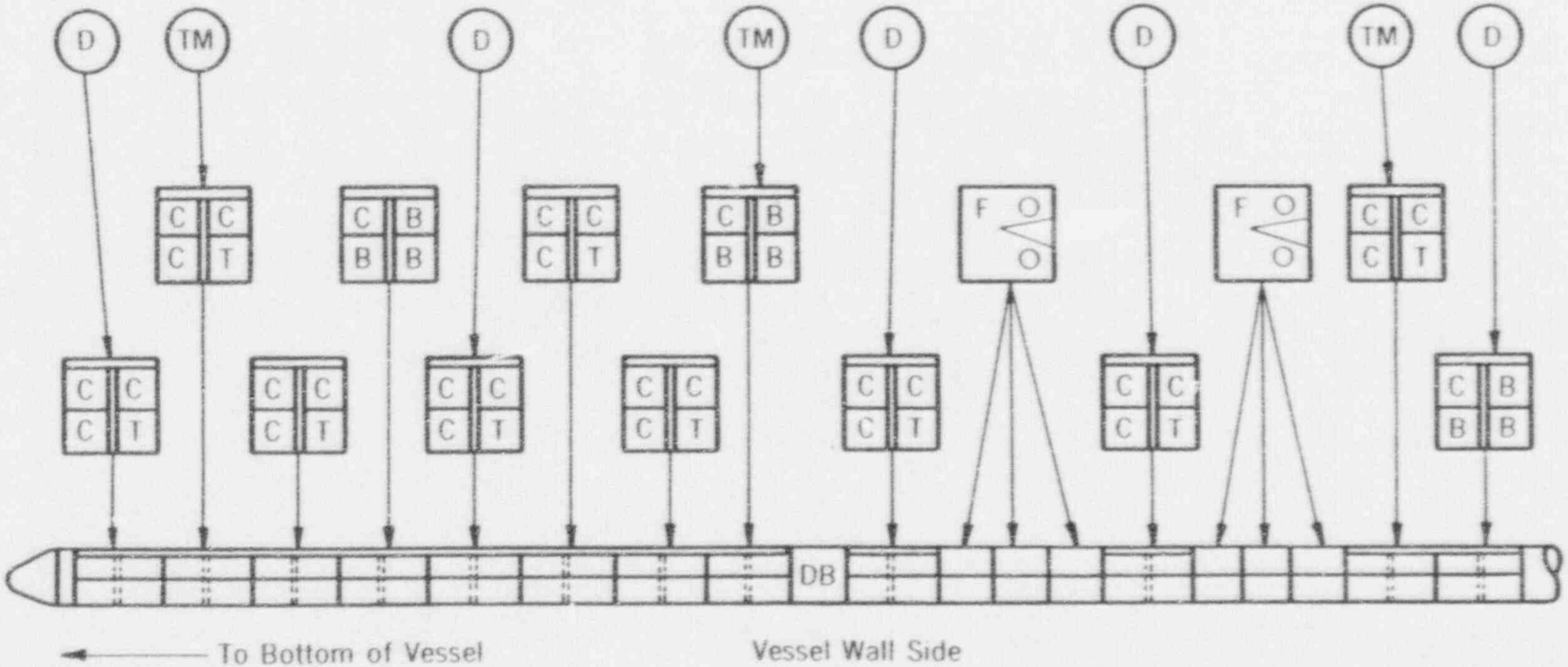


Figure 3-21 Placement of Specimens, Neutron Dosimeters and Thermal Monitors in HUPCAP W1

LEGEND

B - Slow-bend(3-point) Specimen
C - Charpy V-notch Specimen
D - Dosimeter (Radiometric) Set
DB - Np-U 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 621°F. By determining which monitors have melted, the peak temperature at various locations within the capsule is determined. Table 3-15 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.[38] 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 3-15. 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

* HUPCAP #2 does not contain thermal monitors

experiments. Plans will include irradiation of a well-characterized material 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 descri[illegible]d in each plant's RVSP report, where applicable. Additional data has been [illegible]ined on specimens from the weld metals used in the SUPCAPS. The type and number of specimens of each material are described in Table 3-11. Baseline data is also available for the eight welds in the SUPCAP program from the HSST baseline studies and is identified in Table 3-12. Unirradiated SA-1484 has not been tested; specimens are being fabricated as part of the HUPCAP effort for this purpose. Table 3-16 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 e[illegible]posure 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 3-16. 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.[39] 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 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. The schedules are found in BAW-1543, Rev. 4 Supplement. All capsules in the plant-specific RVSPs will be irradiated, but some of the capsules will not be tested since they would not provide 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 E 185-82 requirements is given in Table 8 of the BAW-1543, Rev. 4 Supplement.

3.8.2. B&W Plant-Specific RVSPs

The B&W plant-specific RVSP portion of the MIRVP is essentially complete. A few standby capsules are being irradiated to satisfy regulatory requirements. The remaining standby 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 needed.

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 1.7E19 in the SUPCAPS. Capsule S will be irradiated to the estimated IS-EOL fluence and tested. Capsule P is a standby capsule and is scheduled to be withdrawn and tested after receiving a fluence equivalent to 48 EFPY. Capsule N is a standby capsule and is 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

*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).

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. Capsules P and N only contain Charpy V-notch weld specimens; these two 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. Capsule X is the designated fourth RVSP capsule and will be withdrawn and tested as such. Capsules S, U, Y, and Z are designated as standbys and are scheduled to be withdrawn at one to two times the IS-EOL fluence (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 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, X and V, have been evaluated at fluences of 3.0E18 and 1.9E19, respectively. 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 as the fourth RVSP capsule. 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, SA-1229 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.[40] 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. 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 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. 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 materials are well characterized in the SUPCAPS, HUPCAPS, and the Crystal River Unit 3 and Oconee Units 2 and 3 RVSPs. Zion-2 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-2 capsule Z be irradiated to a IS-EOL fluence to fulfill regulatory requirements. 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 vessels 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 being irradiated in Surry Unit 2. The Westinghouse plant-specific capsules are irradiated in their respective plants. The schedules are shown in Tables III through V in the BAW-1543, Rev. 4 Supplement.

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 domestic PWR reactor vessels 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. The principal objective of the RVI Program is to assure the continued licensability of all ten participants and their sixteen reactor vessels. The MIRVP provides the data required by 10CFR50, Appendix G to accomplish this objective. Details of the MIRVP are provided in previous sections of 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 and capsules that have been added through shared resources to expand the data base for Linde 80 weld material. The term "integrated" in this instance refers to unified data sharing among all participants. The RVSPs in place prior to instituting this concept remain mostly 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 to delay the testing of a few low priority capsules as discussed in Section 3.

The power reactor portion of the MIRVP is comprised of two principal parts. The first is the continuation of the plant-specific surveillance programs that monitor the irradiation damage to selected materials. 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 specific weld metals.

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. Sixteen reactor vessels are included in this integrated program. Seven are B&W 177-FA design having very similar operating and design features and nine are of Westinghouse design. All 16 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

The relative neutron energy spectrum is a function of the geometry and materials of the reactor internals components. As shown in Tables 3-2 and 3-5, 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 10°F of each other during full power operation and approximately 25°F when operating at partial powers of 70% and above; $\pm$ 25°F is stated in Regulatory Guide 1.99, Revision 2,[42] 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).

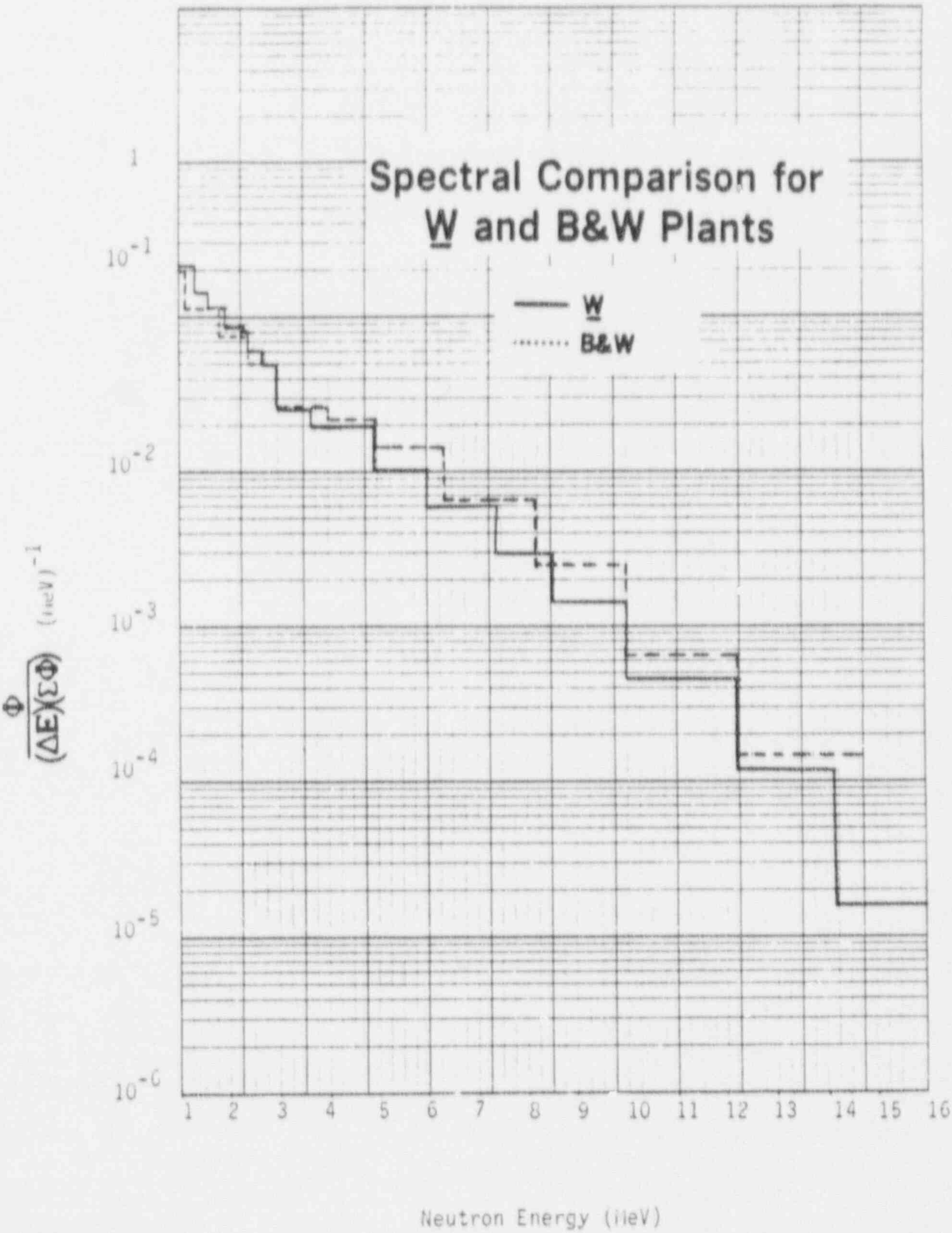
Gamma heating affects the 1/4T environment of all these reactor vessels similarly, since gamma heating is a function of the neutron energy spectrum, which, as discussed above, is similar for these reactors.

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. Cavity dosimeters have been installed with the benchmarking expected to be completed in 1993. 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 information. In addition, several utilities with Westinghouse-design reactors have completed or are conducting cavity dosimetry programs (see Appendix F).

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 justify the cost and man-rem exposure required to test the specimens. It should be noted, though, that 14 additional capsules are being irradiated (6 SUPCAPS and 8 HUPCAPS). These capsules contain test material of the greatest pertinence, improving the quality of the program beyond that of the original capsules.

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



D. Previous revisions were submitted to the NRC to obtain the necessary integrated program approval. BAW-1543, Rev. 3 was approved in 1991.[80]

E. The similarity of the 16 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 welds 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 joined the program in 1988 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 of B&W-fabricated PWR vessels prepared in accordance with the requirements of 10CFR50, Appendixes G and H. This revision includes the establishment of a supplement document (BAW-1543, Rev. 4 Supplement) which contains capsule withdrawal schedules.

L S Harbison 1-25-93
L.S. Harbison, Engineer II 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.

L B Gross 1/25/93
L.B. Gross, Advisory Engineer Date
Reactor Vessel Integrity Program

Verification of independent review.

K. E. Moore 1-26-93
K.E. Moore, Manager Date
Materials and Structural Analysis Unit

This report has been approved for release.

D L Howell 1/28/93
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 sixteen 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[39] and BAW-2121P[41]) 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 are reported in BAW-1803, Revision 1, "Correlation for Predicting the Effects of Neutron Radiation on Linde 80 Submerged Arc Welds."[43] 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	Chemical Composition, %									Impact Properties					
	C	Mn	P	S	Si	Ni	Cr	Mo	Cu	T_{NDT}, F	RT_{NDT}, F	C_v USE, ft-lb	UTS, ksi	YS, ksi	
Base Metal A	0.21	1.42	0.015	0.015	0.23	0.50	0.17	0.4	[illegible]	0	20	141[f]	86.1	64.3	L
										0	20	108[f]	86.5	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	118[e]	89.6	68.7	L
										20	20	115[e]	89.5	68.3	T
Weld Metal[b]	0.08	1.47	0.016	0.015	0.54	0.59	0.07	0.40	0.32	-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 9¾ h, brine quench	1200-1220F for 9½ h, brine quench	1100-1150F for 40 h, furnace cool
Base Metal B	C2800-2	SA 302 Gr B	Lukens	1600-1650F for 9½ h, brine quench	1200-1225F for 9½ h, brine quench	1100-1150F for 40 h, furnace cool
Weld Metal	WF-112	N/A[d]	N/A	N/A	N/A	1100-1150F for 40 h, furnace cool

(a) BAW-1820[22]

(b) BAW-1803, Revision 1[43]

(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.[44]

(f) BAW-2050.[45]

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

Material ID	Chemical Composition, %									Impact Properties			UTS,	YS,	
	C	Mn	P	S	Si	Ni	Cr	Mo	Cu	T_{NDT}, F	RT_{NDT}, F	C_v USE, ft-lb	ksi	ksi	
Base Metal A	0.24	0.63	0.006	0.012	0.25	0.75	0.36	0.62	0.04	20	20	152[f]	89.2	68.0	L
										20	20	133[f]	89.6	68.7	T
Base Metal B	0.21	0.62	0.010	0.010	0.23	0.80	0.39	0.58	0.02	-10	-10	160[e]	89.9	69.5	L
										-10	-10	138[e]	87.8	67.1	T
Weld Metal[b]	0.11	1.55	0.022	0.010	0.65	0.58	0.09	0.39	0.36	-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	1590 ± 20F for 4 h, water quench	1260F ± 20F for 10 h, water quench	1100-1150F for 33 h, furnace cool
Base Metal B	4P-1885; AWG 164	SA 508 Cl.2	Ladish	1590 ± 10F for 4 h,	Same as above	Same as above
Weld Metal	WF-209-1	N/A[d]	N/A	N/A	N/A	1100-1150F for 33 h, furnace cool

(a) BAW-1820[22]

(b) BAW-1803, Revision 1[43]

(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.[46]

(f) BAW-2051.[47]

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

	Chemical Composition, %									Impact Properties				
Material ID	C	Mn	P	S	Si	Ni	Cr	Mo	Cu	T_{NDT}, F	RT_{NDT}, F	C_v USE, ft-lb	UTS, ksi	YS, 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[b]	0.08	1.63	0.017	0.012	0.61	0.58	0.10	0.39	0.30	-20	25	66[b]	90.5	75.0

				Heat Treatment[c]		
Material ID	Heat No.	Spec No.	Supplier	Austenitizing	Tempering	Stress Relief
Base Metal A	522194; ANK191	SA 508 Cl.2	Ladish	1590 ± 20F for 4 h, water quench	1250F ± 20F for 10 h, water quench	1100-1150F for 30 h, furnace cool
Base Metal B	522314; AWS 192	SA 508 Cl.2	Ladish	Same as above	1240F ± 20F for 10 h, water quench	Same as above
Weld Metal	WF-209-1	N/A[d]	N/A	N/A	N/A	1100-1150F for 30 h, furnace cool

(a) BAW-1820[22]

(b) BAW-1803, Revision 1[43]

(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.[48]

(f) BAW-1697.[49]

Table A-4. Three Mile Island 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}, F	RT_{NDT}, F	C_vUSE, ft-lb	UTS, ksi	YS, ksi	
Base Metal A	0.24	1.36	0.010	0.017	0.23	0.57	0.19	0.51	0.09	10		131[f]	94.0	71.[illegible]	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]	83.2	60.4	L
										-10	20	112[e]	82.4	59.4	T
Weld Metal[b]	0.09	1.62	0.013	0.015	0.46	0.66	0.10	0.40	0.33	-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	1510-1535F for 5 h, brine quench	1200-1225F for 5 h, brine quench	1100-1150F for 27½ h, furnace cool
Base Metal B	C3307-1	SA 302 Gr B	Lukens	1600-1650F for 9½ h, brine quench	1200-1225F for 9½ h, brine quench 1225-1250F for 9½ h, brine quench	1100-1150F for 27½ h, furnace cool
Weld Metal	WF-25	N/A[d]	N/A	N/A	N/A	1100-1150F for 27½ h, furnace cool

(a)BAW-1820[22]

(b)BAW-1803, Revision 1[43]

(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.[50]

(f)BAW-1901.[51]

Table A-5. Crystal River 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}, F	RT_{NDT}, F	C_v USE, ft-lb	UTS, ksi	YS, 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[c]	0.08	1.65	0.021	0.013	1.00	0.10	0.07	0.45	0.41	---[b]	50[b]	79[c]	93.9[b]	77.1[b]
Weld Metal B[i]	0.10	1.57	0.018	0.009	0.54	0.60	0.094	0.43	0.35	---	---	---	---	---

Material ID	Heat No.	Spec No.	Supplier	Heat Treatment[c]: Austenitizing	Tempering	Stress Relief
Base Metal A	C4344-1 Cl.1	SA 533 Gr B	Lukens	1550-1600F for 4½ h, brine quench	1175-1200F for 6 h, brine quench	1100-1150F for 27 h, furnace cool
Base Metal B	C4344-2 Cl.1	SA 533 Gr B	Lukens	Same as above	Same as above	Same as above
Weld Metal A	Atypical	N/A[e]	N/A	N/A	N/A	1100-1150F for 27 h, cooling not reported
Weld Metal B	WF-209-1	N/A	N/A	N/A	N/A	1100-1150F for 27 h, cooling not reported

[a] BAW-1820[22]

[b] BAW-10144[52]

[c] BAW-1803, Revision 1.[43]

[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.[53]

[g] BAW-1898.[54]

[h] BAW-2049.[55]

[i] BAW-1500.[39]

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}, F	RT_{NDT}, F	C_vUSE, ft-lb	UTS, ksi	YS, ksi
Base Metal A	0.21	1.32	0.010	0.016	0.20	0.52	0.19	0.57	0.15	10	10	132[f]	94.9	72.0 L
										10	30	96[f]	94.6	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	147[g]	90.1	67.7 L
										-20	10	107[g]	90.3	67.8 T
Weld Metal[b]	0.09	1.49	0.016	0.016	0.51	0.59	0.06	0.39	0.28	-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	1550-1600F for 4½ h, brine quench	1200-1225F for 5 h, brine quench	1100-1150F for 29 h, furnace cool
Base Metal B	C5114-2 Cl.1	SA 533 Gr B	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 29 h, furnace cool

[a]BAW-1820[22]

[b]BAW-1803, Revision 1.[43]

[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.[56]

[f]BAW-2075.[57]

[g]BAW-1698.[58]

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

	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	0.20	1.33	0.010	0.015	0.19	0.53	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

				Heat Treatment[c]		
Material ID	Heat No.	Spec No.	Supplier	Austenitizing	Tempering	Stress Relief
Base Metal A	C5070-1	SA 533 Gr B Cl.1	Lukens	1550-1600F for 4½ h, brine quench	1200-1225F for 5 h, brine quench	1100-1150F for 28 h, furnace cool
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.[43]

(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.[59]

(f)BAW-2074.[60]

Table A-8. Davis-Besse 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	RT_{NDT}, F	C_v USE, ft-lb	UTS, ksi	YS, 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	0.40	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	1590 ± 10F for 4 h, water quench	1240F ± 10F for 5 h, air cool	1100-1150F for 15½ h, furnace cool
Base Metal B	123X244 AKJ233	SA 508 Cl.2	Ladish	Same as above	1240F ± 10F for 6 h, air cool	Same as above
Weld Metal	WF-182-1	N/A[(d)]	N/A	N/A	N/A	1100-1150F for 15½ h, furnace cool

(a) BAW-1820[(22)]

(b) BAW-1803, Revision 1.[(43)]

(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.[(61)]

(f) BAW-1882.[(62)]

Table A-9. R.E. Ginna 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	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

Material ID	Heat No.	Spec No.	Supplier	Heat Treatment[(c)]		
				Austenitizing	Tempering	Stress Relief
Base metal A	125P666	SA508, Cl.2	Beth.	1550F for 9 h, water quench	1220F for 12 h, air cool	1100F for 11½ h, furnace cool
Base metal B	125S255	SA508, Cl.2	Beth.	1550F for 15½ h, water quench	1220F for 18 h, air cool	Same as above
Weld metal	SA-1036	N/A[(d)]	N/A	N/A	N/A	1100F for 11½ h, furnace cool

(a) WCAP-7254[(23)] and WCAP-11026.[(63)]

(b) BAW-1803, Revision 1.[(43)]

(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			UTS, ksi	YS, ksi
	C	Mn	P	S	Si	Ni	Cr	Mo	Cu	T_{NDT}, F	RT_{NDT}, F	C_vUSE, ft-lb		
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 quench	1225F, 7 h, air cool	1125F for 11½ h, furnace cool
Base metal B	C1423	SA302 Gr.B	Lukens	Same as above	Same as above	1125F for 10½ h, furnace cool
Weld metal	SA-1263	N/A[f]	N/A	N/A	N/A	1125F for 11½ h, furnace cool

[a]WCAP-7513[24] and WCAP-8739.[65]

[b]Docket No. 50-266, response to 10CFR50.61, January 20, 1986.[66]

[c]WCAP-8743.[67]

[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.[43]

[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_v USE, 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 9½ h, water quench	1200F for 12 h, air cool	1125F for 12 h, furnace cool
Base metal B	122W195 VA1	A508 Cl.2	Beth.	1550F for 8 h, water quench	Same as above	Same as above
Weld metal	WF-193	N/A[e]	N/A	N/A	N/A	1125F for 11½ h, furnace cool

[a] WCAP-7712[25] and WCAP-9331.[68]

[b] BAW-1803, Revision 1.[43]

[c] WCAP-8738.[69]

[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	Chemical Composition, %									Impact Properties			UTS, ksi	YS, ksi
	C	Mn	P	S	Si	Ni	Cr	Mo	Cu	T_{NDT}, F	RT_{NDT}, F	C_v USE, ft-lb		
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	Tempering	Stress Relief
Base metal A	C4326-1	SA 533 Gr.B Cl.1	Lukens	1650-1700F for 9 h, water quench	1210F for 9 h, air cool	1125F for 15½ h, furnace cool
Base metal B	C4415-1	SA 533 Gr.B Cl.1	Lukens	Same as above	1200F for 9 h, air cool	Same as above
Weld metal	SA-1526	N/A[c]	N/A	N/A	N/A	1125F for 15½ h, furnace cool

(a) WCAP-7723[26] and WCAP-11415.[70]

(b) BAW-1803, Revision 1.[43]

(c) N/A - Not Applicable.

Table A-13. Surry 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	1.30	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

				Heat Treatment[a]		
Material ID	Heat No.	Spec No.	Supplier	Austenitizing	Tempering	Stress Relief
Base metal A	C4339-1	SA 533 Gr.B Cl.1	Lukens	1625F for 9 h, brine quench	1212F for 9 h, brine quench	1140F for 15¼ h, furnace cool
Weld metal A	R3008	N/A[b]	N/A	N/A	N/A	1140F for 15¼ h, furnace cool

[a] WCAP-8085[27] and WCAP-11499.[71]

[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			UTS, ksi	YS, ksi
	C	Mn	P	S	Si	Ni	Cr	Mo	Cu	T_{NDT}, F	RT_{NDT}, F	C_vUSE, ft-lb		
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	0.195	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 quench	1210F for 8 h, air cool	1125F for 10½ h, furnace cool
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 10½ h, furnace cool

[a] WCAP-7656[28] and Final Report SWRI Project No. 02-5131.[72]

[b] BAW-1500.[39]

[c] Safety Evaluation Report, Memorandum, S. Varga to J. W. Williams, April 26, 1984[73]

[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, % 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]	0.22	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	0.60	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 10½ h, water quench	1200F for 18 h, air cool	1125F for 10½ h, furnace cool
Base metal B	122S180 VA-1	A508 Cl.2	Lukens	1550F for 10¼ h, water quench	1210F for 18 h, air cool	Same as above
Weld metal A	SA-1094	N/A[b]	N/A	N/A	N/A	1125F for 10¼ h, furnace cool

[a] WCAP-7660[29] and Final Report SWRI Project No. 02-4221,[74] and Final Report SWRI Project No. 02-5380.[75]

[b] N/A - Not Applicable.

[c] Safety Evaluation Report, Memorandum, S. Varga to J. W. Williams, April 26, 1984[73]

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 9¾ h, brine quench	1212F for 9¾, brine quench	1125F for 25 h, furnace cool
Weld metal	WF-209-1	N/A[c]	N/A	N/A	N/A	1125F for 23 h, furnace cool

[a] WCAP-8064[30] and WCAP-9890.[76]

[b] BAW-1803, Revision 1.[43]

[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 9¾h, brine quench	1200-1225F for 9¾ h, brine quench	1100-1150F for 30 h, furnace cool
Weld metal	WF-209-1	N/A[c]	N/A	N/A	N/A	1100-1150F for 30 h, furnace cool

[a]WCAP-8132.[31]

[b]BAW-1803, Revision 1.[43]

[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

	Chemical Composition, wt.%									Impact Properties				
Ident. No.	C	Mn	P	S	Si	Ni	Cr	Mo	Cu	T_{NDT}, F	RT_{NDT}, F	C_vUSE, ft-lb	UTS, ksi	YS, ksi
SA-1094	0.10	1.44	0.014	0.011	0.50	0.60	0.14	0.36	0.30	--	--	66	90.8	70.2
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
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-1263	0.09	1.47	0.019	0.024	0.49	0.57	0.13	0.39	0.22	--	--	65	76.2	53.5
SA-1484	0.08	1.56	0.016	0.020	0.47	0.59	0.09	0.39	0.26	--	--	--	--	--
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	--
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	--
WF-25(6)[1]	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-25(9)[2]	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
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-70(N)[3]	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-70(B)[4]	0.09	1.62	0.018	0.011	0.59	0.59	0.10	0.40	0.35	-60	-2	69	--	--
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
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
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

1: 6 = Material from the TMI-2 nozzle drop-out (weld)

2: 9 = Material from the OC-3 nozzle drop-out (weld)

3: N = Material from the 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
SA-1094	Mn,Mo,Ni	Linde 80	Sub. arc	10-1/4 h at 1100-1150F
SA-1101	Mn,Mo,Ni	Linde 80	Sub. arc	25 h at 1100-1150F
SA-1135	Mn,Mo,Ni	Linde 80	Sub. arc	48 h at 1100-1150F
SA-1263	Mn,Mo,Ni	Linde 80	Sub. arc	11-1/4 h at 1100-1150F
SA-1484	Mn,Mo,Ni	Linde 80	Sub. arc	48 h at 1100-1150F
SA-1526	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
WF-25(6)[1]	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
WF-67	Mn,Mo,Ni	Linde 80	Sub. arc	48 h at 1100-1150F
WF-70(N)[3]	Mn,Mo,Ni	Linde 80	Sub. arc	48 h at 1100-1150F
WF-70(B)[4]	Mn,Mo,Ni	Linde 80	Sub. arc	30 h at 1100-1150F
WF-112	Mn,Mo,Ni	Linde 80	Sub. arc	48 h at 1100-1150F
WF-182-1	Mn,Mo,Ni	Linde 80	Sub. arc	15-1/2 h at 1100-1150F
WF-209-1	Mn,Mo,Ni	Linde 80	Sub. arc	23 h at 1100-1150F

1: 6 = Material from the TMI-2 nozzle drop-out (weld)

2: 9 = Material from the OC-3 nozzle drop-out (weld)

3: N = Material from the 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 tension test specimens, respectively. The Westinghouse plant-specific capsules contain standard 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 shown 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). The configurations and sizes of these specimens are described below.

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 configuration 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 unirradiated. The 0.417-inch specimen was 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 and are the largest specimens that could be accommodated in the capsules. The round compact fracture specimens conform to the requirements for disk-shaped specimens of ASTM E 399-81; Figure C-8 illustrates this specimen. 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 E 399-70T. 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 the more recently developed 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 E 813-87. The specimen configuration, 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 77. 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 configuration 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.[78] 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 nonside-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, 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 in his 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, inch: 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

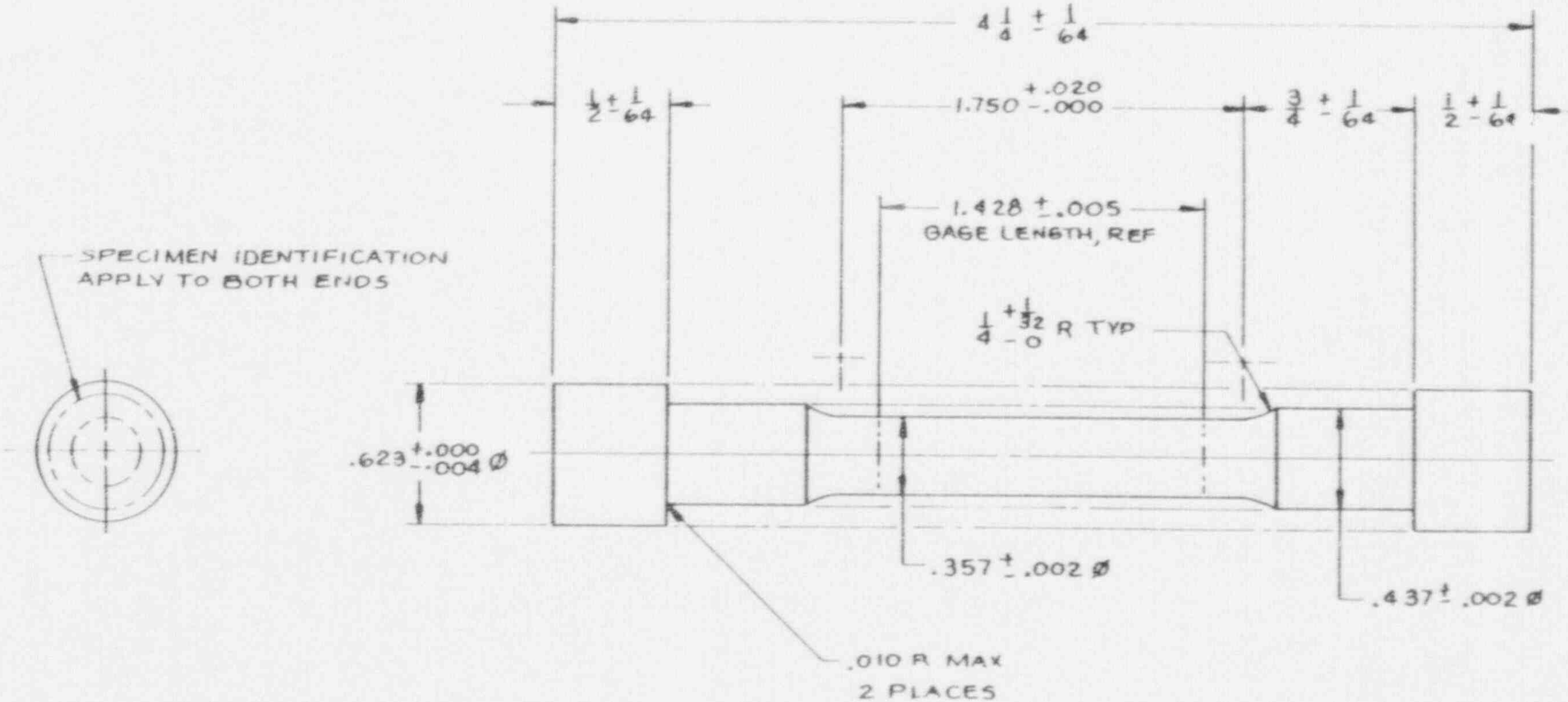


Figure C-2 Subsize Tension Test Specimens -- Used in SUPCAP Type R2 and HUPCAP Capsules

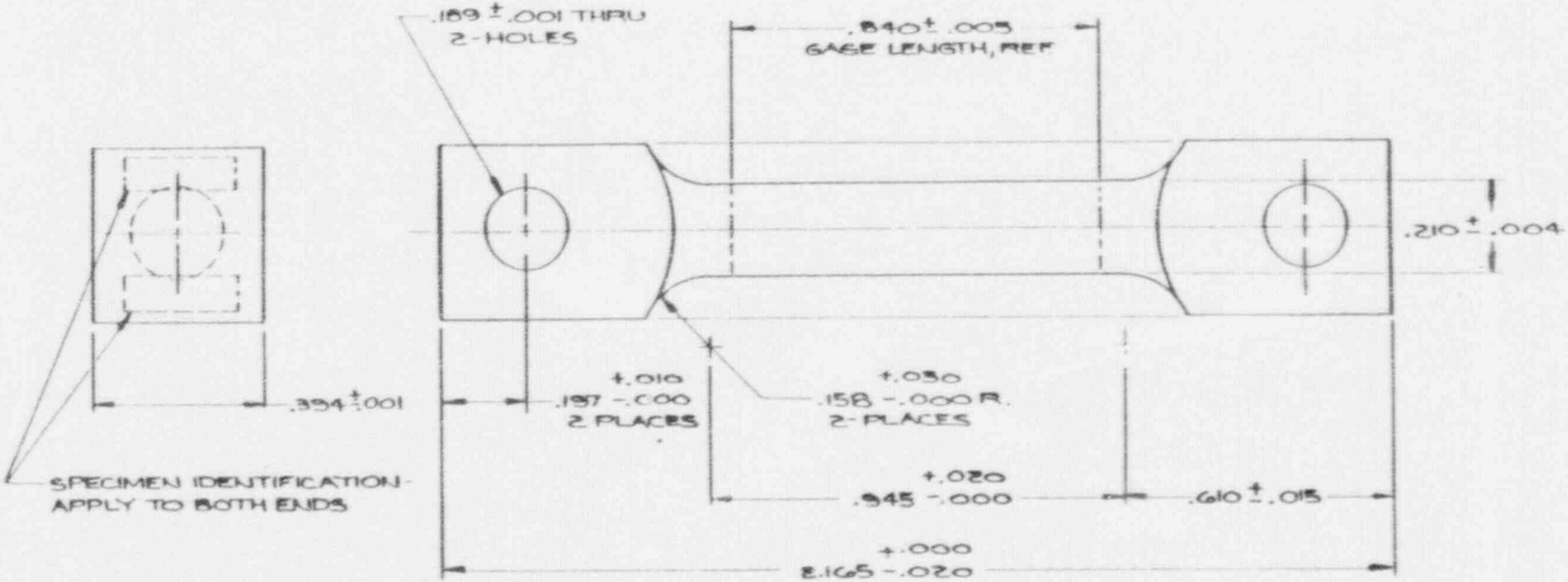


Figure C-3 Tension Test Specimens Used in Westinghouse Plant-Specific Capsules

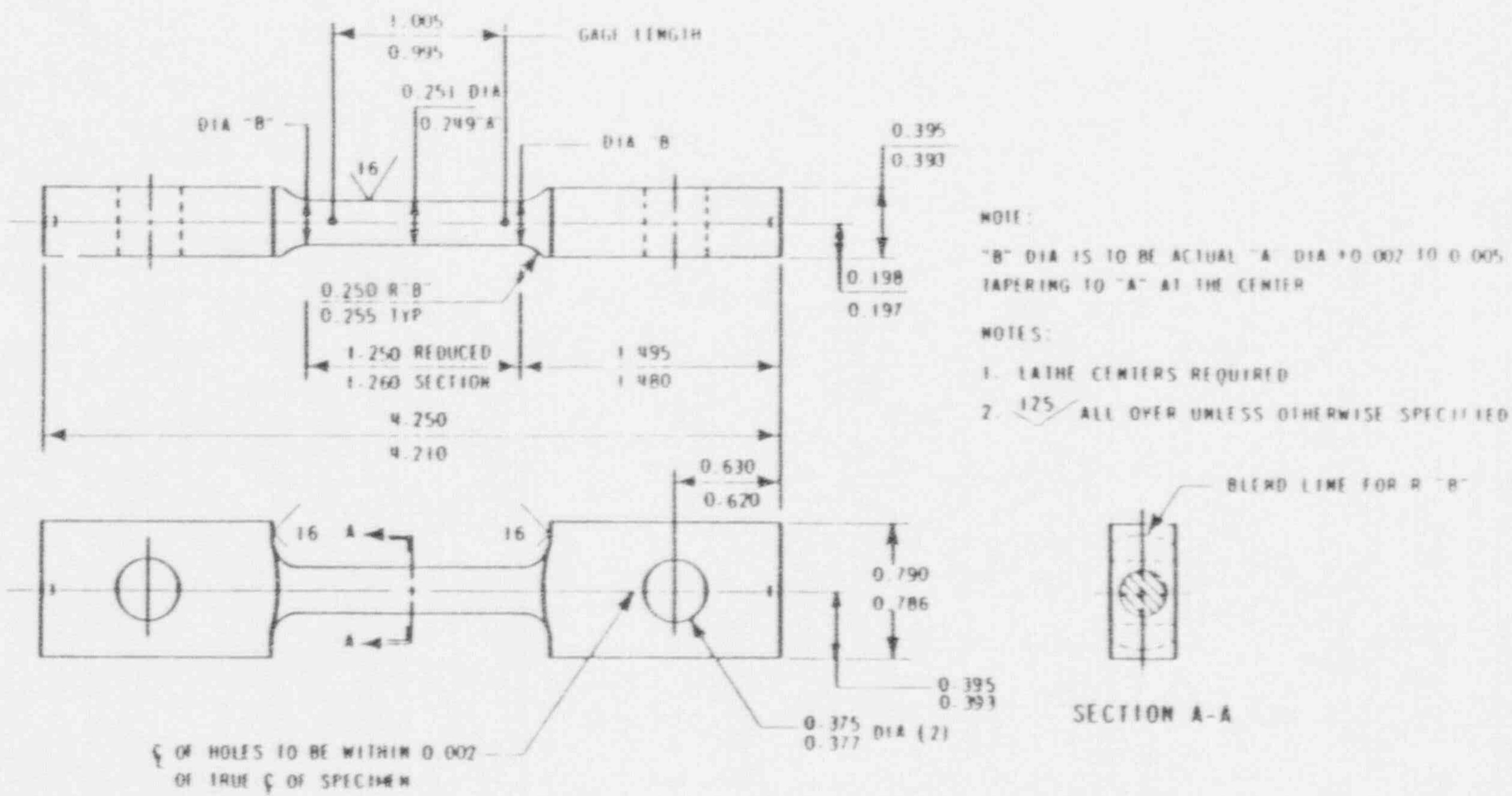


Figure C-4 Charpy V-Notch Specimens Used in B&W Plant-Specific and SUPCAP Capsules

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

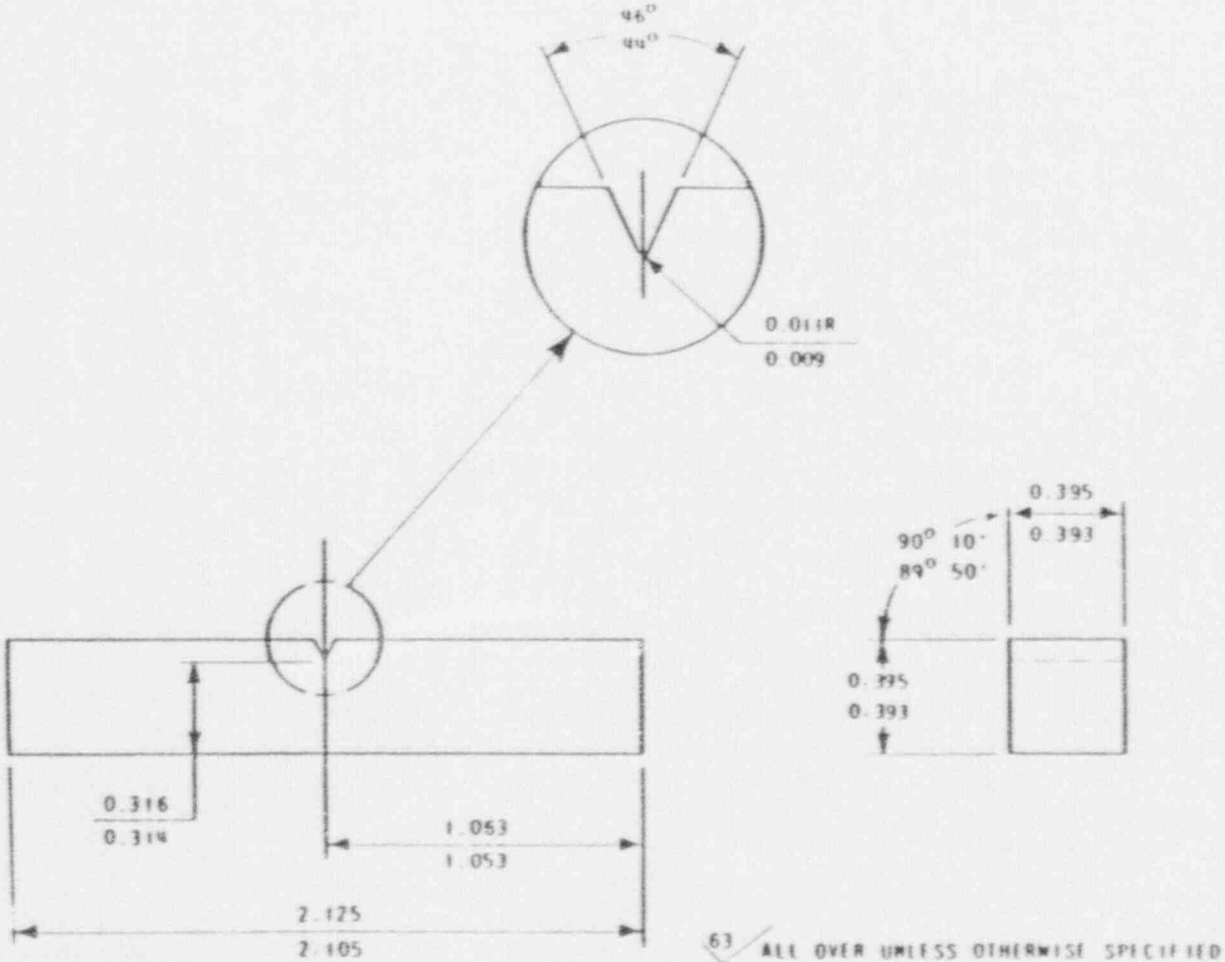


Figure C-6 Charpy V-Notch Specimens Used in HUPCAPS

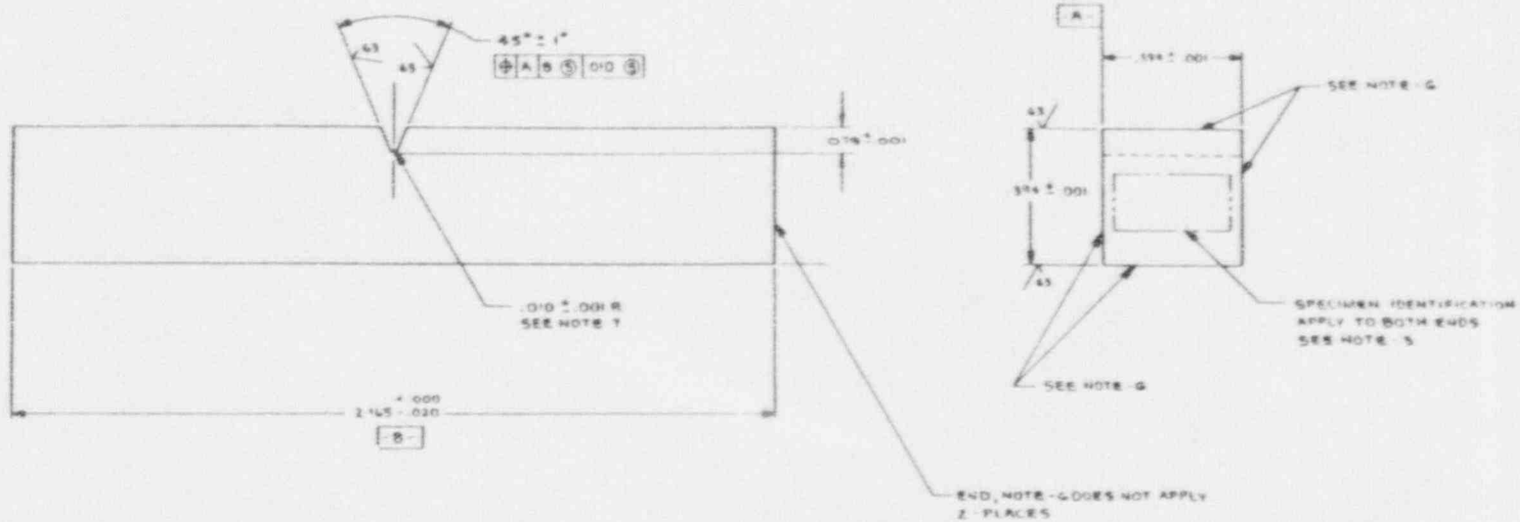


Figure C-7 Rectangular Compact Fracture Toughness Specimens -- Standard Proportions and Modifications for Measurement of Displacement at Load Line

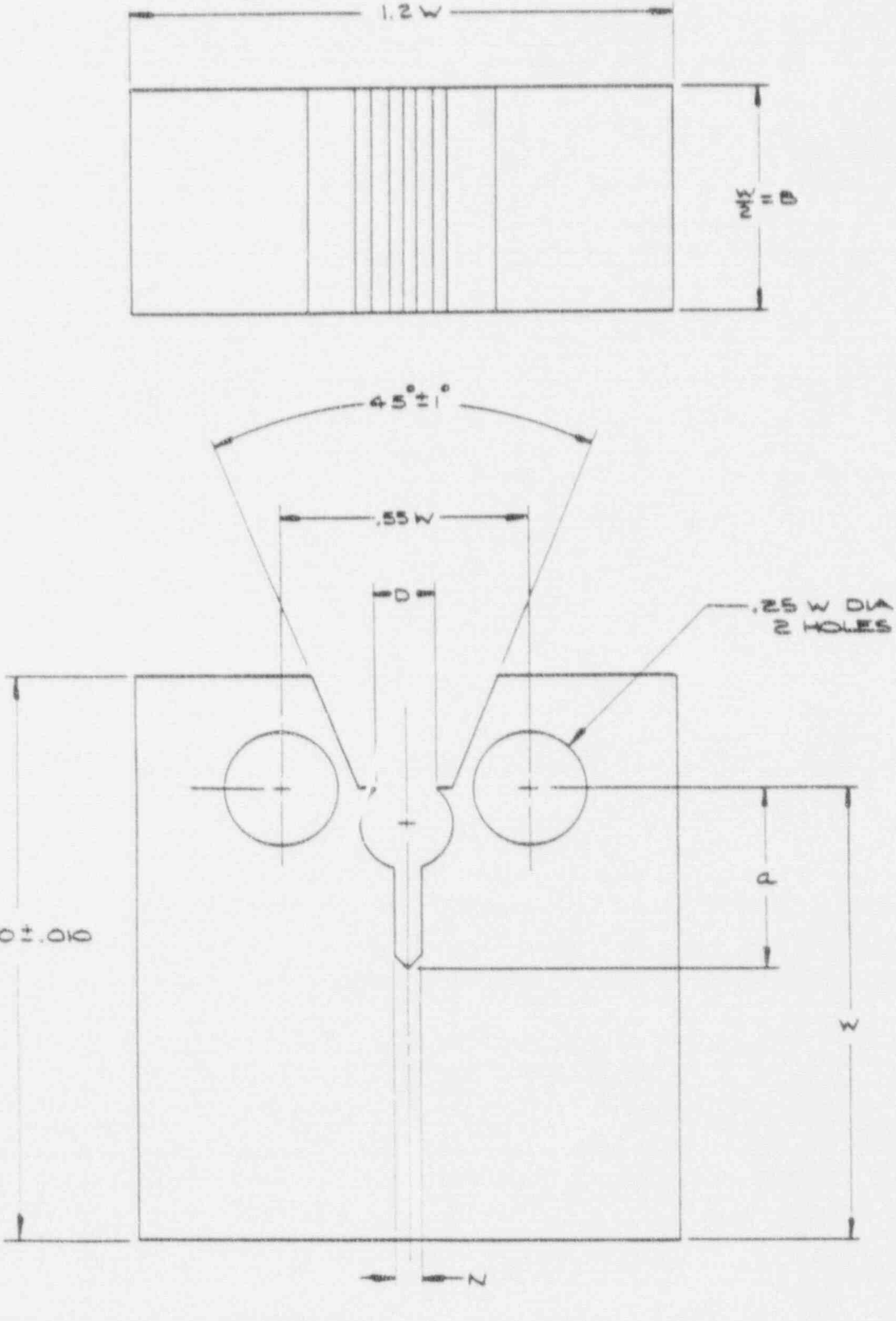


Figure C-8 Round Compact Fracture Toughness Specimens -- Dimensions and Modification for Measurement of Displacement at Load Line

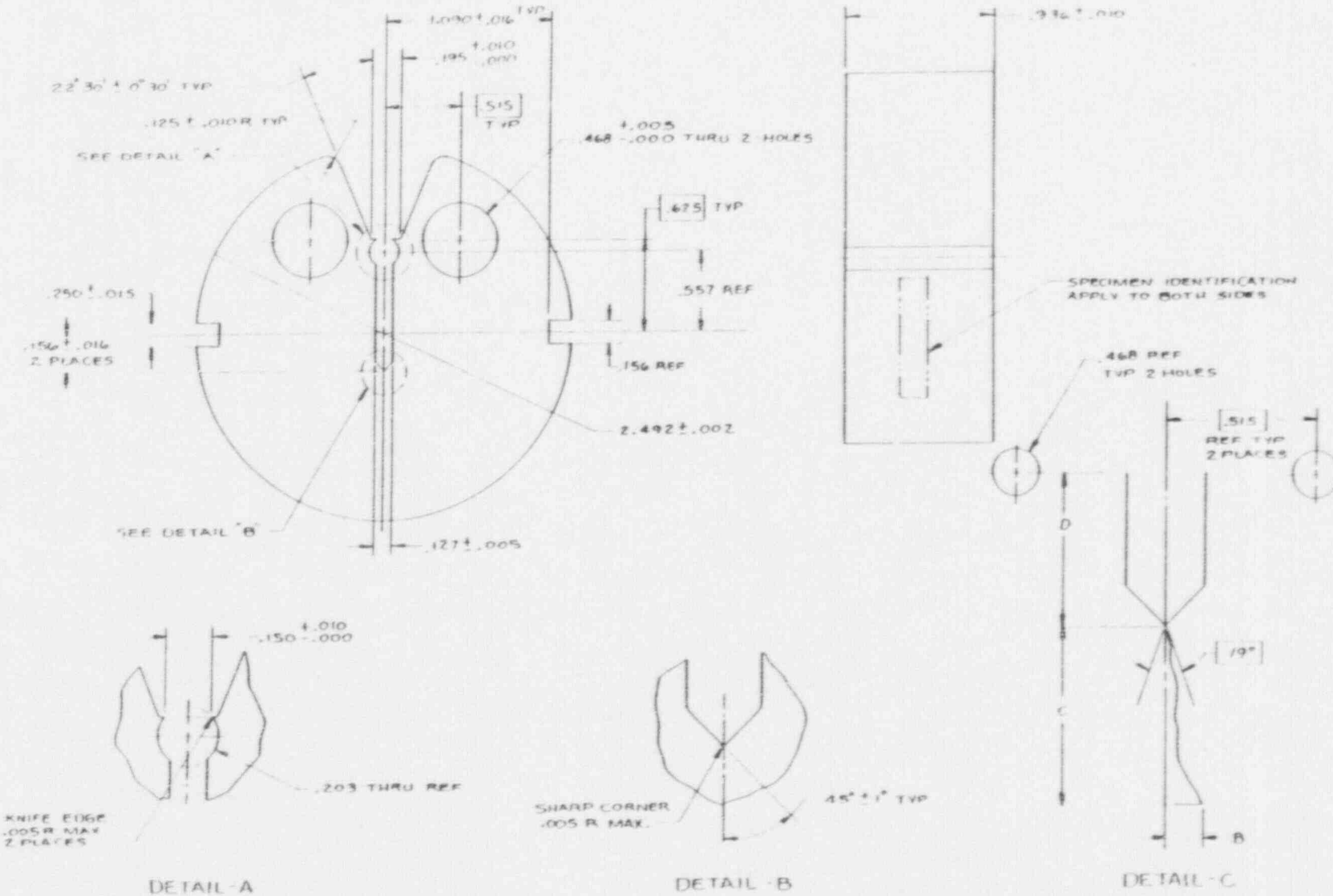
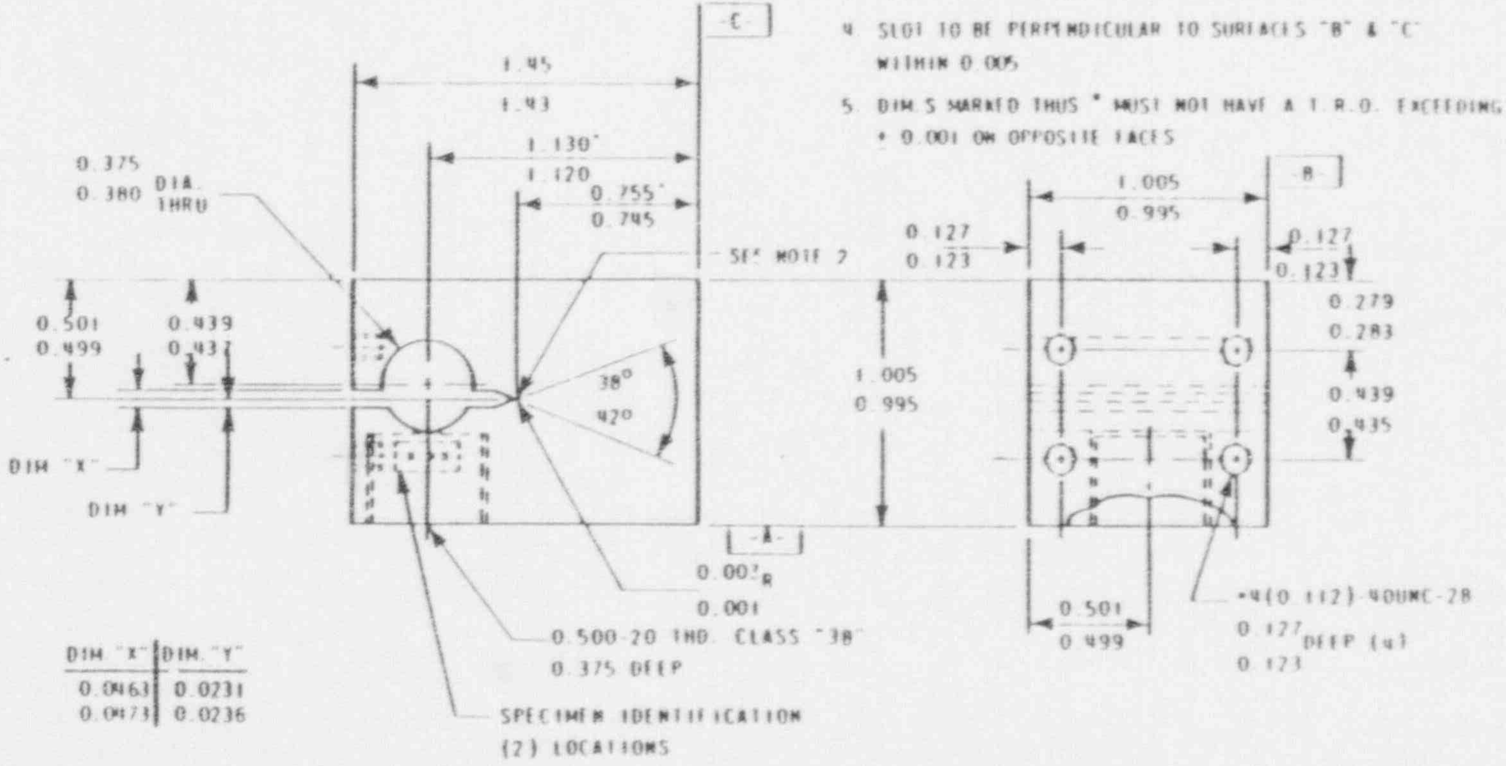


Figure C-9 Wedge-Opening-Loading Specimens

NOTES:

1. 32 ALL OVER
2. NOTCH DEPTH TO BE EXTENDED BY 0.09-0.156 BY FATIGUE CRACKING
3. SLOT TO BE PARALLEL TO SURFACE "A" WITHIN 0.005 TOTAL
4. SLOT TO BE PERPENDICULAR TO SURFACES "B" & "C" WITHIN 0.005
5. DIM.S MARKED THUS * MUST NOT HAVE A T.R.O. EXCEEDING ± 0.001 ON OPPOSITE FACES



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

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

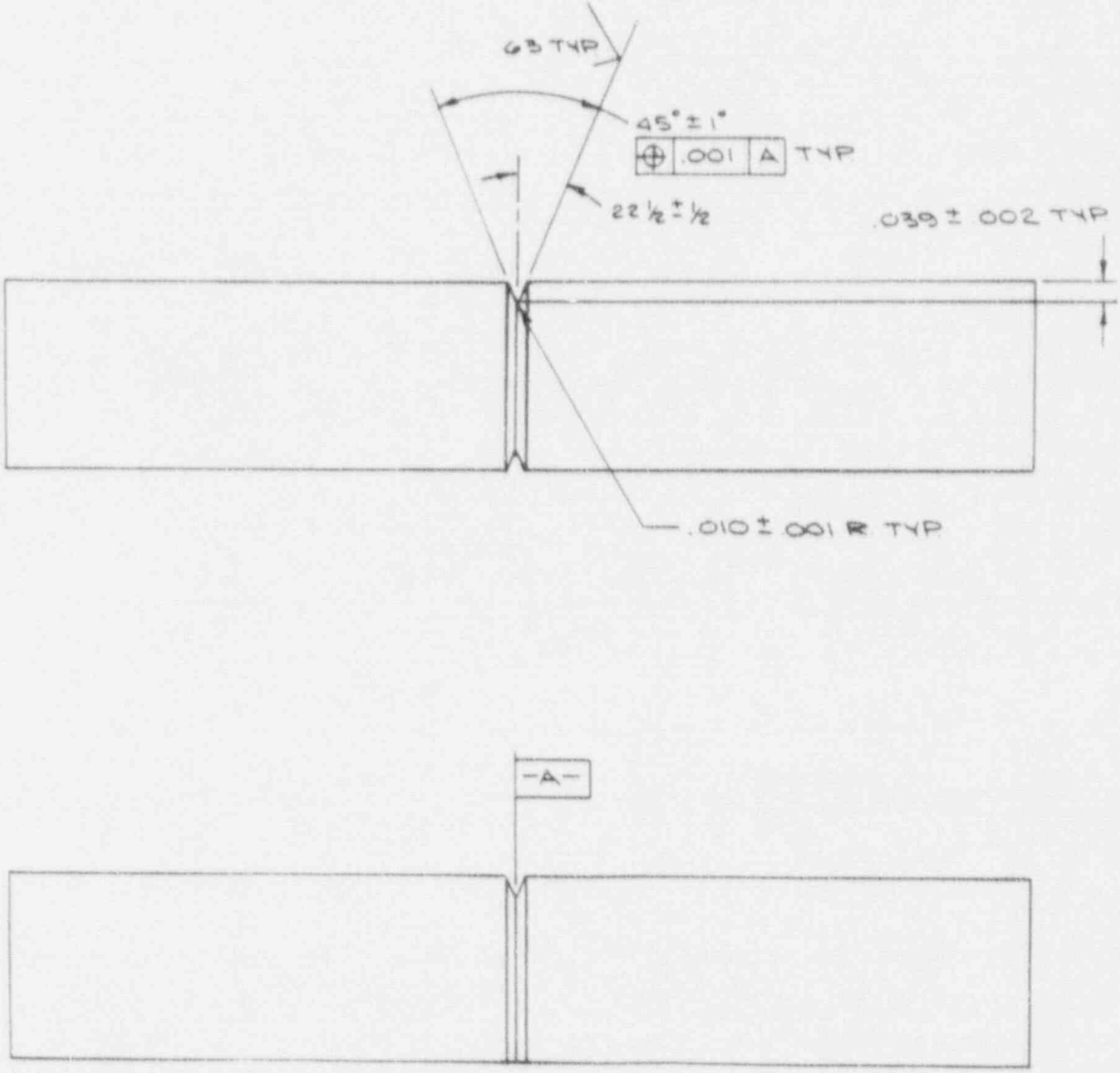
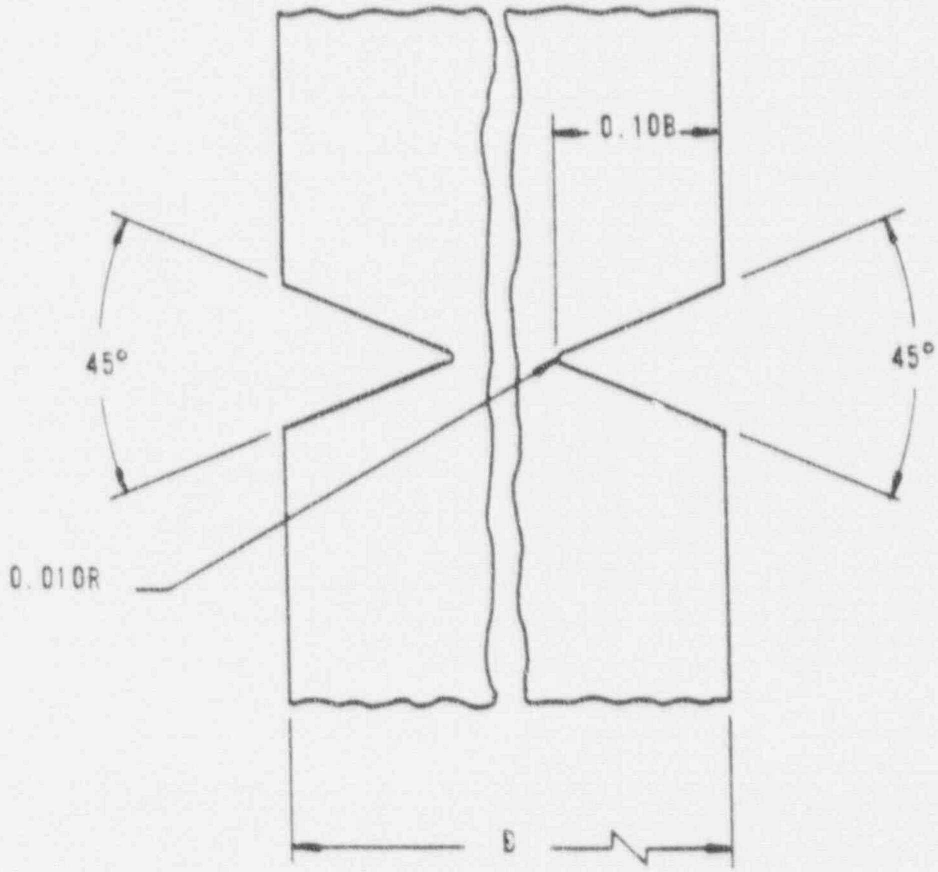


Figure C-11 Geometry of Side Grooves for 0.936 TRCT Specimen



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	9	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 Surveillance 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 Surveillance 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

Reactor Vessel Neutron Dosimetry Information

E.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.

E.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.

E.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.

E.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[79] which differs from the methodology described in Paragraph E.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. The numerical values for final period flux uncertainty will be determined after completion of the benchmark experiment presently underway at Davis-Besse Unit 1.

E.2. Summary

As discussed in Section E.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 E.1.1., E.1.2., and E.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 F

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