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BAW-1543

November 1981

INTEGRATED REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM

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by

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SUMMARY

This report describes the integrated reactor vessel surveillance program — an innovative approach to monitoring the irradiation-induced material changes of the steels and weldments routinely used in reactor vessels. Alterations to the material properties of reactor vessel materials — tensile strength, Charpy energy level, and fracture toughness — are characterized by irradiating appropriate test specimens in operating reactors.

Federal regulations require that all operating nuclear reactors have surveillance programs that involve preparation, irradiation, scheduled retrieval, and subsequent testing and evaluation of irradiated specimens. The integrated reactor vessel surveillance program not only complies with these requirements but also enhances the data acquired. The latter is accomplished by making data-sharing possible among the nine participating power plants as well as acquiring the fracture toughness data necessary to ensure the continued licenseability of the various reactors.

Specifically, the integrated reactor vessel surveillance program, initiated in 1976, assesses data from two separate but interrelated projects: (1) the plant-specific surveillance program integrates the various plant-specific surveillance programs to ensure the availability of data on a timely basis and which meets the basic requirement that each reactor have a surveillance program, and (2) the power reactor program, which will provide the fracture toughness properties of eight weld metals, which will complement the data obtained from the plant-specific capsules. The first program separates the participating power plants into two classes — those from which specimens come (guests) and those in which the specimen irradiations are performed (hosts). The nine power reactors involved (six guests, three hosts) are similar in both design and operating conditions. Specimens are enclosed in two different types of specially designed cylindrical capsules — normal plant-specific capsules and the larger research capsules.

A brief description of the federal guidelines and legislation and the combination of events that stimulated Babcock & Wilcox's development of the integrated reactor vessel surveillance program are discussed. The overall program is described and is divided into plant-specific reactor vessel surveillance programs and the power reactor research capsule program. Detailed descriptions of the types and properties of materials being investigated, the types of capsules used to contain the surveillance specimens, and the program and capsule designations are given in the appendixes.

Babcock & Wilcox
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Report BAW-1543

November 1981

Integrated Reactor Vessel Material Surveillance Program

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Key Words: Reactor Vessel Material Surveillance, Weldments, Base Metal, Tensile Strength, Charpy Energy, Fracture Toughness, Postirradiation Examination, Plant-Specific Capsules, Research Capsules

ABSTRACT

An integrated reactor vessel material surveillance program was designed when the surveillance capsule holder tubes in a number of reactors were damaged and could not be repaired without a complex and expensive repair program and considerable radiation exposure to personnel. The integrated program is feasible because of the similarity of the design and operating characteristics of the affected plants. Three plants were selected for the role of irradiation sites (host reactors), and the capsules of the other six plants (guest reactors) were irradiated on an integrated irradiation schedule with the capsules of the host reactors. The program consists of two parts — the first is the plant-specific program, which is the continued irradiation of the surveillance capsules removed from those reactors in which the capsule holder tubes were damaged along with the capsules from the host reactors; the second is made up of a number of special research capsules designed to provide fracture toughness data on a series of weld metals predicted to exhibit high sensitivity to irradiation damage.

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1. INTRODUCTION

1.1. General

The integrated reactor vessel surveillance program (RVSP) is the result of two events: failure of the capsule holder tubes in operating plants, and the necessity to obtain fracture toughness data for irradiated weld metals to ensure the continued licenseability of operating plants.

The original design of the B&W 177-fuel assembly (177-FA) class reactors included three reactor vessel surveillance capsule holder tubes (SCHTs) located near the reactor vessel inside wall, as shown in Figures 1-1 and 1-2. Each of the tubes was designed to hold two capsules containing reactor vessel surveillance specimens. In 1976, the SCHTs in a number of the 177-FA reactor vessels were found to be damaged. Subsequently, all operating 177-FA plants were shut down for inspection of the holder tubes. This inspection revealed that all of the SCHTs had been damaged to some extent. To prevent further damage and to eliminate the possibility of overall system damage if parts of the holder tube were dislodged, all the surveillance capsules and holder tubes that had either failed or were of the same design were removed from the vessels. Plants involved were Oconee Units 1, 2, and 3; Arkansas Nuclear One, Unit 1; Rancho Seco; and Three Mile Island Unit 1.

During the same period another event occurred that led to the need to improve the kind of data that were being obtained from the existing reactor vessel surveillance programs of operating 177-FA plants. It was found that certain weld metals used in the fabrication of the early-generation 177-FA reactor vessels may not meet current design requirements because of initial properties and unusual sensitivity to neutron embrittlement. This problem was further complicated by the fact that the capsules in the affected plants did not contain appropriate specimens, and, in some cases, the proper weld metals, to permit the required analyses of the vessels.

The integrated RVSP was developed in response to the immediate problems of the capsule-holder failures and the longer-range requirement of revamping existing RVSPs to improve the quality and quantity of fracture toughness data. In this cooperative data-sharing system, surveillance capsules removed from vessels with damaged tubes are placed in similar reactors for irradiation. The plants with the damaged SCHTs are called "guest reactors;" those in which the irradiations are to be done are the "host reactors." The following pairings of capsules and reactors were agreed upon by the affected reactor owners.

Guest reactors	Owners	Host reactors
Oconee 1, 2, and 3	Duke/Florida Power	Crystal River Unit 3
Arkansas Nuclear One, Unit 1 Rancho Seco	AP&L/Toledo Edison SMUD	Davis-Besse Unit 1
Three Mile Island Unit 1	Met Ed/Met Ed	Three Mile Island Unit 2

The rationale motivating the implementation of this unique program is discussed in the ensuing paragraphs of this section.

Damage to the original SCHTs precipitated the design, manufacture, and testing of improved tubes. SCHTs of this improved, NRC-approved design were installed in Davis-Besse 1, Crystal River 3, and Three Mile Island 2 prior to their initial startup, i.e., before neutron activation of the reactor internals.

However, installing the redesigned SCHTs in already-irradiated B&W reactors presented substantial difficulties, primarily because precision machining, alignment, and inspection had to be performed remotely and under water. These circumstances could have caused significant radiation exposure — up to about 100 man-rem per reactor — to plant personnel. Therefore, an alternative program that did not involve reinstalling SCHTs in irradiated plants was proposed.

Since Crystal River 3, Davis-Besse 1, and Three Mile Island 2 had the same reactor design as those six reactors from which damaged holder tubes were removed and were scheduled for startup in an appropriate time-frame, it was cost-effective and technically acceptable to use the three new plants as hosts for the capsule specimens of the other plants. The exchange plan presented previously was devised and the integrated RVSP was implemented.

The most important aspect controlling the success of the integrated program is the commonality of the irradiation sites and reactors involved. In the case of the B&W 177-FA plants participating in this integrated program, there are no dissimilarities among the plants except for the materials of construction. This is a natural dissimilarity because heats of steel are not large enough to make more than one reactor vessel; however, the various vessels are made of the same type and grade of material, which is the primary consideration. The design characteristics of various host reactors and their guest units are compared in Tables 1-1, 1-2, and 1-3.

Other reactor parameters that are significant in evaluating the similarity of the host and guest reactors are (1) the relative neutron flux energy spectrum, (2) the irradiation dose rate, and (3) the irradiation temperature. The relative neutron energy spectrum is a function of the geometry and materials of the reactor internals components. As shown in the tables, the dimensions and materials of the host and guest reactors are essentially the same. Thus, no difference in the relative neutron energy spectra is expected. Similarly, differences in irradiation dose rates between the guest and host reactors would be due only to variations in power levels. Since the licensed power levels are comparable, variations in the irradiation rate are attributable to plant maneuvering. Averaged over time, the variations in power level due to those maneuvers will have no significant effect, as confirmed by surveillance results from a number of plants.

The reactor vessel beltline region and normal surveillance specimens are exposed to reactor coolant inlet conditions when being irradiated in the host reactors. Two factors that could contribute to differences in the irradiation environments of the capsules are design variations among the plants and power level changes due to maneuvering.

The variations due to design differences between the host and guest reactors are insignificant, as shown in Tables 1-1, 1-2, and 1-3. Between partial- (~15%) and full-load conditions, the inlet temperature will vary by about 20F as an inverse function of power level. The duration of this variation due to maneuvering is comparable among plants over time; this is supported by the comparability of available reactor vessel surveillance results. In any case, the inlet condition temperatures are considered too low to cause significant self-annealing. The inlet temperature will also vary about 40F between hot zero power and partial load conditions. This variation is a direct function of power level (0 to 15%) and again is not significant because of the low temperature and the expected comparability in duration over the long term.

1.2. Objective

The integrated RVSP is designed to provide the fracture toughness data for the materials in the 177-FA reactor pressure vessels necessary to ensure the continued licenseability of the plants.

As originally designed, the individual reactor surveillance programs for the plants with failed holder tubes did not provide the fracture toughness data needed to perform the currently required analytical evaluations of reactor vessel integrity. Although the original RVSPs were designed and fabricated in complete accordance with the existing standards and regulations, these changed with the development of fracture mechanics techniques as a method of ensuring reactor system integrity. Consequently, changes developed in the required materials data and the type of specimens necessary to obtain the data. Although the integrated program eliminates the need to replace the failed specimen holder tubes on operating plants and thus eliminates the high radiation exposure, it does provide a systematic and redundant program to develop the new materials data needed to assess the future integrity of this group of reactor plants.

Table 1-1. Comparison of Plant Parameters for Host Reactor Davis-Besse 1 and Guest Reactors Rancho Seco and Arkansas Nuclear One-1

Plant Parametera	Host reactor: Davis-Besse 1	Guest reactors: Rancho Seco	ANO-1
Design heat output (core), MWt	2772	2772	2568
Design overpower, % des power	112	112	112
System pressure (nom), psia	2200	2200	2200
Coolant flow rate, 10^{-6} lbm/gpm	131.3/352,000	137.8/369,000	131.3/352,000
Coolant temperatures, F			
Nominal inlet	555	557	554
Avg rise in vessel	53	51	48
Avg in vessel	582	582	579
No. of fuel assemblies	177	177	177
Type of fuel assemblies	Mark B (15×15)	Mark B (15×15)	Mark B (15×15)
Core barrel ID/OD, in.	141/145	141/145	141/145
Thermal shield ID/OD, in.	147/151	147/151	147/151
Core structural characteristics			
Core equiv diam, in.	128.9	128.9	128.9
Core active fuel height, in.	144	144	144
Reflector thickness, compos'n			
Top (water + steel), in.	12	12	12
Bottom (water + steel), in.	12	12	12
Side (water + steel), in.	18	18	18
Reactor vessel design parameters			
Principal material	SA508 Cl.2	SA533 GrB Cl.1	SA533 GrB Cl.1
Design pressure, psig	2500	2500	2500
Design temperature, F	650	650	650
Shell ID, in.	171	171	171
Overall vessel-closure head height[a], ft/in.	40/8.75	40/8.75	40/8.75
Core barrel-thermal shield principal material	Type 304 SS	Type 304 SS	Type 304 SS

[a]Over cladding and instrumentation nozzles.

Table 1-2. Comparison of Plant Parameters for Host Reactor Crystal River 3 and Guest Reactors Oconee Units 1, 2, and 3

Plant parameters	Host reactor: Crystal River 3	Guest reactors		
		Oconee 1	Oconee 2	Oconee 3
Design heat output (core), MWt	2452	2568	2568	1568
Design overpower, % des power	114	112	112	112
System pressure (nominal), psia	2200	2200	2200	2200
Coolant flow rate, 10^{-6} lb/h/gpm	131.3/352,000	131.3/352,000	131.3/352,000	131.3/352,000
Coolant temperatures, F				
Nominal inlet	555	554	554	554
Avg rise in vessel	48	50	50	50
Avg in vessel	579	579	579	579
No. of fuel assemblies	177	177	177	177
Type of fuel assemblies	Mark B (15×15)	Mark B (15×15)	Mark B (15×15)	Mark B (15×15)
Core barrel ID/OD, in.	141/145	141/145	141/1[illegible]	141/145
Thermal shield ID/OD, in.	147/151	147/151	147/[illegible]	147/151
Core structural characteristics				
Core equiv diameter, in.	128.9	128.9	128.9	128.9
Core active fuel height, in.	144	144	144	144
Reflector thickness, compos'n				
Top (water + steel), in.	12	12	12	12
Bottom (water + steel), in.	12	12	12	12
Side (water + steel), in.	18	18	18	18
Reactor vessel design parameters				
Principal material	SA533 GrB Cl.1	SA302 GrB Cl.1[a]	SA508 Cl.2	SA508 Cl.2
Design pressure, psig	2500	2500	2500	2500
Design temperature, F	650	650	650	650
Shell OD, in.	171.375	171	171	171
OD across nozzles, in.	249	249	249	249
Overall vessel-closure head height, ft/in.[b]	40/8.875	40/8.75	40/8.75	40/8.75
Core barrel-thermal shield principal material	Type 304 SS	Type 304 SS	Type 304 SS	Type 304 SS

[a] As modified by Code Case 1339.

[b] Over cladding and instrumentation nozzles.

Table 1-3. Comparison of Plant Parameters for Host Reactor Three Mile Island 2 and Guest Reactor Three Mile Island 1

Plant parameters	Host reactor: TMI-2	Guest reactor: TMI-1
Design heat output (core), MWt	2772	2568
Design overpower, % design power	112	112
System pressure (nominal), psia	2200	2200
Coolant flow rate, 10^{-6} lbm/gpm	137.8/369,000	131.3/352,000
Coolant temperatures, F		
Nominal inlet	557	554
Avg rise in vessel	51	50
Avg in vessel	582	579
No. of fuel assemblies	177	177
Type of fuel assemblies	Mark B (15×15)	Mark B (15×15)
Core barrel ID/OD, in.	141/145	141/145
Thermal shield ID/OD, in.	147/151	147/151
Core structural characteristics, in.		
Core equivalent diameter	128.9	128.9
Core active fuel height	144	144
Reflector thickness, composition, in.		
Top (water + steel)	12	12
Bottom (water + steel)	12	12
Side (water + steel)	18	18
Reactor vessel design parameters		
Principal material	SA533, Gr B, Cl.1	SA302, Gr B modified
Design pressure, psig	2500	2500
Design temperature, F	650	650
Shell ID, in.	171	171
OD across nozzles, in.	249	249
Overall vessel-closure head height[a], ft/in.	40/8.75	40/8.75
Core barrel-thermal shield principal material	Type 304 SS	Type 304 SS

[a] Over cladding and instrumentation nozzles.

Figure 1-1. Reactor Vessel Arrangement Showing Original Surveillance Capsule Holder Tube Location

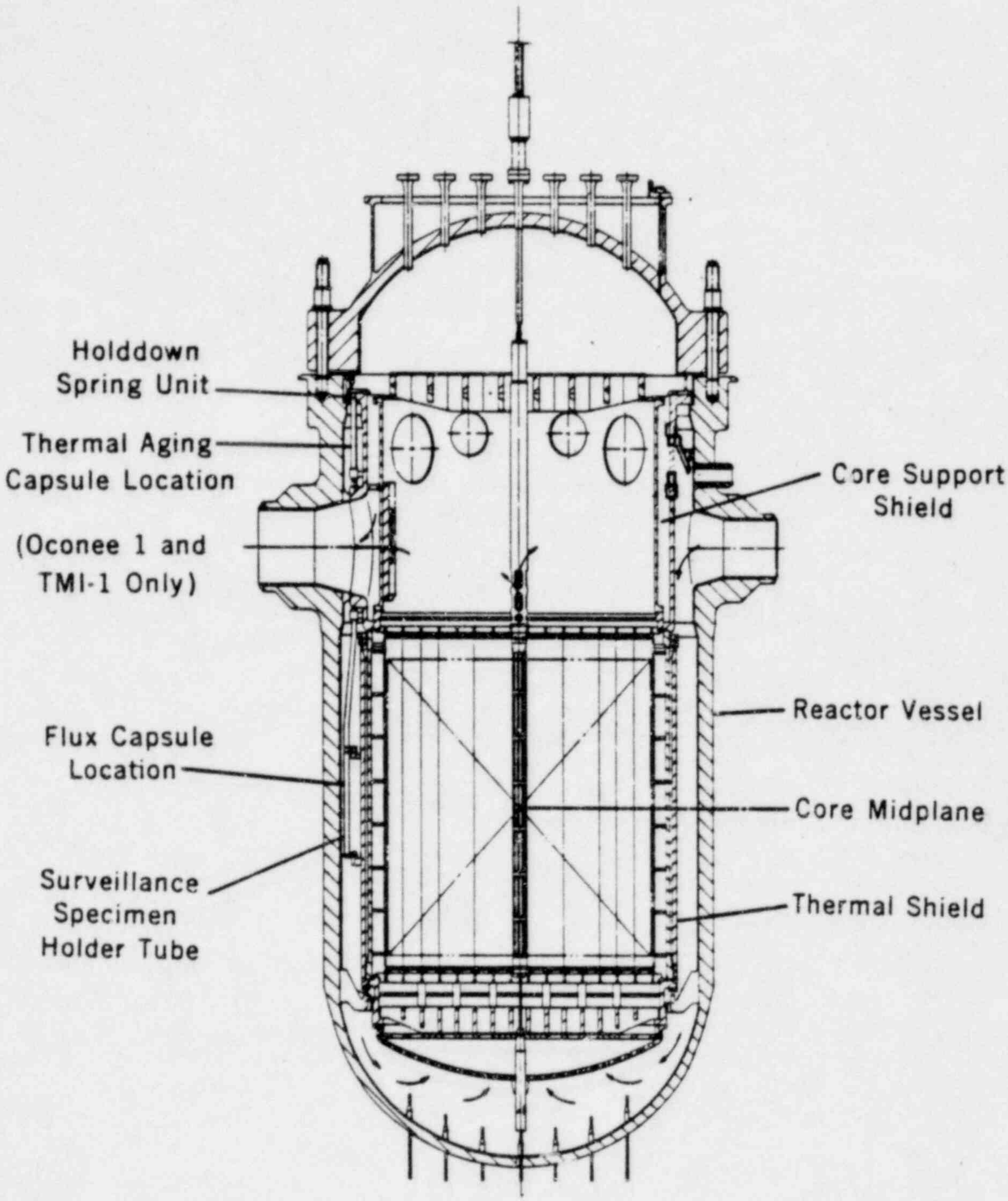
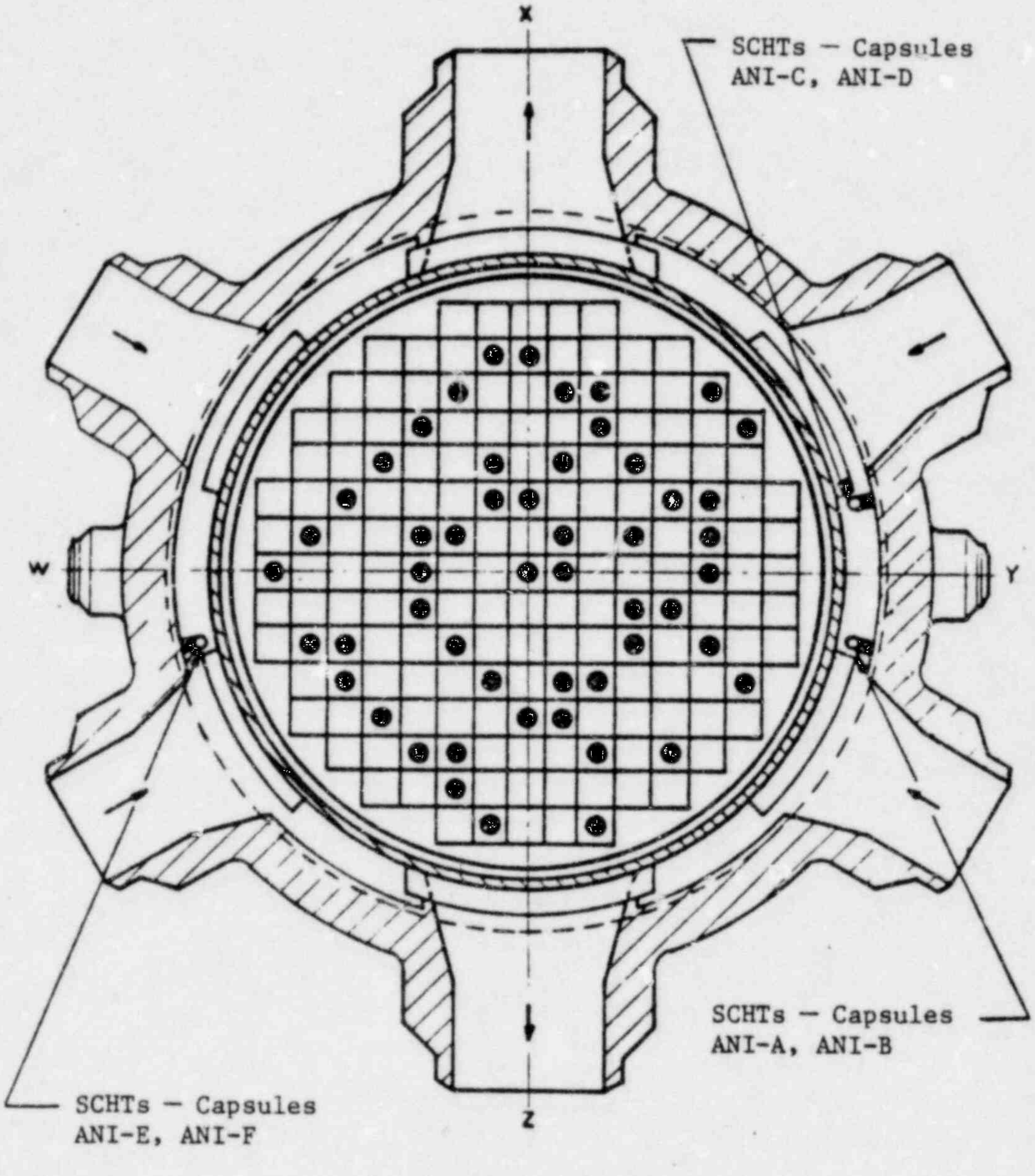


Figure 1-2. Reactor Vessel Cross Section Showing Original Surveillance Capsule Locations



2. BACKGROUND

It became apparent in the late 1950's that the neutron embrittlement sensitivity of steels and weldments used in reactor vessels vary significantly from steel to steel, heat to heat, and even weld to weld. Accordingly, a research and development program to address this phenomenon was initiated and in 1961, guidelines (ASTM E-185) for establishing a reactor vessel surveillance program (RVSP) were adopted — "Standard Practice for Conducting Surveillance Tests for Light Water-Cooled Nuclear Power Reactor Vessels."* Both commercial power plants and test reactors# in several national laboratories were used in determining the metallurgical parameters controlling the sensitivity of commonly used reactor vessel steels to neutron irradiation.[1-4] Selected specimens of these steels were encased in specially designed capsules. The encapsulated samples were then placed in test reactors where the neutron flux and temperature experienced were comparable to those seen by the reactor vessel wall. In this way, embrittlement (characteristics) could be assessed and predicted before it occurred in operating reactor vessels.

In 1970, the copper and phosphorus contents in pressure vessel materials were identified by the Naval Research Laboratory in cooperation with The Babcock & Wilcox Company (B&W) as the principal parameter(s) influencing neutron embrittlement sensitivity.[5-8] However, despite research by these and other organizations, the effects of differing neutron spectra, radiation rates, and temperature were not clearly delineated. Also, information on particular types of materials, such as submerged-arc weld metal and SA508 Class 2 forgings, was quite limited.

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*This document was revised in 1966, 1970, 1973, and 1979 to reflect knowledge gained. These revisions are compared in Table 2-1.

#Specimens were irradiated at flux rates 100 to 200 times higher than those observed at the reactor vessel wall of a commercial power plant, and at nominal reactor vessel temperatures below 550F.

In 1973, in a concerted effort to acquire the necessary information for licensing PWRs and to standardize the existing industry RVSPs, Appendix H of 10 CFR 50 ("Reactor Vessel Materials Surveillance Program Requirements") made the RVSP (complying with ASTM E-185-73) mandatory. Up to this point, the data gathered from the RVSPs had been diversified because the ASTM E-185 requirements were broadly defined and gave considerable latitude to the designer.

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'a	Specimen orientation	Index temp for measuring ΔT	Capsule with-drawal schedule	Dosimetry requirements	Temperature monitor requirements	Special requirements and recommendations
1966	1) Base metal with the highest trans temp. 2) Any weld metal 3) HAZ metal	13 or more	8 Charpy 2 tension	15 Charpy 3 tension	Parallel to major working direc'n	Charpy energy fix temp as identified by NDT unirr. drop wt tests (normally 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	1) Base metal with the highest trans temp. 2) Representative weld metal (same wire or rod & flux as one of the high-flux region welds) 3) HAZ of base metal	3 or more	8 Charpy 2 tension	15 Charpy 3 tension	Parallel to major working direc'n	Same as above	One corresponding to 30% of design life; one to 100% life; others not specified	Determined per ASTM E-261; Fe & unshielded Co dosimeters to be included; Ni-Cd-shielded Co & Cu suggested also	Same as above	1) Desirable to include correlation monitor spec's. 2) Thermal control specimens desirable 3) Consider inserting capsules at later time 4) Test material chemistry shall be determined
1973 Case B	Detailed selection procedure (beltline reg.) 1) Base metal 2) Weld metal (same wire or rod and type of flux as one of the welds) 3) HAZ of base metal	5	12 Charpy 2 tension	15 Charpy 3 tension	Normal to major working direc'n	Measured at 30 ft-lb	First 3 capsules withdrawn at specific times; 4th & 5th capsules standby	Determined per ASTM E-261; Fe & unshielded Co dosimeter required	Same as above	1) Capsule neutron lead shall not exceed three 2) Chemistry (including Cu,P,S,V) of test materials shall be determined 3) Consider inserting capsules at later time
Prop 1979 Case ΔT > 200F	General guidance for selection controlling materials 1) Controlling base metal 2) Controlling weld metal (same heat of weld wire and lot of flux as beltline region controlling weld 3) HAZ of base metal	5	12 Charpy 3 tension & fracture mechanics	24 Charpy 3 tension & fracture mechanics	Normal to major working direc'n	ΔT @ 30 ft-lb + RT_{NDT} ΔT @ 50 ft-lb & ΔT @ 35 mils for information only	First 4 capsules withdrawn at specific times; 5th 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 shall be between 1 and 3 3) Complete chemistry of test materials shall be determined 4) Add'l fracture toughness specimen per ASTM E-636 shall be included in special cases 5) Capsule and attachment design shall permit insertion of replacement capsules 6) Accelerated capsules optional 7) Test equipment shall be calibrated

3. INTEGRATED REACTOR VESSEL SURVEILLANCE PROGRAM

The integrated surveillance program is more than a combination of nine separate projects and the resultant sharing of irradiation sites. 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 licenseability of the participating reactor pressure vessels.

The integrated reactor vessel surveillance program correlates data from both plant-specific and test reactor surveillance monitoring. However, since the test reactor irradiations are not performed in B&W 177-fuel assembly operating reactors, the following discussions are limited to the power reactor program, which comprises two principal parts. The first is the continuation of the plant-specific surveillance programs which monitor the irradiation damage to selected materials, as originally planned. The capsules contain samples of plate or forging material and heat-affected-zone (HAZ) material from the vessel beltline as well as weld metal; thus, this part of the program will continue to monitor the long-term effects of neutron irradiation on the reactor materials and will be the basis for the plant-specific fluence analysis.

The second part of the program consists of a series of specially designed capsules, (research capsule program) 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 chemistry and low initial Charpy upper shelf energies. These test capsules contain specimens primarily for obtaining fracture toughness properties of individual weld metals and are located in high-lead-factor irradiation holder tubes so that the needed data will be provided in a relatively short time. The data from these capsules will be compared with data obtained on the same material be various test reactor research programs. This comparison will permit evaluation of the effects of flux density and neutron energy spectrum on the irradiation damage to these materials.

The integrated reactor vessel surveillance program (RVSP) complies with ASTM E185-73 and also addresses the additional requirements of Appendix H, which affect the individual RVSPs of the plant involved. These additional changes are described below.

1. The definition of the beltline region was changed to include more of the shell course material above and below the effective height of the fuel element assemblies.
2. The transition temperature adjustment (ΔRT_{NDT}) for beltline region materials is based on the temperature shift of the Charpy V-notch curve measured at the 50 ft-lb level or at the 35-mil lateral expansion. The property exhibiting the greater shift is used to define the adjustment in reference temperature.
3. The pressure-temperature operational limitations of the reactor vessel are established in accordance with Appendix G to Section III of the ASME Code. The highest adjusted RT_{NDT} and the lowest upper shelf energy level of all the beltline region materials, as determined by the RVSP, are used to calculate these operating limitations.
4. Since the test materials in the capsules of the plant-specific RVSP were not selected in accordance with ASTM E 185-73, the data to be generated are not necessarily applicable to any specific plant. That is, the materials monitored by the RVSP are usually not the materials judged by Appendix H to be most likely to 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 from which to determine the probable values and predict the irradiation behavior of those welds for which there are no specific data.

The two parts of the power reactor program are discussed separately in sections 3.1 and 3.2.

3.1. Plant-Specified Surveillance Program

The plant-specific surveillance program includes the irradiation of (1) the surveillance capsules removed from reactors without capsule holder tubes and (2) the capsules from those plants in which the irradiations are being conducted.

Each plant participating in the integrated surveillance program has a plant-specific surveillance program which was designed to meet the requirements of the NRC and ASTM E-185 at the time the programs were developed. The topical reports describing the appropriate RVSPs for B&W's 177-fuel assembly (177-FA) units participating in the integrated surveillance program are as follows:

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

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

Each RVSP consists of three holder tubes, each of which contains two capsules. Four of these six capsules are the prime data-collecting capsules, while the other two are considered "standby" capsules. The prime capsules are withdrawn at designated time intervals so that the data collected correspond to irradiation levels ranging from a low level to that equal to the vessel inner surface at end of life (EOL). The standby capsules provide any necessary additional data late in the operating life of the plant. Table 3-1 is a typical withdrawal schedule.

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. Three basic types of specimens, either singly or in combination, are placed in these capsules: Charpy, tensile, and compact fracture. (Appendix B describes the specimens in detail.) The Charpy V-notch specimen is 0.394 inch square, 2.165 inches long, and conforms to ASTM E23-72. The tensile specimen is 4.25 inches long and conforms to ASTM E8-69T. The compact fracture specimen is 0.5 inch thick, 1.25 by 1.20 inches, and conforms to the basic requirements of ASTN E-399.

Spec?men identity is maintained throughout the program by die-stamping the top and bottom of each specimen with an identitifcation code (a combination of six letters and numbers).

In addition to the specimens, each capsule contains dosimeters and temperature monitors. Figure 3-1 and 3-2 show typical capsules and the orientation of their specimens, dosimeters, and temperature monitors. The voids in the capsule are filled with aluminum spacers to minimize movement of the specimens inside and improve heat transfer characteristics. After the specimens and filler blocks are positioned within the capsule, it is purged with an inert gas before the last end cap is welded in place.

As stated previously, the integrated RVSP organizes and evaluates the data accumulated in the individual surveillance programs designed for nine different reactors. Within this common network are three types of surveillance programs (types A, B, and C), in which six capsule types (I-VI) are irradiated. Surveillance program A uses capsule types I and II; program B uses capsule 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 section 3.1.1, while the dosimeters and temperature monitors are discussed in sections 3.1.2 and 3.1.3. The three separate programs (A, B, and C) and the types of capsules (I-VI) are described in section 3.1.4.

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

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. An average heat rate of 0.45 W/cm^3 was used for the design of specimens and holders. This rate was based on a computer analysis of gamma heating from the core and from neutron capture in the internals at a core rated power of 2772 MWt.

Two cases were analyzed. In the first case the holder tube was considered solid; in the second it was considered perforated, with the area of the holes equal to 25% of the surface area of the solid tube. The perforated tube allowed enough coolant to reach the specimen capsules to cool them to within 14F of the temperature of the entering coolant water. Thereforem a perforated design was selected for the holder tube.

The capsules are locked into the holder tube by a removable closure device that subjects the capsules to a compressive load and the holder tube to an equal tensile load. This loading is designed to minimize flow-induced vibration. (The tight inner packing also minimizes flow-induced vibrations within the capsule.) The perforated holder tube also causes the capsule to be subjected to the reactor coolant pressure. Structurally, the capsules are designed to withstand the compressive preload and the external pressure without failure.

The capsule is designed to maintain specimens to within ±25F of the reactor vessel temperature at the 1/4 thickness location.* The heat transfer analysis considers the differences in thermal properties of the materials and the helium-filled gaps between the capsule and the internal components. Conservative maximum temperatures were calculated for each different cross section within the capsule. The coolant temperature serves as the lower bound and is within ±25F of the vessel temperature at 1/4T.

The capsules are placed in the holder tubes (two per tube), which are then 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. (Holders are adjacent to the thermal shield in the integrated program, whereas in other RVSPs they were adjacent to the reactor vessel inside wall. Further, in the original surveillance programs, the self-shielding effect of the specimens was minimized by a design feature that permitted 180-degree rotation of the capsule during refueling. However, since the self-shielding effects have been found to be negligible, the new capsule holder tubes do not permit rotation.)

3.1.2. Dosimeters

Dosimeters are placed in the specimen capsules to determine the actual neutron fluence levels experienced by the specimens. Each capsule contains four dosimeter tubes, each tube accommodating six different dosimeters; the dosimeter types are defined in Table 3-2.

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

Dosimeter tube placement within the capsules is shown in Figures 3-1 and 3-2. The tubes run the length of the specimen stacks so the actual fluence experienced by the specimens can be determined.

3.1.3. Temperature Monitors

A number of low-melting fusible alloy temperature monitors are included in each capsule to determine the maximum temperature during the irradiation exposure. The temperature monitors and their alloy composition and melting temperatures are given in Table 3-3. The locations of the temperature monitors within the capsule are shown in Figures 3-1 and 3-2.

3.1.4. Types of Surveillance Programs and Capsules

As stated previously, there are three types of surveillance programs using six types of capsules in the integrated RVSP. The basic programs and capsule types are described briefly below, and more detailed information is presented in Appendix D. An overview of the program and capsule types is given in Table 3-4.

3.1.4.1. Surveillance Program A

Surveillance program A consists of capsule types I and II; it is described in topical report BAW-10006, Rev. 3. Types I and II are the upper and lower capsules in the holder tube, respectively.

Capsule Type I — Capsule type I contains eight tensile specimens and 36 Charpy specimens. Tensile specimens are prepared from weld metal and base metal A in the transverse direction. Charpy specimens are prepared from weld metal, the heat-affected zone (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 also contains eight tensile specimens and 36 Charpy specimens. Tensile specimens are prepared from the HAZ of heat B in the longitudinal direction and base metal heat B in the longitudinal direction. Charpy specimens are 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.

3.1.4.2. 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 tensile and Charpy specimens, compact tensile specimens 0.5 inch thick are included in capsule type IV. Types III and IV are the upper and lower capsules in the holder tube, respectively. This program is designed for those reactor vessels with marginal material properties.

Capsule Type III — Capsule type III contains four tensile specimens and 54 Charpy specimens. Tensile specimens are prepared for the weld metal and base metal heat A in the transverse direction. Charpy specimens are prepared for the weld metal, HAZ heats A and B in the transverse direction, and correlation monitor plate.

Capsule Type IV — Capsule type IV contains four tensile specimens, 36 Charpy specimens, and eight compact fracture specimens 0.5 inch thick. Tensile specimens are prepared for the weld metal and base metal heat A in the transverse direction. Charpy specimens are prepared for 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 are prepared for the weld metal.

3.1.4.3. 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 are the upper and lower capsules in the holder tube, respectively. This program is for those reactor vessels with controlled chemistry and adequate initial material properties.

Capsule Type V — Capsule type V contains four tensile specimens and 54 Charpy specimens. Tensile specimens are prepared for the weld metal and base metal heat A in the transverse direction. Charpy specimens are prepared for 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 four tensile specimens and 54 Charpy specimens. The tensile specimens are prepared from the weld metal and base metal A in the transverse direction. Charpy specimens are prepared for 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. Research Capsule Program

3.2.1. Introduction

The research capsule program is designed to evaluate eight weld metals (W1, W2, W3, W4, W5, W6, W8, and W9) contained in six research capsules. The capsules are irradiated in the three host reactors (section 2) of the integrated RVSP. These host reactors (each containing two research capsules) are Three Mile Island Unit 2, Crystal River Unit 3, and Davis Besse Unit 1. The six research capsules are labeled TMI2-LG1, TMI2-LG2, CR3-LG1, CR3-LG2, DB1-LG1, and DB1-LG2. The first letters and the first number of these labels are the initials of the host reactor. The letters LG are an abbreviation for "large," and the last number identifies the two capsules at each reactor site. The research capsules are considerably larger than the original plant-specific capsules irradiated at the same reactor sites. The original capsules have an inside diameter of 2.0 inches and an effective length of approximately 18.3 inches. The large research capsules have an inside diameter of 2.5 inches and an effective length of 22.0 inches. Each research capsule contains Charpy, Tensile, and compact fracture specimens from three welds. However, not all the capsules are alike; they have been categorized as types R-1 and R-2. The two capsule designs are shown in Figures 3-3 and 3-4. The type R-2 capsule represents an improved design since it utilizes miniature (Charpy size) tensile specimens. The miniature specimens allow the addition of five more tensile and three more compact fracture specimens per capsule then the original design. In addition, there are small variations between types R-1 and R-2 in terms of the location of the temperature monitors and neutron dosimeters. This section describes the content of each type of capsule as well as the types of specimens.

The TMI-2 capsules are type R-1, and the CR-3 and DB-1 capsules are type R2. Table 3-5 identifies the weld metals irradiated in each capsule as well as the distribution of specimens. The specimens listed as 0.394 TCT, 0.500 TCT, and 0.936 TCT are the compact fracture specimens of 0.394-, 0.500-, and 0.936-inch thickness, respectively. The 0.394 TCT and 0.500 TCT specimens are modifications of ASTM E-399 standard geometry, and the 0.936 TCT specimen is round. The number of Charpy and tensile specimens per weld per capsule is adequate to characterize the toughness and tensile properties for each weld metal and irradiation condition. Other related research programs are expected to generate sufficient information to properly identify the methods (i.e., static versus dynamic) and test temperatures at which these research capsule compact fracture specimens should be tested. The combination of compact fracture specimens is believed to be adequate to confirm the toughness curves. The information generated by the research capsules and the HSST program will enable accurate prediction of the toughness of the reactor vessel welds.

3.2.2. Research Capsule Design

The cylindrical research capsules, like the RVSP capsules described previously, contain Charpy, tensile, and compact fracture specimens as well as dosimeters and temperature monitors. The specimens are placed in stacks and are held in place with aluminum spacers. The cylindrical capsule is the principal characteristic of the B&W design. The unique advantage of the cylindrical capsule is that it allows for capsule replacement and for uniform specimen temperatures. However, the research capsules are larger than the standard capsules — in both length and diameter. The research capsules have a larger diameter because they are used to irradiate relatively thick compact fracture specimens. The original standard capsules will hold only eight 0.5-inch-thick compact fracture specimens. When the research capsules were designed, many uncertainties were associated with the measuring capacity of the smaller compact fracture specimens as well as the required number of specimens needed to fully characterize the fracture properties. The size of the capsule was also determined by the physical constraints within the reactor vessel as well as the restrictions on the specimen metal temperature during irradiation.

The end fittings are wedge-shaped and chrome-plated to minimize surface wear. The material of construction is type 304 stainless steel for both the shell and end filling. Aluminum spacers hold the specimens, dosimeters, and temperature monitors in place and fill the gaps within the capsule. The remaining spaces are filled with helium gas. The capsules are locked in place in a holder tube assembly. The wedge-shaped end fittings are used to position and lock the capsule inside the holder tube. The individual capsules can be withdrawn and replaced with other capsules whenever necessary. The capsules are normally withdrawn during plant refueling. The capsule holder tube assembly is designed to permit remote removal and replacement of the capsules from the fuel handling bridge without removing the plenum assembly. When all the capsules containing specimens have been irradiated to the desirable neutron exposure, empty dummy capsules are installed to fill the available capsule locations in the surveillance specimen holder tubes.

The type R-1 capsules were designed before the R-2's and used the type of tensile specimens found in the standard capsule design of the 177-FA RVSPs. By the time the CR-3 and DB-1 research capsules were designed, it was recognized that the standard sized tensile specimens were not necessary (see Figure C-8). The use of miniature Charpy-sized tensile specimens (see Figure C-9), permitted the inclusion of three additional compact fracture specimens, which are the most important specimens included in the capsules. Also, the use of the miniature specimens permitted the addition of five more tensile specimens. The tensile specimens are also important because it is expected that at the upper shelf temperature the fracture resistance properties of the material will be dependent on the tensile properties.

Each capsule contains specimens from three different weld metals. The weld metals and distribution of specimens per weld are described in Table 3-5 of this report. The size and number of tensile specimens are sufficient to obtain the tensile properties at several temperatures. The number of Charpy V-notch specimens is that recommended by ASTM E-185 as the minimum number required to obtain a full Charpy V-notch data curve. The size and number of compact fracture specimens per weld is adequate for determination of the fracture toughness properties of the corresponding welds (based on current state-of-the-art). The tension, Charpy and compact fracture specimens are described in Appendix C.

Each capsule also contains dosimeters to measure fluence and temperature monitors to measure irradiation temperature. The dosimeters and temperature monitors are described later in this section.

The arrangements of the specimens, dosimeters, and temperature monitors within the capsules are illustrated in Figures 3-3 and 3-4 for capsule types R-1 and R-2.

3.2.3. Holder Tube and Capsule Location

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 so that the midspan elevation of the tube is at the core midplane.

The azimuthal locations of the holder tubes are shown in Figures 3-5, 3-6, and 3-7 for the DB-1, CR-3, and TMI-2 plants, respectively. Table 3-7 provides a list of the locations of the six capsules and their azimuthal locations.

3.2.4. Structural, Hydraulic, and Thermal Characterization of Research Capsules

The capsules are locked into the holder tube by a removable closure device that subjects them to a compressive load and the holder tube to a tensile load. This preloading is designed to minimize flow-induced vibration. (The tight inner packing also minimize flow-induced vibration within the capsule.) As a structural member, the capsules are designed to withstand the compressive preload and the external pressure without failure, although permanent deformation of the capsule cylindrical wall may occur.

The capsule is designed to maintain specimens at temperatures within ±25F of the reactor vessel temperature at the 1/4T vessel wall location. Figure 3-8 illustrates the calculated vessel wall temperature distribution for steady-state normal operation. The temperature gradient is caused by gamma heating and by the fact that the reactor vessel outside wall is insulated. The temperature calculated for the 1/4T vessel wall location is 576F. The maximum specimen temperature is calculated to be 595F and the minimum 554F, which is the temperature of the coolant at steady state. The capsule heat transfer analysis accounts for the differences in the thermal properties of the materials and the helium-filled gaps between the capsule and internal components. The 595F maximum expected specimen metal temperature is for the specimens at the cluster of the Charpy specimen stack.

3.2.5. Dosimetry

Each capsule contains dosimeter tubes, which in turn contain neutron-detecting element wires in sufficient variations and quantities to measure integrated flux, fast neutron spectrum, and thermal neutron spectrum. A variety of neutron detecting elements was chosen in accordance with ASTM Standard Recommended Practice E-482. The dosimeters are distributed throughout the capsule to measure the dosage rates at various locations.

3.2.5.1. Dosimeters

Table 3-8 lists the neutron detecting elements and provides the energy range and shielding required for each element. The gadolinium or aluminum shielding containing the neutron detecting elements is 15 to 50 mils thick (the lower bound) due to the necessity for eliminating thermal neutrons which cause competing reactions and the upper bound to prevent significant absorption of fast neutrons. The neutron detecting elements, along with their shielding, are then stacked in aluminum holder tubes, which may contain from 5 to 10 elements.

3.2.5.2. Dosimeter Locations

Seven sets of dosimeters are distributed throughout the capsule in order to measure the neutron flux at various specimen stack locations. The Charpy and tensile specimen stacks are monitored by three dosimeters; the design is such that two of them are at the 0° and 180° locations in the capsule with respect to the center of the reactor core. The third dosimeter is located in the center of the Charpy and tensile stacks. Finally, two dosimeters are placed through the openings in the round compact fracture specimen stacks. Figures 3-3 and 3-4 show the exact locations of the dosimeters in the capsules.

3.2.5.3. Temperature Monitors

Temperature monitors are distributed throughout the capsule to measure specimen temperatures. Each set of temperature monitors contains five low-melting-point elements or eutectic alloys whose melting points range from 580 to 621F. By determining which monitors have melted, the peak temperature at various locations within the capsule is determined.

Melting Point Elements and Eutectic Alloys — Table 3-9 lists the five temperature monitors and their respective melting temperatures. Gap dimensions within the capsule have been sized based on a heat transfer analysis to maintain the specimen temperature within ±25F of the temperature at the reactor vessel 1/4T location. The range of 580 to 621F is adequate to monitor the expected specimen temperatures.

Location of Temperature Monitors — Six temperature monitors are placed in the capsule to measure the specimen temperatures. The locations of these monitors are shown in Figures 3-3 and 3-4 for the types R-1 and R-2 capsules, respectively. Five monitors are used for the Charpy and tensile stacks, and one monitor is placed in the round compact fracture specimen stack.

3.2.6. Insertion and Withdrawal Schedules

The research capsules are incorporated in the surveillance capsule insertion and withdrawal schedules of the reactors in the program.

3.2.7. Unirradiated Control Data

The unirradiated baseline data needed to support the evaluation of the irradiated capsule data from the research capsules will be obtained from two sources. The primary sources for these data are sets of specimens that have been prepared from the same weld metal used in the research capsules. These sets of specimens are similar to those included in the capsules but of a longer quantity, so that a better data base can be established. The type and number of specimens of each material are described in Table 3-9.

Some material in excess of the needs of the program was donated to the Heavy Section Steel Technology Program to obtain test reactor irradiation data. Since this program would be obtaining baseline unirradiated data of the same type as needed by the Research Capsule 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 research capsule program are identified in Table 3-10.

3.3. Irradiation Schedule

The capsule irradiation schedule is important to the integrated RVSP because it is not physically possible to irradiate all the capsules simultaneously. Therefore, the schedule must ensure that each participating plant will have capsules being irradiated and removed that lead the irradiation damage the reactor vessel is experiencing. To ensure that the capsules lead their respective plants, the lead factors of the holder tubes are greater than normally permitted, but since most of the reactors have had at least one capsule withdrawn prior to initiation of the integrated RVSP, this is a basis for evaluating any abnormal behavior that may be attributable to the higher lead factor.

Figures 3-5, 3-6, and 3-7 show the locations of the holder tubes in the various plants along with the lead factors for the holder tubes. Tables 3-11, 3-12, and 3-13 define the insertion and withdrawal schedules for each host reactor. These schedules are based on insertions and withdrawals only during refueling outages since the radiation levels for individual capsules are not defined precisely. The schedules include the capsules of both the guest and host reactors and the research capsules.

Table 3-1. Typical Withdrawal Schedule for Six-Capsule Program

Capsule	Purpose of withdrawal
First capsule	When the highest predicted RT_{NDT} shift of all the capsule materials is approximately 50F
Second capsule	When the capsule's accumulated neutron fluence (E > 1 Mev) corresponds to an intermediate value between those of the first and third capsules
Third capsule	When the capsule's accumulated fluence (E > 1 Mev) corresponds to that at the 1/4T reactor vessel wall location at approximately the end of vessel's design service life
Fourth capsule	When the capsule's accumulated neutron fluence (E > 1 Mev) corresponds to that of the reactor vessel inner wall location at approximately the end of the vessel's design service life
Fifth and sixth capsules	Standby

Table 3-2. Surveillance Capsule Dosimeters

Dosimeter	Eff cross section thresh. energy	Counting technique
Co-Al	Thermal	γ - 5.3 yr ^{60}Co
^{54}Fe	2.5 MeV	γ - 314 d ^{54}Mn
Cd-shielded ^{238}U	1.1 MeV	Appropriate fission products
Cd-shielded ^{237}Np	0.5 MeV	Appropriate fission products
Cd-shielded Co	0.5 eV	γ - 5.3 yr ^{60}Co
Cd-shielded Ni	2.3 MeV	γ - 71 d ^{58}Ni

Table 3-3. Temperature Monitor Wires

Approximate melting point, F	Reference materials
580	97.5% Pb, 2.5% Ag
588	97.5% Pb, 1.5% Ag, 1.0% Sn
598	98.8% Cd, 1.2% Cu
610	100% Cd
621	100% Pb

Note: The melting point of each alloy heat or batch shall be verified by the fabricator from its final form and reported to the system designer.

Table 3-4. Reactor Vessel Surveillance Program — Detailed Summary

Oconee Unit 1

Capsule ID	Capsule Type	Table of mat specs	Date tested
A	I	B-1	--
B	II	B-3	--
C	I	B-3	--
D	II	B-3	--
E	I	B-3	June '76
F	II	B-3	July '75

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Oconee Unit 2

Capsule ID	Capsule Type	Table of mat specs	Date tested
A	I	B-2	Nov '81
B	II	B-2	--
C	I	B-2	June '76
D	II	B-2	--
E	I	B-2	--
F	II	B-2	--

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Three Mile Island Unit 1

Capsule ID	Capsule Type	Table of mat specs	Date tested
A	I	B-4	--
B	II	B-4	--
C	I	B-4	--
D	II	B-4	--
E	I	B-4	June '76
F	II	B-4	--

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Three Mile Island Unit 2

Capsule ID	Capsule Type	Table of mat specs	Date tested
A	III	B-5	--
B	IV	B-5	--
C	III	B-5	--
D	IV	B-5	--
E	III	B-5	--
F	IV	B-5	--

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Crystal River 3

Capsule ID	Capsule Type	Table of mat specs	Date tested
A	III	B-7	--
B	IV	B-7	June '81
C	III	B-7	--
D	IV	B-7	--
E	III	B-7	--
F	IV	B-7	--

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Arkansas Nuclear One Unit 1

Capsule ID	Capsule Type	Table of mat specs	Date tested
A	I	B-6	--
B	II	B-6	--
C	I	B-6	--
D	II	B-6	--
E	I	B-6	Aug '76
F	II	B-6	--

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Rancho Seco 1

Capsule ID	Capsule Type	Table of mat specs	Date tested
A	III	B-8	--
B	IV	B-8	--
C	III	B-8	--
D	IV	B-8	--
E	III	B-8	--
F	IV	B-8	--

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Davis Besse Unit 1

Capsule ID	Capsule Type	Table of mat specs	Date tested
A	III	B-9	--
B	IV	B-9	--
C	III	B-9	--
D	IV	B-9	--
E	III	B-9	--
F	IV	B-9	--

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Oconee Unit 3

Capsule ID	Capsule Type	Table of mat specs	Date tested
A	V	B-3	July '76
B	VI	B-3	Oct '81
C	V	B-3	--
D	VI	B-3	--
E	V	B-3	--
F	VI	B-3	--

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

Table 3-5. Research Capsules — Material and Specimens per Capsule

Weld metal ID per capsule	Tensile	Charpy	0.394 TCT	0.500 TCT	0.936 TRCT
Capsule TMI2-LG1					
W1	3	12	2	4	3
W4	2	12	2	4	3
W5	2	12	2	4	3
Capsule TMI2-LG2					
W4	2	12	2	4	3
W5	2	12	2	4	3
W8	3	12	2	4	3
Capsule CR3-LG1					
W1	4	12	2	4	4
W6	4	12	2	4	4
W8	4	12	2	4	4
Capsule CR3-LG2					
W1	4	12	2	4	4
W3	4	12	2	4	4
W6	4	12	2	4	4
Capsule DB1-LG1					
W1	4	12	2	4	4
W2	4	12	2	4	4
W9	4	12	2	4	4
Capsule DB1-LG2					
W1	4	12	2	4	4
W2	4	12	2	4	4
W9	4	12	2	4	4

Table 3-6. Contents of Research Capsules

No. of test specimens	Type R-1	Type R-2
Standard tensile	7	--
Miniature tensile	--	12
Charpy	36	36
0.394 TCT	6	6
0.50 TCT	4	4
0.936 TRCT	9	12

Table 3-7. Azimuthal Location of Power Reactor Research Capsules

Capsule ID	Capsule Type	Reactor	Vertical location with reference to core midplane	Azimuthal location
DB-LG1	R-2	Davis Besse 1	Upper	10.9° off W
DB-LG2	R-2	Davis Besse 1	Lower	26.5° off Z
CR3-LG1	R-2	Crystal River 3	Upper	10.9° off W
CR3-LG2	R-2	Crystal River 3	Lower	10.9° off W
TMI2-LG1	R-1	TMI-2	Upper	10.9° off W
TMI2-LG2	R-1	TMI-2	Lower	26.5° off Z

Table 3-8. Neutron-Detecting Elements

Neutron-sensitive energy range	Neutron-sensitive element	Shield Material	Shield Thickness, in.	Tube Material	Tube Max thickness, in.
0.4 eV	Co coated with Cd	Al	0.015-0.050	Al	0.030
≥0.5 MeV	^{237}Np	Gd	0.015-0.050	Al	0.030
≥1.1 MeV	$^{238}U^{(a)}$	Gd	0.015-0.050	Al	0.030
≥2.3 MeV	$^{58}Ni^{(b)}$	Gd	0.015-0.050	Al	0.030
≥2.5 MeV	^{54}Fe	Gd	0.015-0.050	Al	0.030
≥6.1 MeV	$Cu^{(b)}$	Gd	0.015-0.050	Al	0.030
All levels	Co	Al	0.015-0.050	Al	0.030
Data redundancy verification	Co + Fe	Al	0.015-0.050	Al	0.030

(a) Minimum limit ^{235}U 380 ppm.

(b) Maximum limit Co 0.5 ppm

Table 3-9. Matrix of Control Specimens for Welds W2, W4, W6, W8 Under Power Reactor Program

Power reactor program	Weld metal ident	Tensile	Charpy	0.394 TCT	0.500 TCT	0.936 TRCT	1.000 TCT	2.000 TCT
TMI-2	W4	4	22	5	8	5	--	2
	W8	4	22	5	8	5	--	2
DB-1	W2	4	22	5	8	5	--	2
	W6	4	22	5	8	7	2	--
CR-3	W8	4	22	5	8	5	--	2

Table 3-10. Identification of Programs and Control Specimens for Eight Research Capsule Program Welds

Weld ID	Program
W1	HSST Task 3
W2	Power Reactor Program -- DB-1
W3	HSST Task 3
W4	Power Reactor Program — TMI-2
W5	HSST Tasks 2 and 3
W6	Power Reactor Program — DB-1
W8	Power Reactor Program — TMI-2, CR-3
W9	HSST Task 3

Table 3-11. Surveillance and Research Capsule Insertion and Withdrawal Scheduled for Crystal River Unit 3

Holder tube	Remove	Location in holder tube	Insert
Installed at Initial Fuel Load			
XW		Top	CR3-B
XW		Bottom	CR3-D
End of First Fuel Cycle			
WZ		Top	CR3-LG1
WZ		Bottom	CR3-LG2
ZY		Top	CR3-C
ZY		Bottom	CR3-A
YZ		Top	OCII-A
YZ		Bottom	OCI-A
YX		Top	OCII-E
YX		Bottom	OCIII-D
XW	CR3-B	Top	CR3-E
WX		Top	OCIII-B
WX		Bottom	CR3-F
End of Second Cycle			
YZ	OCII-A	Top	OCI-C
WX	OCIII-B	Top	TMI-1C
End of Fourth Cycle			
WZ	CR3-LG1	Top	OCII-B
End of Fifth Cycle			
ZY	CR3-C	Top	OCIII-B
XW	CR3-D	Bottom	OCIII-C
End of Seventh Cycle			
YZ	OCI-A	Bottom	OCI-B
WX	TMI-1C	Top	Dummy #2

Table 3-11. (Cont'd)

Holder tube	Remove	Location in holder tube	Insert
End of Ninth Cycle			
WZ	OCII-B	Top	OCII-D
YX	OCII-E	Top	OCII-F
YX	OCIII-D	Bottom	OCI-D
End of Tenth Cycle			
WX	Dummy #2	Top	None
WX	CR3-F	Bottom	None
XW	OCIII-C	Bottom	Dummy #1
End of Eleventh Cycle			
WX	CR3-LG2	Bottom	OCIII-F
YZ	OCI-C	Top	Dummy #2
End of Fourteenth Cycle			
ZY	CR3-A	Bottom	Dummy #1
XW	CR3-E	Top	None
XW	Dummy #1	Bottom	None
End of Sixteenth Cycle			
YZ	Dummy #2	Top	None
YZ	OCI-B	Bottom	None
End of Eighteenth Cycle			
ZY	OCIII-E	Top	None
ZY	Dummy #1	Bottom	None
WZ	OCII-D	Top	Dummy #1
YX	OCII-F	Top	None
YZ	OCI-D	Bottom	None
End of Twentieth Cycle			
WZ	Dummy #1	Top	None
WZ	OCIII-F	Bottom	None

Table 3-12. Surveillance and Research Capsule Insertion and Withdrawal Schedule for Davis-Besse Unit 1

Holder tube	Capsule ID	Location in holder tube	
Initial Installation			
WZ	ANI-B	Upper	
WZ	RSI-B	Lower	
ZY	TEI-B	Upper	
ZY	TEI-F	Lower	
YZ	ANI-A	Upper	
YZ	ANI-C	Lower	
YX	RSI-D	Upper	
YX	TEI-C	Lower	
XW	TEI-D	Upper	
XW	RSI-C	Lower	
WX	TEI-A	Upper	
WX	RSI-F	Lower	
At End of First Fuel Cycle			
	Remove		Insert
WZ	ANI-B	Upper	DB-LG1
WZ	RSI-B	Lower	RSI-E
ZY	TEI-F	Lower	DB-LG2
At End of Second Fuel Cycle			
YX	RSI-D	Upper	RSI-A
At End of Third Fuel Cycle			
WZ	DB-LG1	Upper	ANI-D
ZY	TEI-B	Upper	TEI-E
YZ	ANI-A	Upper	ANI-F
At End of Fifth Fuel Cycle			
WX	TEI-A	Upper	Dummy

Table 3-12. (Cont'd)

Holder tube	Capsule ID (remove)	Location in holder tube	Insert
At End of Seventh Fuel Cycle			
YZ	ANI-C	Lower	Dummy
YX	RSI-A	Upper	Dummy
At End of Ninth Fuel Cycle			
YX	TEI-C	Lower	None
YX	Dummy	Upper	None
WX	RSI-F	Lower	None
WX	Dummy	Upper	None
At End of Tenth Fuel Cycle			
WZ	RSI-E	Lower	Dummy
At End of Eleventh Fuel Cycle			
ZY	DB-LG2	Lower	Dummy
YZ	ANI-F	Upper	None
YZ	Dummy	Lower	None
At End of Twelfth Fuel Cycle			
WZ	ANI-D	Upper	None
WZ	Dummy	Lower	None
XW	TEI-D	Upper	None
XW	RSI-C	Lower	None
At End of Sixteenth Fuel Cycle			
ZY	TEI-E	Upper	None
ZY	Dummy	Lower	None

Table 3-13. Surveillance and Research Capsule Insertion and Withdrawal Schedule for Three Mile Island, Unit 2

Holder tube	Remove	Location in holder tube	Insert
At Initial Fuel Load			
WX		Top	TMI2-LG1
WX		Bottom	TMI-2B
XW		Top	TMI-2A
XW		Bottom	TMI-2D
ZY		Top	TMI-1A
ZY		Bottom	TMI2-LG2
End of First Cycle			
WX	TMI2-LG1	Top	TMI-1B
XW	TMI-2A	Top	TMI-2F
End of Third Cycle			
WX	TMI-2B	Bottom	TMI-1D
ZY	TMI-1A	Top	TMI-2C
End of Sixth Cycle			
WX	TMI-1B	Top	TMI-1F
XW	TMI-2D	Bottom	TMI-2E
End of Tenth Cycle			
ZY	TMI2-LG2	Bottom	Dummy #1
End of Eleventh Cycle			
WX	TMI-1D	Bottom	Dummy #2
End of Thirteenth Cycle			
XW	TMI-2F	Top	None
End of Fifteenth Cycle			
WX	TMI-1F	Top	None
WX	Dummy #1	Bottom	None

Table 3-13. (Cont'd)

Holder tube	Remove	Location in holder tube	Insert
End of Sixteenth Cycle			
ZY	TMI-2C	Top	Dummy #1
End of Nineteenth Cycle			
XW	Dummy #2	Top	None
XW	TMI-2E	Bottom	None
End of Twenty-third Cycle			
ZY	Dummy #1	Top	None

Figure 3-1. Surveillance Capsule Arrangement

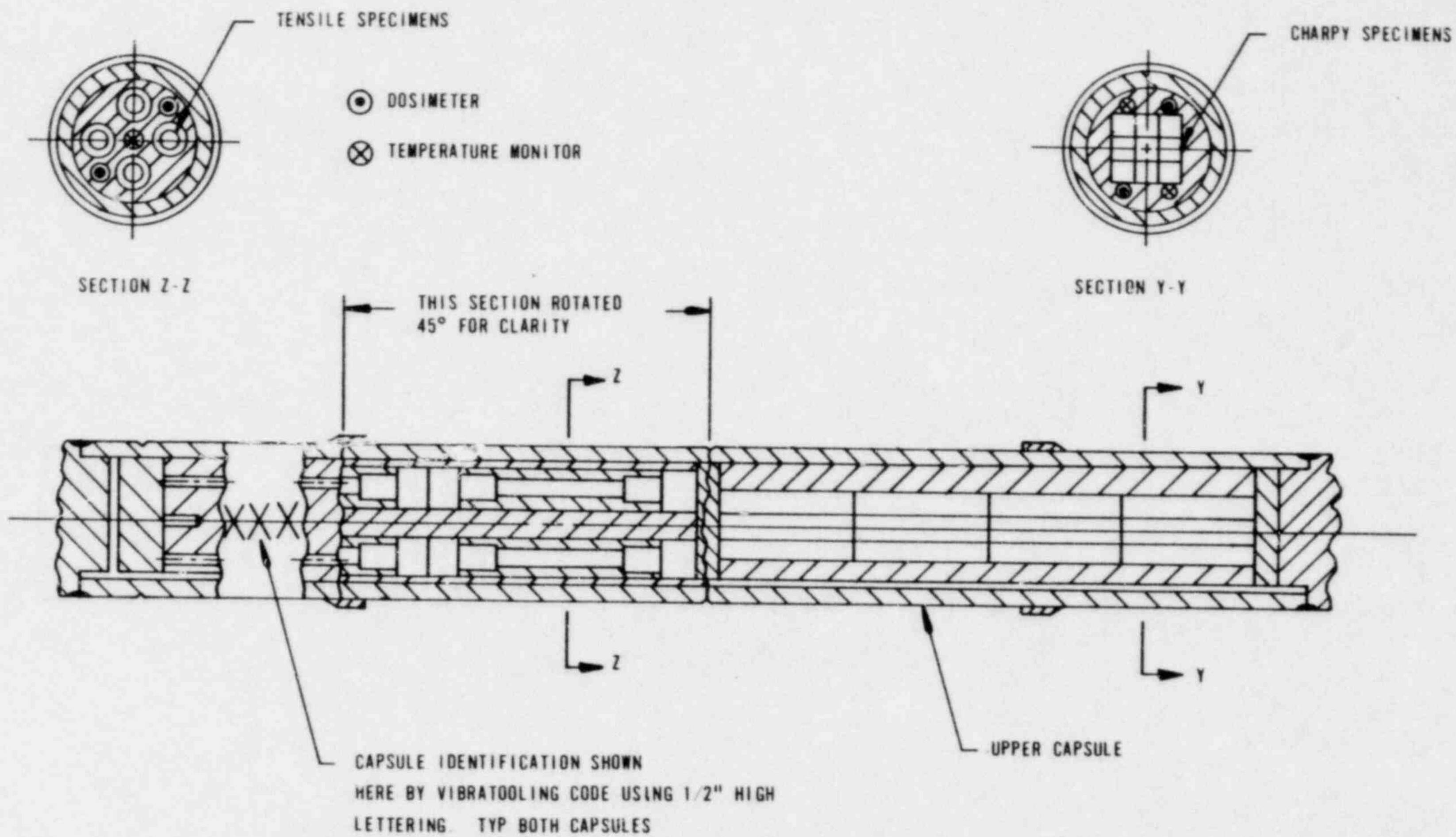


Figure 3-2. Surveillance Capsule Arrangement — With Compact Tension Specimens

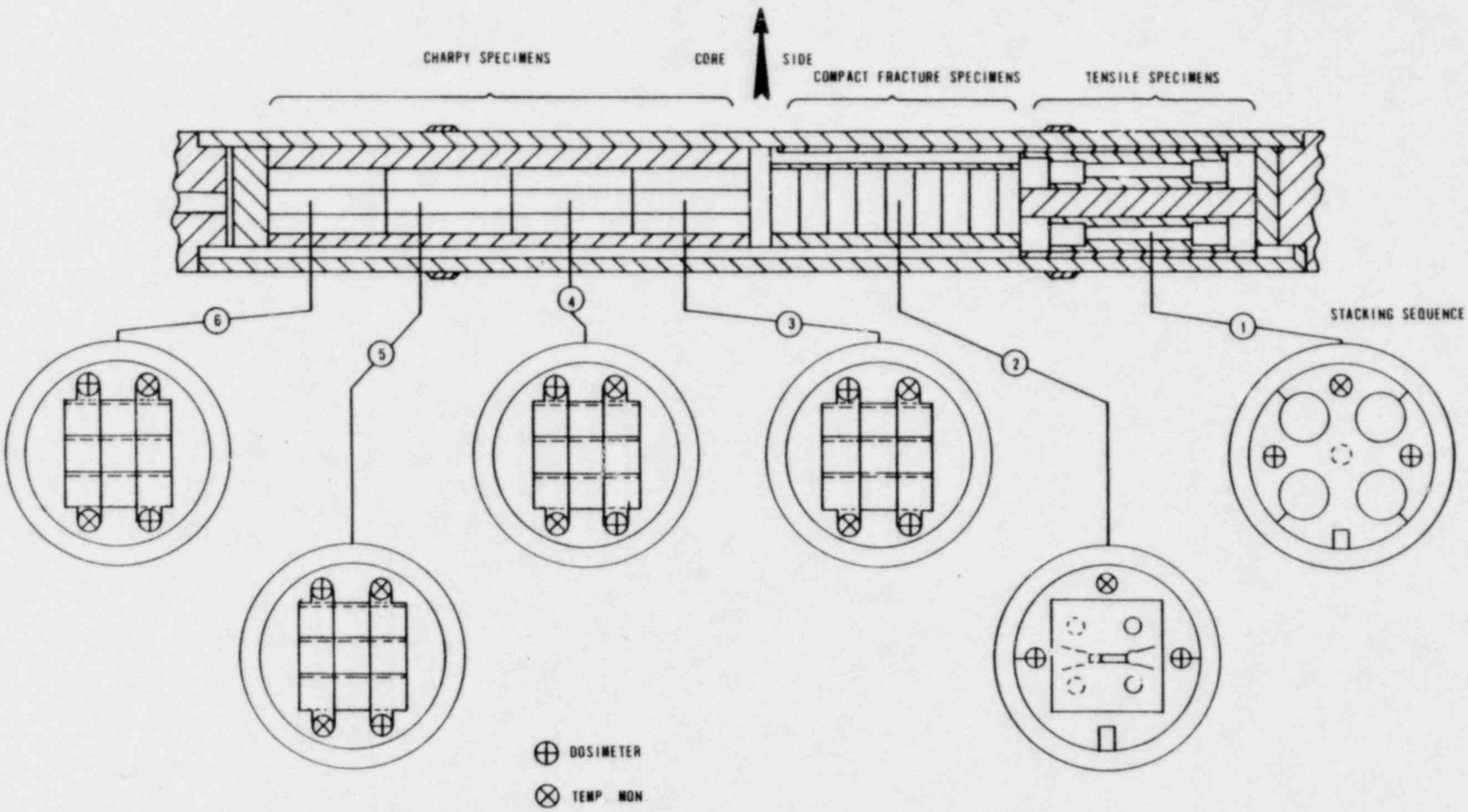
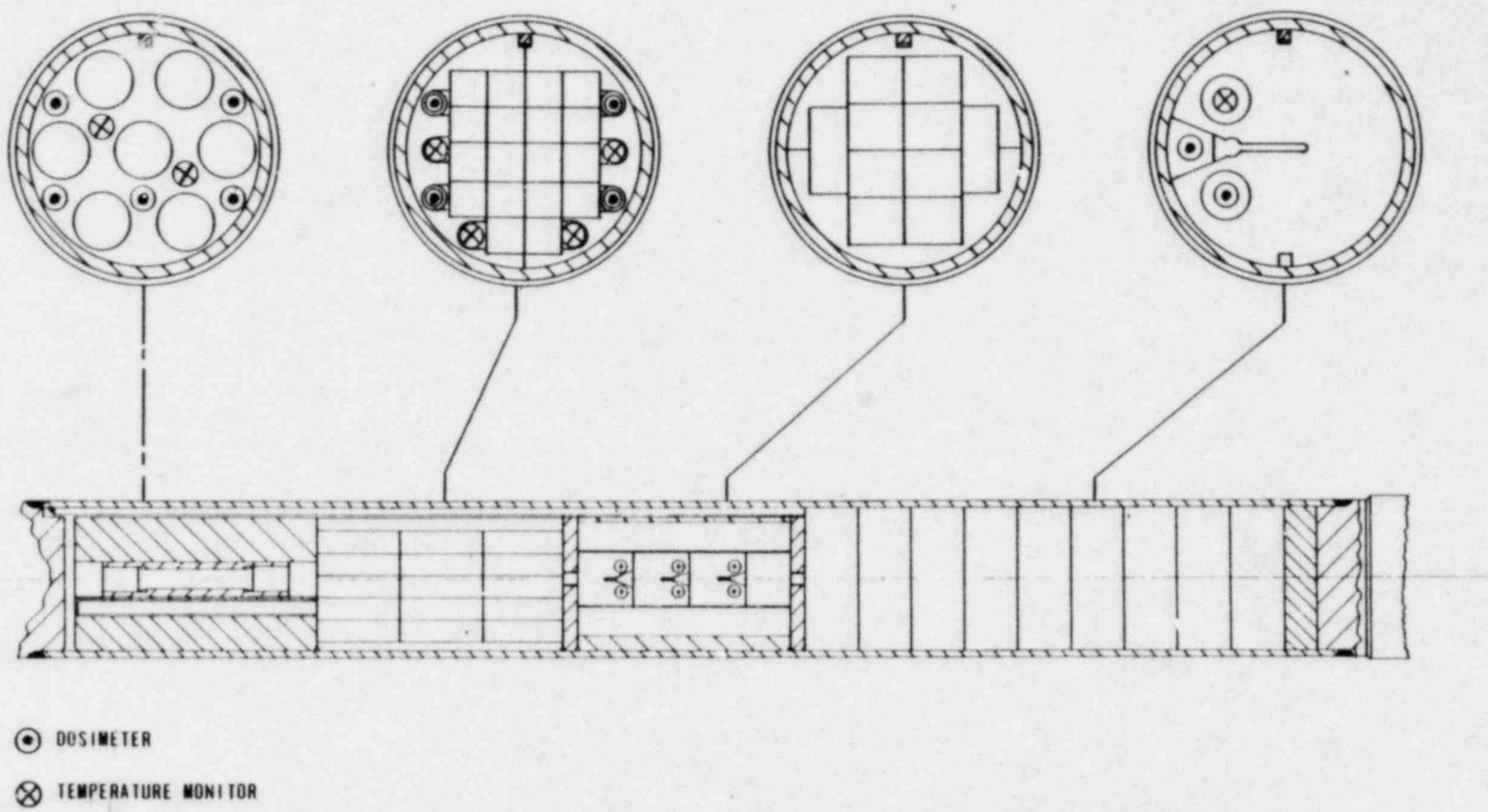


Figure 3-3. Research Capsule, Type R-1



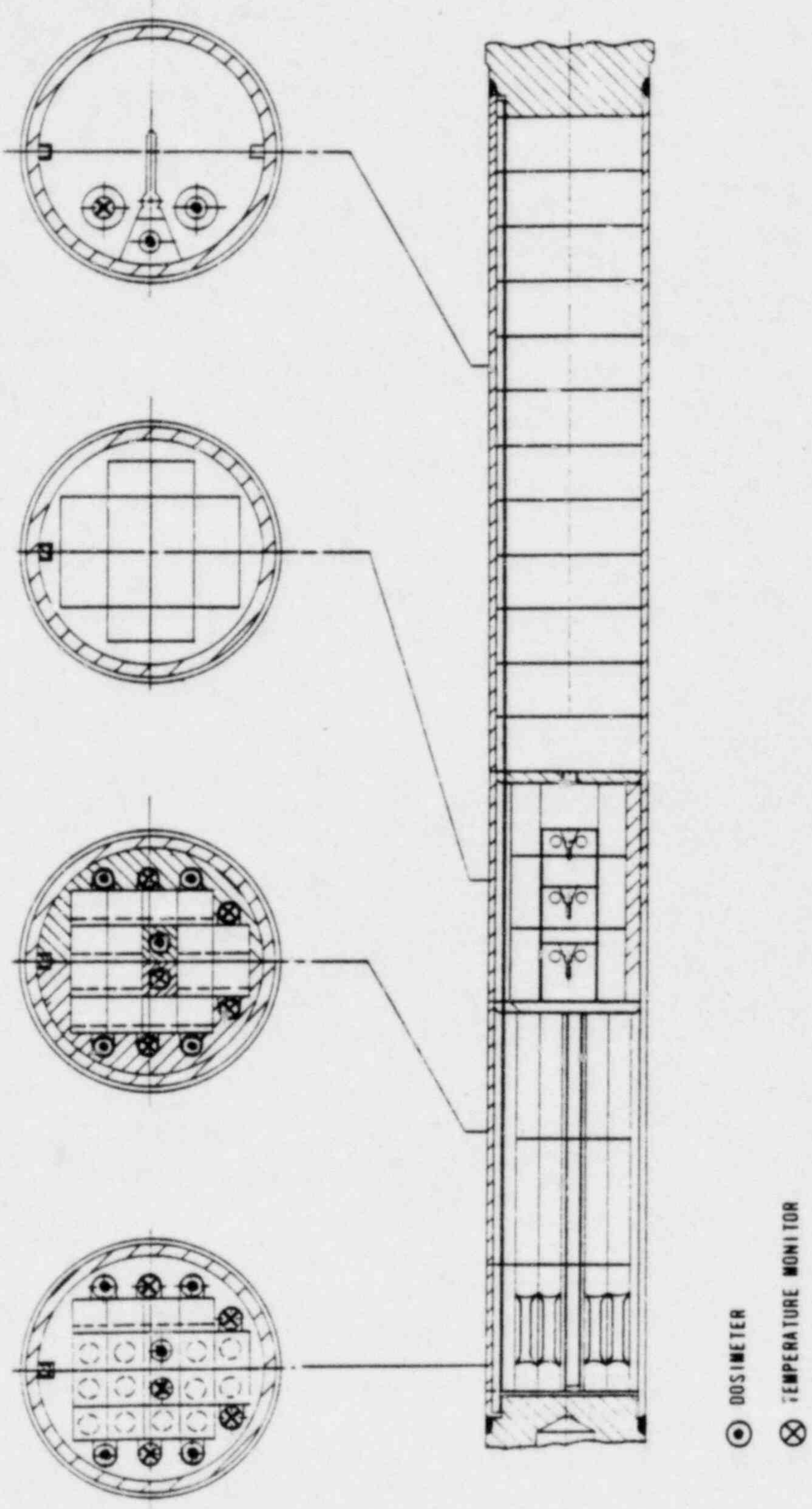


Figure 3-4. Research Capsule, Type R-2

Figure 3-5. Surveillance Capsule Holder Tube Location and Identification for Davis Besse Unit 1

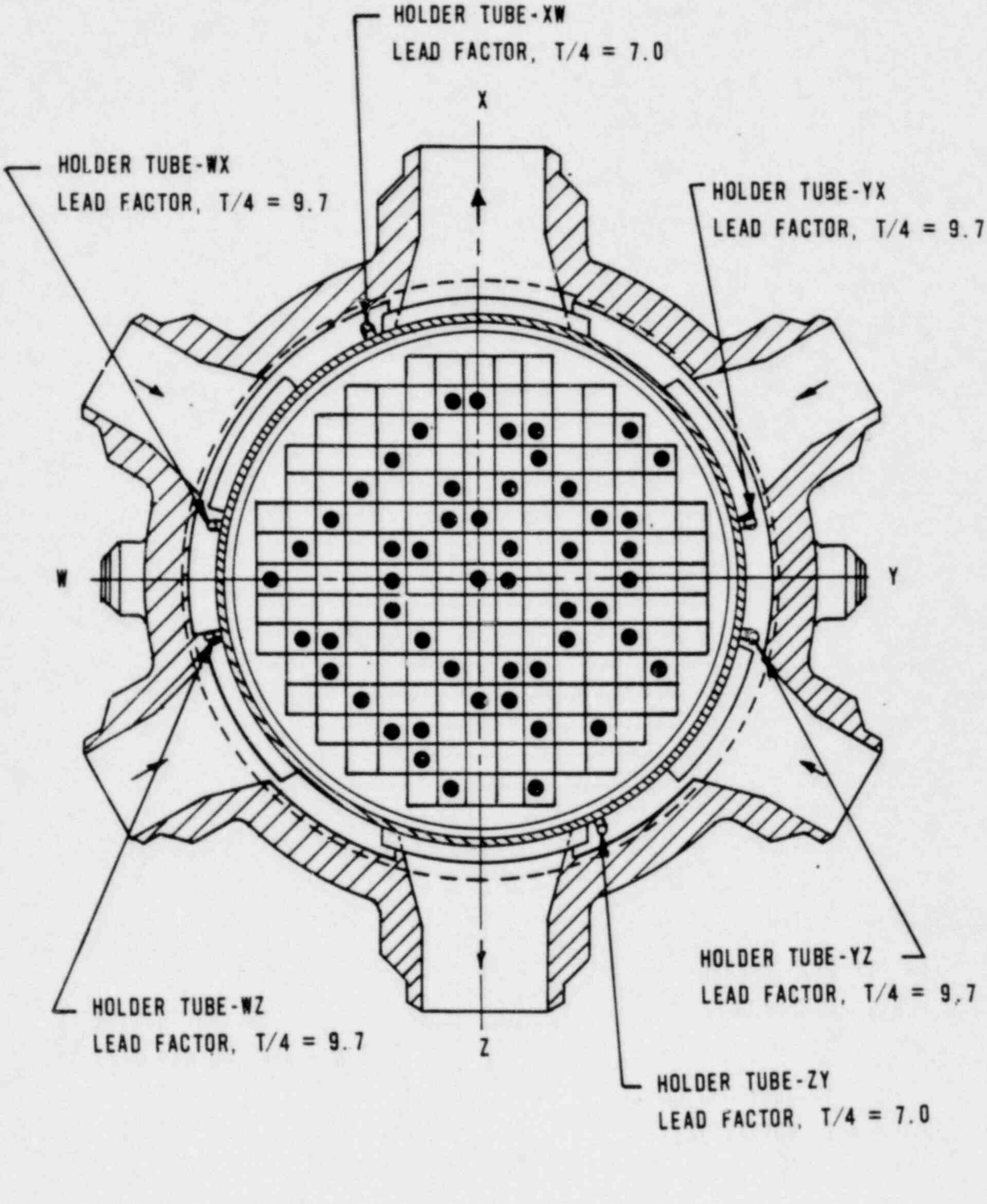


Figure 3-6. Surveillance Capsule Holder Tube Location and Identification for Crystal River Unit 3

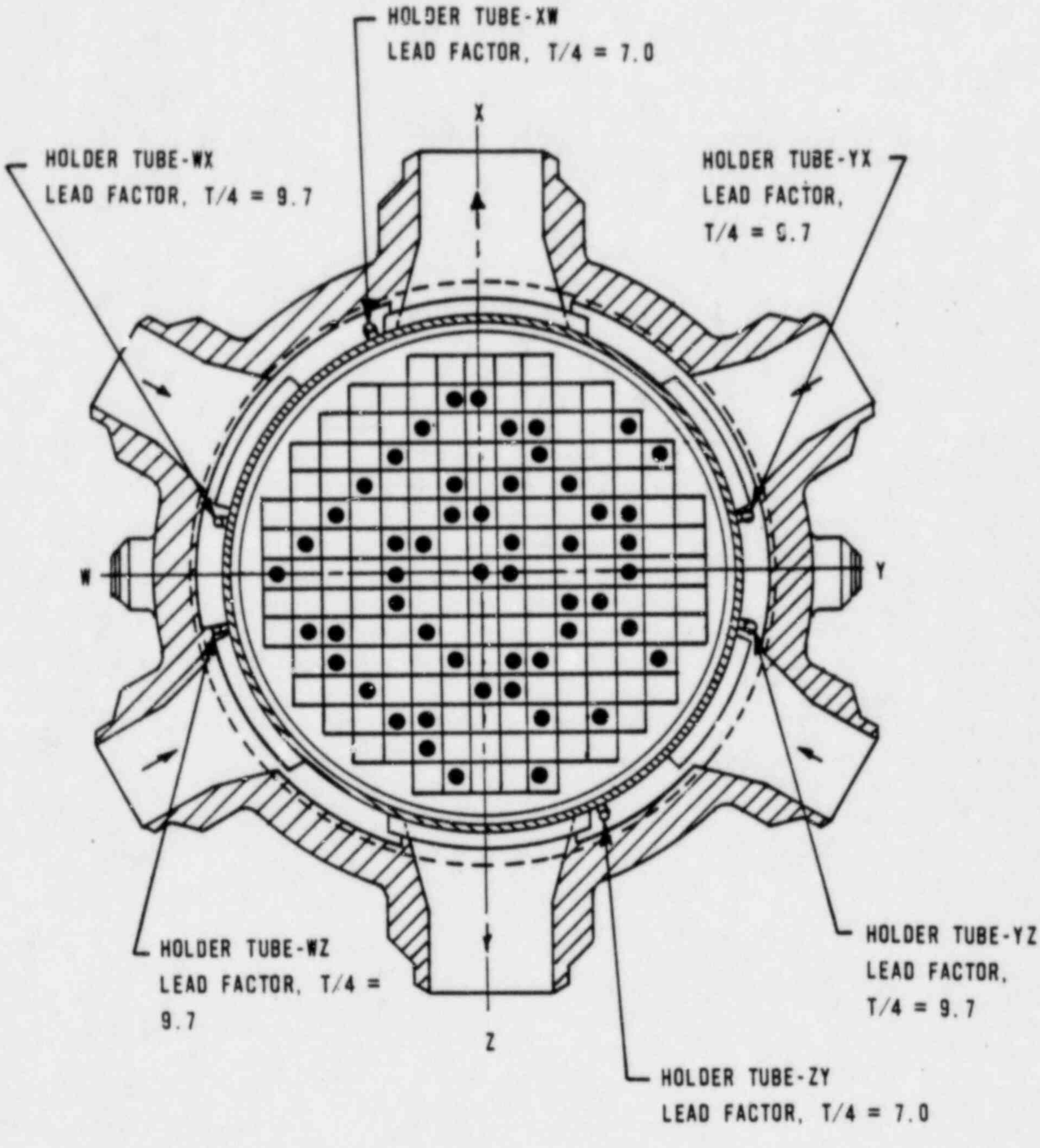


Figure 3-7. Surveillance Capsule Holder Tube Location and Identification for Three Mile Island Unit 2

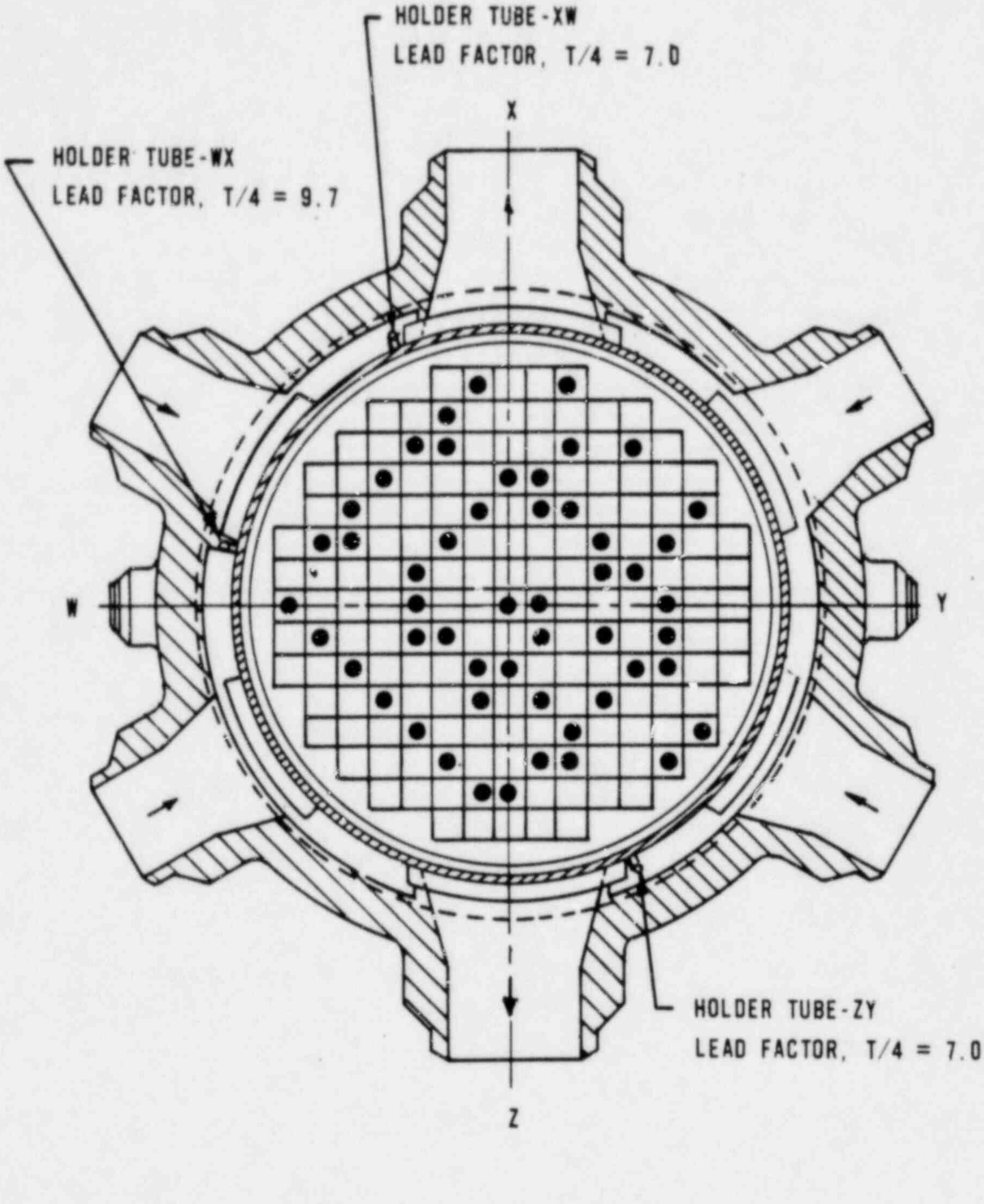
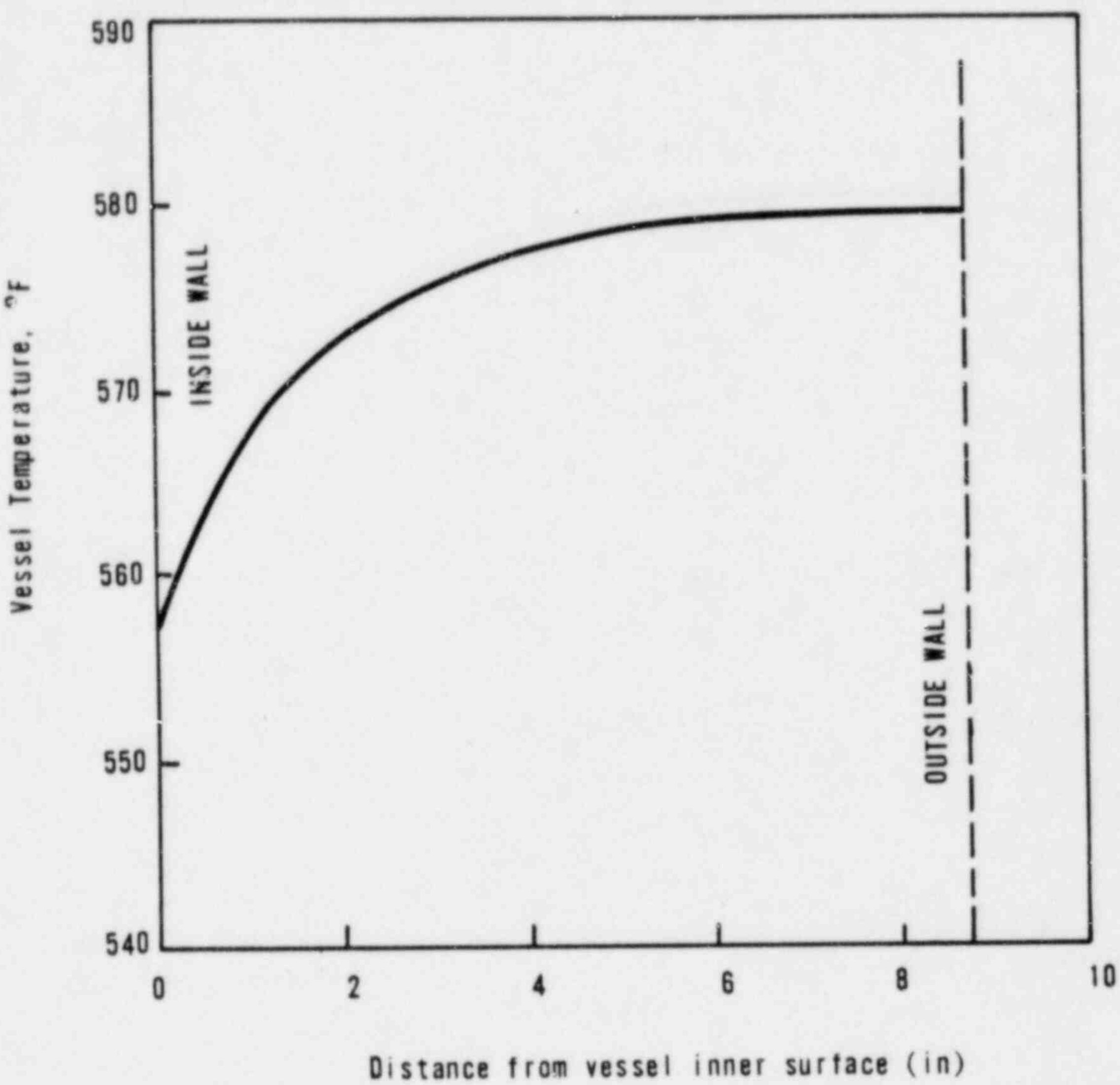


Figure 3-8. 177-FA Reactor Vessel Through-Thickness Temperature Distribution at Steady-State Normal Operation



APPENDIX A

Description and Properties of
RVSP Materials

Table A-1. Oconee 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_v-USE, ft-lb	UTS, ksi	YS, ksi
Base metal A	0.21	1.42	0.015	0.015	0.23	0.50	--	0.49	0.10	0	20	109	87.0	66.0
Base metal B	0.20	1.40	0.012	0.017	0.20	0.63	--	0.50	0.11	20	20	119	88.5	69.0
Weld metal	0.075	1.50	0.024	0.006	0.60	0.58	--	0.51	0.22	-50	0	65	83.0	66.0

Material ID	Heat No.	Spec No.	Supplier	Heat treatment: Austenitizing	Tempering	Stress relief
Base metal A	C3265-1	SA 302 GR.B	Lukens	1600-1650F for 9.75 h, brine quench	1200-1220F for 9.5 h, brine quench	1100-1150F for 40 h, furnace-cooled to 600F
Base metal B	C2800-2	SA 302 GR.B	Lukens	1600-1650F for 9.5 h, brine quench	1200-1225F for 9.5 h, brine quench	1100-1150F for 40 h, furnace-cooled to 600F
Weld metal	WF-112	NA	NA	NA	NA	1100-1150F for 31.0 h, furnace-cooled

Table A-2. Oconee 2 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_v-USE, ft-lb	UTS, ksi	YS, ksi
Base metal A	0.24	0.63	0.006	0.012	0.25	0.75	0.36	0.62	0.04	20	20	133	93.2	72.1
Base metal B	0.21	0.62	0.010	0.010	0.23	0.80	0.39	0.58	0.02	-10	-10	138	71.3	90.6
Weld Metal	0.067	1.58	0.020	0.005	0.56	0.48	0.12	0.33	0.30	-20	10	69	87.5	72.5

Material ID	Heat No.	Spec No.	Supplier	Heat treatment: Austenitizing	Heat treatment: Tempering	Heat treatment: Stress relief
Base metal A	3P-2359	SA 508 CL.2	LADISH	1640F ± 20F held at color 4h, cold water quenched at 1590F ± 20F held at color 4 h, cold water quenched.	1260F ± 20F held at color 10 h, cold water quenched.	1125F ± 25F held at color 40 h, furnace cooled to below 600F
Base metal B	4P-1885	SA 508 CL.2	LADISH	Same as above	Same as above	Same as above
Weld metal	WF-209-1	NA	NA	NA	NA	1100-1150F for 33 h, furnace-cooled

Table A-3. Oconee 3 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	0.24	0.72	0.014	0.012	0.21	0.76	0.34	0.62	0.02	40			86.5	61.3
Base metal B	0.21	0.58	0.011	0.015	0.24	0.73	0.30	0.60	0.01	40			84.8	60.8
Weld metal	0.067	1.58	0.020	0.005	0.56	0.48	0.12	0.33	0.30		60	66	87.5	72.5

Material ID	Heat No.	Spec No.	Supplier	Heat treatment: Austenitizing	Tempering	Stress relief
Base metal A	522194	SA 508 CL.2	LADISH	1640F ± 20F held at color 4 h, cold water quenched. 1590F ± 20F held at color for 4 h, cold water quenched.	1250F ± 20F held at color for 10 h, cold water quenched.	1125F ± 20F, held at color 40 h, furnace cooled to below 600F.
Base metal B	522314K	SA 508 CL.2	LADISH	Same as above	Same as above	Same as above
Weld metal	WF-209-1	NA	NA	NA	NA	1100-1250F for 30 h, furnace-cooled

Table A-4. TMI-1 Description and Properties of Reactor Vessel Surveillance Program Materials

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	1.36	0.010	0.017	0.23	0.59	--	0.51	0.09	10	30	98	92.0	67.0
Base metal B	0.21	1.24	0.010	0.016	0.27	0.55	--	0.47	0.12	-10	20	112	86.0	64.25
Weld metal	0.088	1.50	0.015	0.013	0.45	0.71	0.11	0.33	0.29	-20	-14	81	80.75	66.5

Material ID	Heat No.	Spec No.	Supplier	Heat treatment: Austenitizing	Heat treatment: Tempering	Heat treatment: Stress relief
Base metal A	C2789-2	SA 302 GR.B	LUKENS	1600-1650F for 9.5 h, brine quench 1200-1225F for 9.5 h, brine quench 1600-1650F for 9.5 h, brine quench 1600-1650F for 9.5 h, brine quench 1510-1535F for 5 h, brine quench 1200-1225F for 5 h, brine quench		1100-1150F for 40 h, furnace cooled.
Base metal B	C3307-1	SA 302 GR.B	LUKENS	1600-1650F for 9.5 h, brine quench 1200-1225F for 9.5 h, brine quench 1225-1250F for 9.5 h, brine quench		Same as above
Weld metal	WF-25	NA	NA	NA	NA	1100-1150F for 27.5 h, furnace cooled

Table A-5. TMI-2 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_v-USE, ft-lb	UTS, ksi	YS, ksi
Base metal A	0.22	1.35	0.010	0.015	0.23	0.50	--	0.47	0.12	0	40	100	87.0	63.5
Base metal B	0.22	1.42	0.010	0.015	0.19	0.50	--	0.47	0.14	-10	16	101	91.75	68.5
Weld metal	0.09	1.74	0.13	0.017	0.45	0.61	--	0.46	0.21	-30	19	84	81.0	NA

Material ID	Heat No.	Spec No.	Supplier	Heat treatment		
				Austenitizing	Tempering	Stress relief
Base metal A	C1946-2	SA 533 GR.B, CL. 1	LUKENS	1600-1650F for 9.5 h brine quench.	1150-1200F for 4.5 h, brine quench.	1100-1150F for 40 h, furnace cooled.
Base metal B	C1937-2	SA 533 GR.B, CL. 1	LUKENS	Same as above	1180-1220F for 9.5 h, brine quench. 1200-1225F for 9.5 h, brine quench.	Same as above
Weld metal	WF-182-1	NA	NA	NA	NA	1100-1150F for 48 h, furnace-cooled

Table A-6. ANO-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_v-USE, ft-lb	UTS, ksi	YS, ksi
Base metal A	0.21	1.32	0.010	0.016	0.20	0.52	--	0.57	0.15	10	30	93	92.5	69.0
Base metal B	0.21	1.32	0.010	0.016	0.20	0.52	--	0.57	0.15	20	10	107	89.0	[illegible]5.7
Weld metal	0.065	1.50	0.016	0.008	0.42	0.59	--	0.36	0.19	-20	30	73	82.0	[illegible]

Material ID	Heat No.	Spec No.	Supplier	Heat treatment		
				Austenitizing	Tempering	Stress relief
Base metal A	C5114-1	SA 533 GR.B CL. 1	LUKENS	1650-1700F, held 1 h/in./min and water quenched to 400F.	1200F, held 1 h/in./min and air cooled.	1100-1150F, held 60 h, and furnace cooled within a rate of 35F/h to below 600F.
Base metal B	C5114-2	SA 533 GR.B CL. 1	LUKENS	Same as above	Same as above	Same as above
Weld metal	WF-193	NA	NA	NA	NA	1100-1150F for 27.5 h, furnace-cooled

Table A-7. Crystal River Unit 3 Description and Properties of Reactor Vessel Surveillance Program Materials

Material ID	Chemical Composition, %									Impact properties				
	C	Mn	P	S	Si	Ni	Cr	Mo	Cu	T_{NDT}, F	RT_{NDT}, F	C_v-USE, ft-lb	UTS, ksi	YS, ksi
Base metal A	0.23	1.30	0.008	0.016	0.22	0.54	--	0.55	0.20	-10	20	88	86.7	62.8
Base metal B	0.23	1.30	0.008	0.016	0.22	0.54	--	0.55	0.20	-10	20	88	90.0	66.4
Weld metal	0.067	1.58	0.020	0.005	0.56	0.48	0.12	0.33	0.30	-50	43	63	87.5	72.5

Material ID	Heat No.	Spec No.	Supplier	Heat treatment: Austenitizing	Heat treatment: Tempering	Heat treatment: Stress relief
Base metal A	C4344-1	SA 533 GR.B CL. 1	LUKENS	1650-1700F, held 1 h/in./min water quenched to 400F.	1180F, held 0.5 h/in./min air cooled	1100-1150F held 60 h, furnace cooled to below 600F.
Base metal B	C4344-2	SA 533 GR.B CL. 1	LUKENS	Same as above	Same as above.	Same as above
Weld metal	WF-209-1	NA	NA	NA	NA	1100-1150F for 27 h, furnace-cooled

Table A-8. Rancho Seco 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	0.20	1.33	0.010	0.015	0.19	0.58	--	0.52	0.10	-20	0	92	86.25	65.75
Base metal B	0.20	1.26	0.013	0.017	0.15	0.60	--	0.35	0.12	-10	4	90	85.0	64.0
Weld metal	0.065	1.50	0.016	0.008	0.42	0.59	--	0.36	0.19	-100	15	66	82.0	65.0

Material ID	Heat No.	Spec No.	Supplier	Heat treatment		
				Austenitizing	Tempering	Stress relief
Base metal A	C5070-1	SA 533 GR.B CL. 1	LUKENS	1650-1700F, held 1 h/in./min and water quenched to 400F	1200F, held 0.5 h/in./min and air cooled	60 h at 1100-1150F and furnace cooled below 600F
Base metal B	C5062-1	SA 533 GR.B CL. 1	LUKENS	Same as above	Same as above	Same as above
Weld metal	WF-193	NA	NA	NA	NA	1100-1150F for 27.75 h, furnace-cooled

Table A-9. Davis-Besse 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_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	118	91.6	71.4
Base metal B	0.26	0.68	0.004	0.006	0.30	0.77	0.38	0.64	0.04	20	20	144	89.8	71.5
Weld metal	0.09	1.74	0.013	0.017	0.45	0.61	--	0.46	0.21	-20	2	81	81.0	NA

Material ID	Heat No.	Spec No.	Supplier	Heat treatment		
				Austenitizing	Tempering	Stress relief
Base metal A	5P4056	SA 508 CL.2	LADISH	1640F ± 10F held at color 4 h, cold water quenched. Reaustenitized 1570F ± 10F held at color 4 h cold water quenched	1240F ± 10F held at color 6 h, air cooled	1125F ± 25F held at color 40 h, furnace cooled below 600F
Base metal B	123x244	SA 508 CL.2	LADISH	Same as above	1240F ± 10F held at color 5 h, then air cooled	Same as above
Weld metal	WF-182-1	NA	NA	NA	NA	1100-1150F for 15 h, furnace-cooled

APPENDIX B

Description and Properties of
Research Capsule Program Materials

Table B-1. Chemical Composition and Unirradiated Mechanical Properties of Beltline Region Weld Metals

Ident No.	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		
Weld metal-W1	0.09	1.63	0.018	0.009	0.54	0.59	0.11	0.40	0.42				85.5	69.0
Weld metal-W2	0.075	1.50	0.024	0.006	0.60	0.58	--	0.51	0.22	-50	0	65	83.0	66.0
Weld metal-W3	0.08	1.45	0.016	0.016	0.51	0.59	0.09	0.38	0.21				81.0	NA
Weld metal-W4	0.09	1.53	0.013	0.017	0.53	0.70	0.08	0.42	0.37				88.0	NA
Weld metal-W5	0.09	1.58	0.015	0.016	0.54	0.67	0.09	0.42	0.35					
Weld metal-W6	0.08	1.55	0.021	0.016	0.58	0.60	0.10	0.40	0.22				81.5	64.0
Weld metal-W8	0.09	1.55	0.014	0.015	0.55	0.70	0.08	0.41	0.35					
Weld metal-W9	0.08	1.45	0.011	0.013	0.49	0.59	0.08	0.38	0.27				81.5	67.0

Table B-2. Description of Beltline Region and Surveillance Weld Metals

Ident No.	Filler metal type	Flux type	Welding process	Test qualification post-weld heat treatment
Weld metal-W1	Mn,Mo,Ni	Linde 80	Sub. arc	48 h at 1100-1150F
Weld metal-W2	Mn,Mo,Ni	Linde 80	Sub. arc	48 h at 1100-1150F
Weld metal-W3	Mn,Mo,Ni	Linde 80	Sub. arc	80 h at 1100-1150F
Weld metal-W4	Mn,Mo,Ni	Linde 80	Sub. arc	48 h at 1100-1150F
Weld metal-W5	Mn,Mo,Ni	Linde 80	Sub. arc	
Weld metal-W6	Mn,Mo,Ni	Linde 80	Sub. arc	48 h at 1100-1150F
Weld metal-W8	Mn,Mo,Ni	Linde 80	Sub. arc	
Weld metal-W9	Mn,Mo,Ni	Linde 80	Sub. arc	Eight 6-h cycles at 1100-1150F

APPENDIX C

Description of Surveillance Capsule
Test Specimens — Plant-Specific
and Research Capsules

This appendix describes the tensile, Charpy V-notch, and compact fracture specimens included in the research capsules.

1. Tensile Specimens

Two different sizes of tensile specimens are used in the research capsules; both conform to the requirements of ASTM E8-69T. The Type A research capsules contain the standard size specimens with a gage length of 1.428 inches. The tensile specimens in the other four capsules (Type B) are smaller and fit in a Charpy specimen envelope. The gage length for the miniature tensile specimen is 0.840 inch. Figures C-1 and C-2 illustrate the standard and miniature size tensile specimens, respectively.

2. Charpy V-Notch Specimens

The Charpy V-notch specimens conform to the requirements of ASTM E23-72 and are 0.394 inch square and 2.165 inches long. Figure C-3 describes the Charpy specimen used.

3. Compact Fracture Specimens

There are two configurations of compact fracture specimens: rectangular and round geometry. Two rectangular specimen sizes and one round specimen size were used. The configurations and sizes of specimens are described in the following sections.

3.1. Rectangular Compact Fracture Specimens

The rectangular compact fracture specimens were prepared in accordance with ASTM E 399-74. The specimen geometry is illustrated in Figure C-4. As illustrated in the figure, the specimens were modified for measurement of load versus load line displacement. Two sizes of this type of specimen are included. The specimen sizes (in terms of thickness) are 0.394 and 0.50 inch. The dimensions of these specimens are listed in Table C-1.

3.2. Round Compact Fracture Specimen

When the research capsules were designed, it was recognized that the round compact fracture specimen would make the most efficient use of the capsule volume. Figure C-5 illustrates the round compact fracture specimen with its corresponding dimensions.

3.3. Side-Grooved Specimens

As indicated in section 3.4, the thick specimens for the two research capsules at Crystal River 3 have been side-grooved. These are the 0.936-inch TRCT specimens. The geometry of the side grooves is shown in Figure C-6. 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.[9] In general, the side grooves kept the crack front of the stable crack relatively straight. A large degree of crack tunneling was observed in the testing of non-side-grooved specimen. Shih found that the 25% total side-grooving (12.5% on each side) was sufficient for the tough materials used for his development. Shih tested 12.5, 25, and 50% total side-grooved specimens. The 12.5% side-grooved specimens did not completely suppress the shear tip formation, and the 50% showed higher stable crack growth extension near the tip of the side-grooves than at the center of the specimen. For the materials of this program the 20% side-grooving was selected because it was believed to be adequate and also minimized the reduction of the net section thickness of the specimen (reducing J measuring capacity). The irradiated welds are not expected to be as ductile as the material used by Shih, et al., in their studies. Side-grooving is also expected to affect the slope of the J versus Δa R-curves because of the straightening of the crack front, which affects the determination of Δa. The J-Δa curves determined with side-groove 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.

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

Specimen ID	Dimensions, in. Load line to back face, W	Thickness, B = W/2	Length, 1.25 W	Width, 1.2 W	Load line opening, D	Notch opening, N
0.394 TCT	0.788	0.394	0.985	0.945	0.100	0.064
0.50 TCT	1.000	0.50	1.25	1.20	0.150	0.064
1.00 TCT	2.000	1.00	2.50	2.40		
2.00 TCT	4.00	2.00	5.00	4.80	0.150	0.127

[a]The round compact fracture specimens are illustrated in Figure C-5.

Figure C-1. Standard Size Tensile Specimen — Used on Type A Capsules

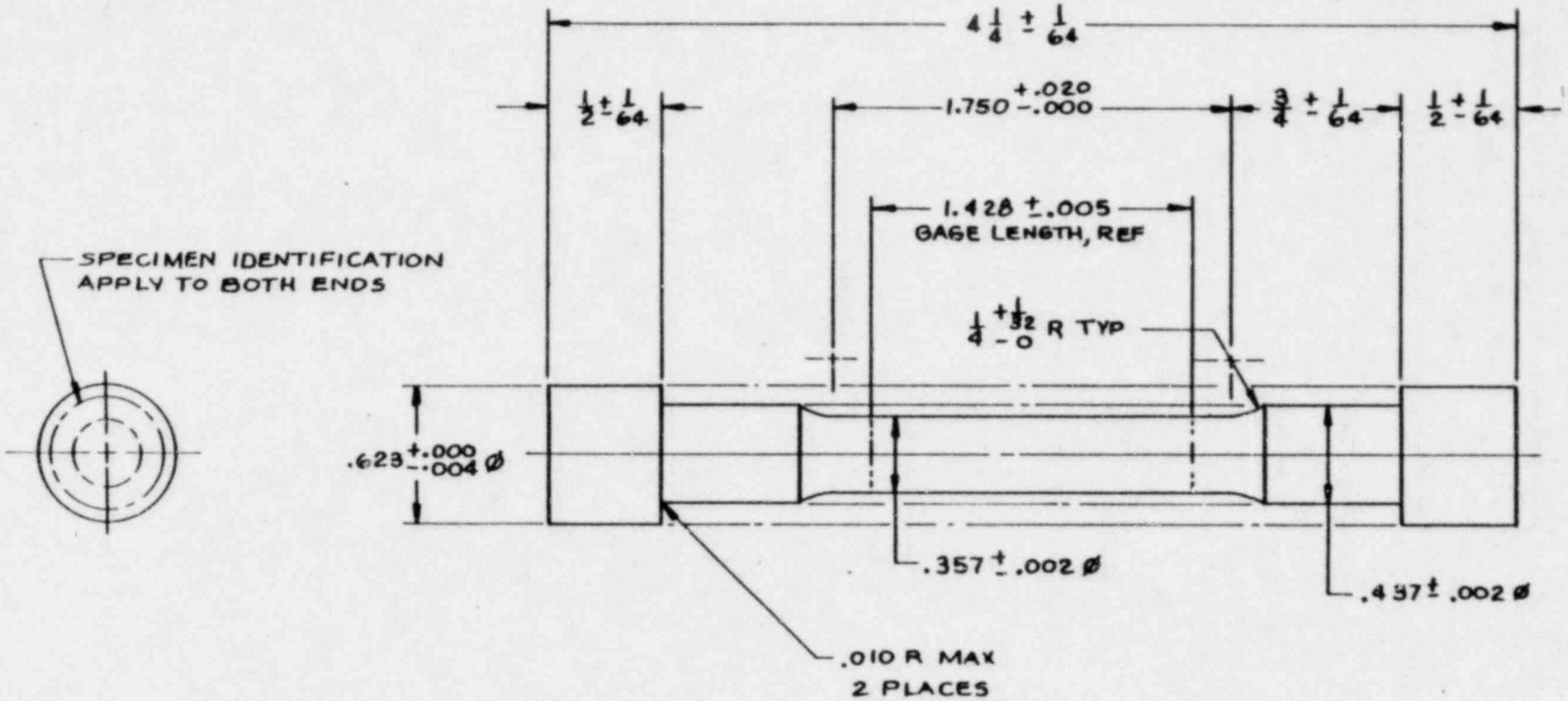


Figure C-2. Miniature Size Tensile Specimens — Used on Type B Capsules

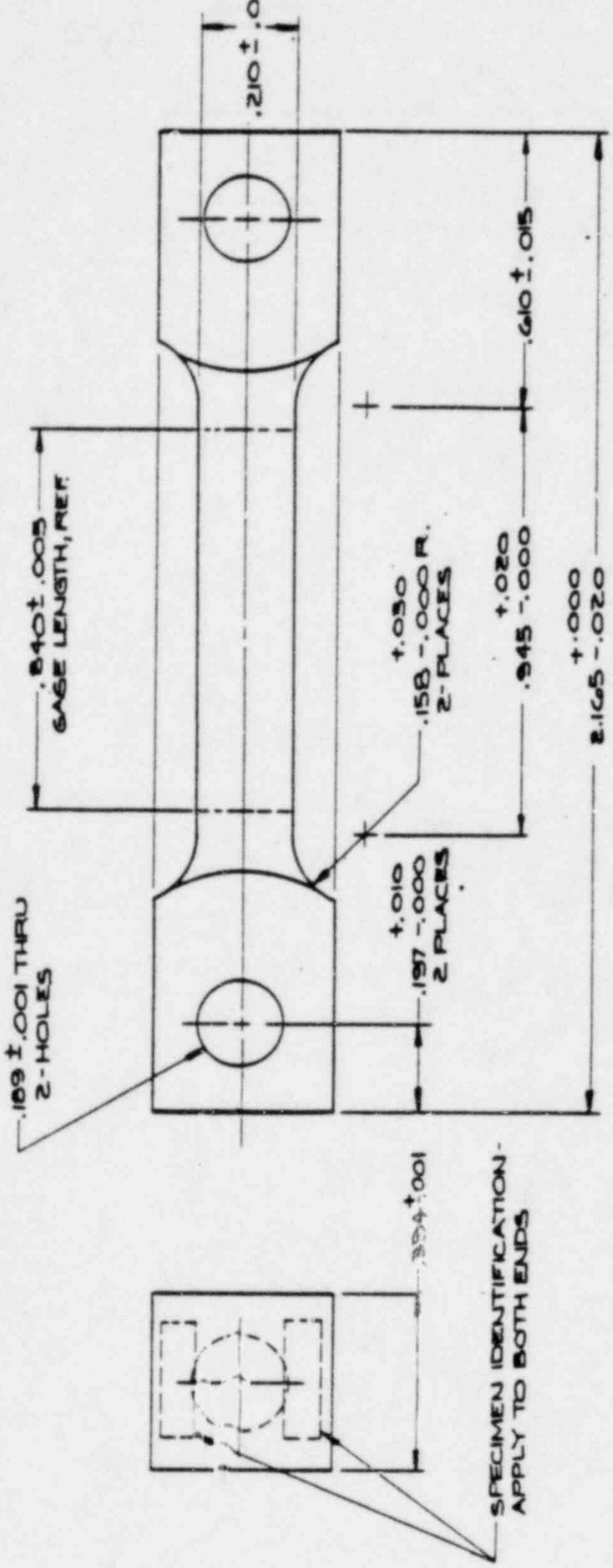


Figure C-3. Charpy V-Notch Specimen

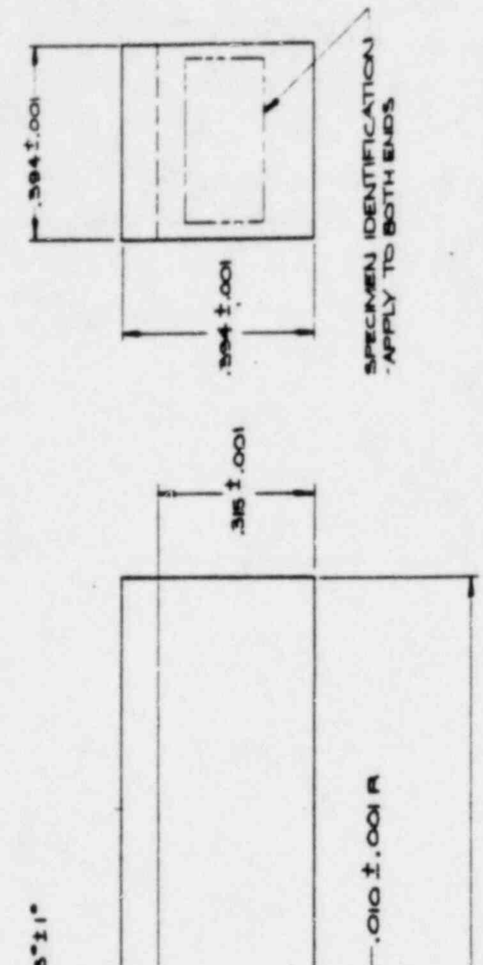


Figure C-4. Rectangular Compact Fracture Specimen — Standard Proportions and Modification for Measurement of Displacement at Load Line

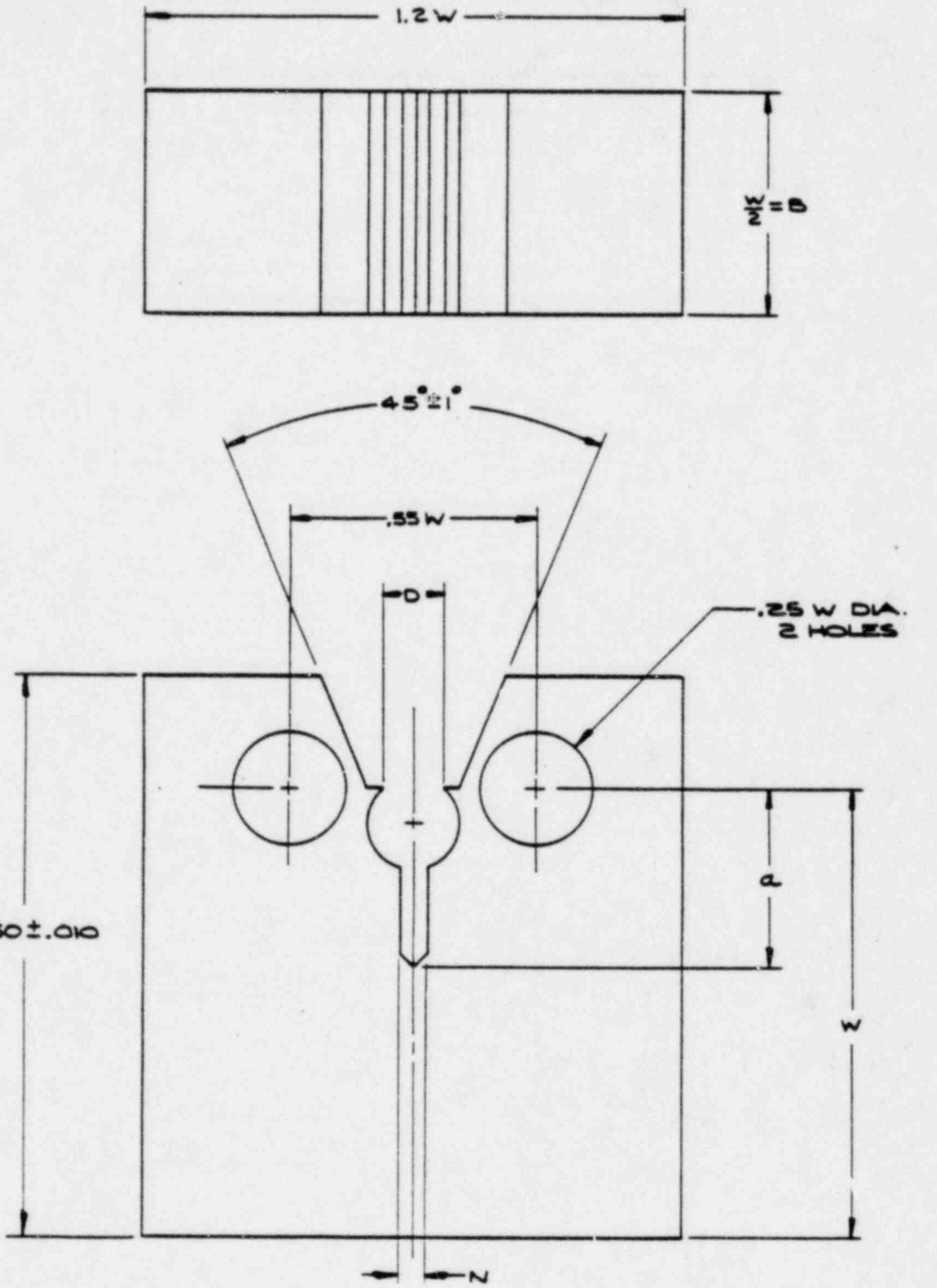


Figure C-5. Round Compact Fracture Specimen -- Dimensions and Modification for Measurement of Displacement at Load Line

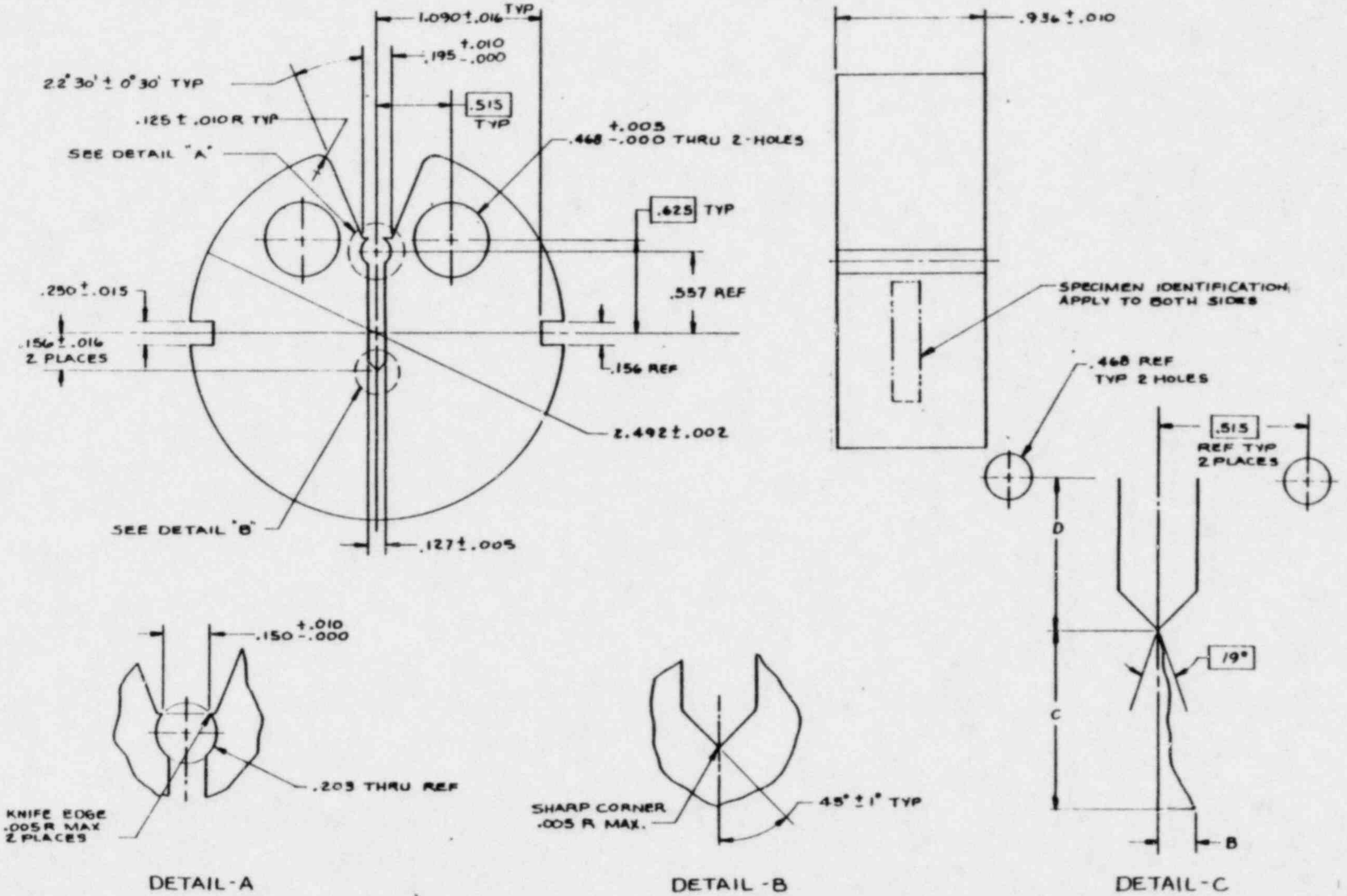
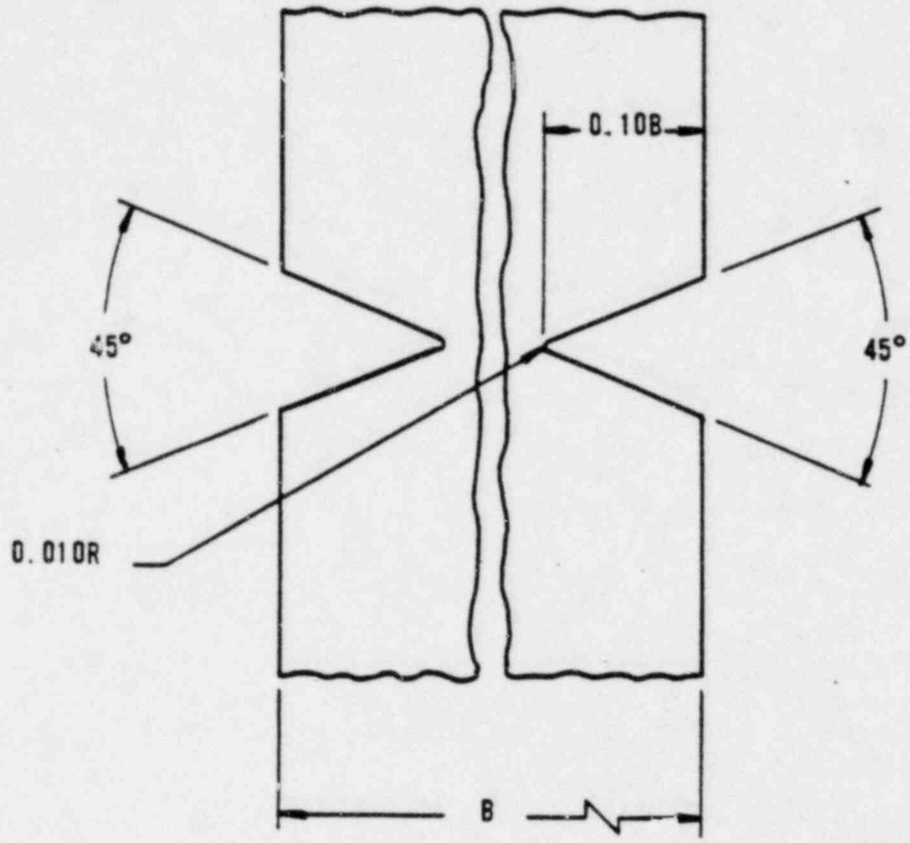


Figure C-6. Geometry of Side Grooves for 0.936 TRCT



APPENDIX D
Program and Capsule Type Designations

Table D-1. RVSP Capsule Types

Material description	No. of specimens Tensile	Charpy
Capsule Type I		
Weld metal	4	8
HAZ, heat A, longitudinal	0	8
Baseline material plate		
Heat A, longitudinal	4	8
transverse	0	4
Correlation material	0	8
Total per capsule	8	36
Capsule Type II		
HAZ, heat B, longitudinal	4	10
Baseline material plate		
Heat B, longitudinal	4	10
transverse	0	8
Correlation material	0	8
Total per capsule	8	36
Capsule Type III		
Weld metal	2	12
Weld-HAZ		
Heat A, transverse	0	12
Heat B, transverse	0	6
Base metal forgings		
Heat A, transverse	2	12
Heat B, transverse	0	6
Correlation material	0	6
Total per capsule	4	54

Table D-1. (Cont'd)

Material description	Tensile	Charpy	0.5 TCT
	No. of specimens		
Capsule Type IV			
Weld metal	2	12	8
Weld-HAZ, heat A, transverse	0	12	0
Base metal, heat A, transverse	2	12	0
Total per capsule	4	36	8
Capsule Type V			
Weld metal	2	12	
HAZ, heat A, longitudinal	0	12	
Baseline material			
Heat A, longitudinal	0	9	
transverse	2	12	
Heat B, transverse	0	9	
Total per capsule	4	54	
Capsule Type VI			
Weld metal, longitudinal	2	12	
Weld-HAZ			
Heat A, longitudinal	0	12	
Heat B, longitudinal	0	6	
Baseline material			
Heat A, longitudinal	0	0	
transverse	2	12	
Heat B, longitudinal	0	0	
transverse	0	6	
Correlation material	0	6	
Total per capsule	4	54	

Table D-2. Materials and Specimens in Surveillance Capsules of Oconee Unit 1

Material description	No. of specimens Tensile	No. of specimens Charpy
Capsules OCI-A, -C, -E		
Weld metal, WF 112	4	8
HAZ		
Heat C3265-1, longitudinal	0	8
Baseline material plate		
Heat C3265-1, longitudinal	4	8
transverse	0	4
Correlation, HSST plate 02	0	8
Total per capsule	8	36
Capsules OCI-B, -D, -F		
HAZ		
Heat C2800-2, longitudinal	4	10
Baseline material plate		
Heat C2800-2, 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 2

Material description	No. of specimens Tensile	No. of specimens Charpy
Capsules OCII-A, -C, -E		
Weld metal, WF 209	4	8
HAZ		
Heat AAW163, longitudinal	0	8
Baseline material plate		
Heat AAW163, longitudinal	4	8
transverse	0	4
Correlation, HSST plate	0	8
Total per capsule	8	36
Capsules OCII-B, -D, -F		
HAZ Heat AWG164, longitudinal	4	10
Baseline material plate		
Heat AWG164, longitudinal	4	10
transverse	0	8
Correlation, HSST plate 02	0	8
Total per capsule	8	36

Table D-4. Materials and Specimens in Surveillance Capsules of Oconee Unit 3

Material description	No. of specimens	
	Tensile	Charpy
Capsules OCIII-A, -C, -E		
Weld metal, WF 209	2	12
HAZ		
Heat A ANK191, longitudinal	0	12
Baseline material		
Heat A ANK191, longitudinal	0	9
transverse	2	12
Heat B AWG192, transverse	0	9
Total per capsule	4	54
Capsules OCIII-B, -D, -F		
Weld metal WF 209		
Longitudinal	2	12
Weld - HAZ		
Heat A ANK191, longitudinal	0	12
Heat B AWG192, longitudinal	0	6
Baseline material		
Heat A ANK191, longitudinal	0	0
transverse	2	12
Heat B AWG192, longitudinal	0	0
transverse	0	6
Correlation HSST plate 02	0	6
Total per capsule	4	54

Table D-5. Materials and Specimens in Surveillance Capsules of Three Mile Island Unit 1

Material description	No. of specimens Tensile	Charpy
Capsules TMI-1A, C, E		
Weld metal, WF 25	4	8
HAZ		
Heat C-2789-2, longitudinal	0	8
Baseline material, plate		
Heat C-2789-2, longitudinal	4	8
transverse	0	4
Correlation, HSST plate 02	0	8
Total per capsule	8	36
Capsules TMI-1B, D, F		
HAZ		
Heat C-3307-1, longitudinal	4	10
Baseline material, plate		
Heat C-3307-1, longitudinal	4	10
transverse	0	8
Correlation, HSST plate 02	0	8
Total per capsule	8	36

Table D-6. Materials and Specimens in Surveillance Capsules of Three Mile Island Unit 2

Material description	No. of specimens Tensile	Charpy	0.5 TCT
Capsules TMI-2A, C, E			
Weld metal, WF 182-1	2	12	
HAZ			
Heat C-1946-2, transverse	0	12	
Heat C-1937-2, transverse	0	6	
Base metal forging			
Heat C-1946-2, transverse	2	12	
Heat C-1937-2, transverse	0	6	
Correlation, HSST plate 02	0	6	
Total per capsule	4	54	
Capsules TMI-2B, D, F			
Weld metal, WF 182-1	2	12	8
HAZ			
Heat C-1946-2, transverse	0	12	0
Base metal forging			
Heat C-1946-2, transverse	2	12	0
Total per capsule	4	36	8

Table D-7. Materials and Specimens in Surveillance Capsules of Crystal River 3

Material description	Tensile	Charpy	0.5 TCT
	No. of specimens		
Capsules CR3-A, -C, -E			
Weld metal, WF 209	2	12	
Weld-HAZ			
Heat C4344-1, transverse	0	12	
Heat C4344-2, transverse	0	6	
Base metal forgings			
Heat C4344-1, transverse	2	12	
Heat C4344-2, transverse	0	6	
Correlation material	0	6	
Total per capsule	4	54	
Capsules CR3-B, -D, -F			
Weld metal, WF 209	2	12	8
Weld-HAZ			
Heat C4344-1, transverse	0	12	0
Base metal			
Heat C4344-1, transverse	2	12	0
Total per capsule	4	36	8

Table D-8. Materials and Specimens in Surveillance Capsules of Arkansas Nuclear One, Unit 1

Material description	No. of specimens: Tensile	No. of specimens: Charpy
Capsules ANI-A, -C, -E		
Weld metal, WF 193	4	8
HAZ		
Heat C5114-1, longitudinal	0	8
Baseline material plate		
Heat C5114-1, longitudinal	4	8
transverse	0	4
Correlation, HSST plate 02	0	8
Total per capsule	8	36
Capsules ANI-B, -D, -F		
HAZ		
Heat C5114-2, longitudinal	4	10
Baseline material plate		
Heat C5114-2, longitudinal	4	10
transverse	0	8
Correlation, HSST plate 02	0	8
Total per capsule	8	36

Table D-9. Materials and Specimens in Surveillance Capsules of Rancho Seco Unit 1

Material description	No. of specimens: Tensile	Charpy	0.5 TCT
Capsules RSI-A, -C, -E			
Weld metal, WF 193	2	12	
Weld-HAZ			
Heat C5062-1, transverse	0	12	
Heat C5070-1, transverse	0	6	
Base metal plate			
Heat C5062-1, transverse	2	12	
Heat C5070-1, transverse	0	6	
Correlation, HSST plate 02	0	6	
Total per capsule	4	54	
Capsules RSI-B, -D, -F			
Weld metal, WF 193	2	12	8
Weld-HAZ			
Heat C5062-1, transverse	0	12	0
Base metal plate			
Heat C5062-1, transverse	2	12	0
Total per capsule	4	36	8

Table D-10. Materials and Specimens in Surveillance Capsules of Davis Besse Unit 1

Materials description	No. of specimens Tensile	Charpy	0.5 TCT
Capsules TE1-A, -C, -E			
Weld metal, WF 182-1	2	12	
Weld-HAZ			
Heat 5P4086, transverse	0	12	
Heat 123x244, transverse	0	6	
Base metal forgings			
Heat 5P4086, transverse	2	12	
Heat 123x244, transverse	0	6	
Correlation material	0	6	
Total per capsule	4	54	
Capsules TE1-B, -D, -F			
Weld metal, WF 182-1	2	12	8
Weld-HAZ			
Heat 5P4086, transverse	0	12	0
Base metal			
Heat 5P4086, transverse	2	12	0
Total per capsule	4	36	8

APPENDIX E

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