BAW-1543A, Rev. 2

MAY 1985

INTEGRATED REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM

by

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K. E. Moore
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Prepared for

B&W Owners Group Materials Committee
Arkansas Power & Light Company
Duke Power Company
Florida Power Company
GPU Nuclear
Sacramento Municipal Utility District
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

March 13, 1985

Mr. J. H. Taylor, Manager, Licensing
Babcock & Wilcox
3315 Old Forest Road
P.O. Box 1260
Lynchburg, VA 24505-1260

Dear Mr. Taylor:

Subject: Acceptance for Referencing of Licensing Topical Report BAW-1543, Rev. 2, "Integrated Reactor Vessel Material Surveillance Program"

We have completed our review of the subject topical report submitted by Babcock & Wilcox (B&W) by letter dated March 14, 1984. We find the report to be acceptable for referencing in license applications to the extent specified and under the limitations delineated in the report and the associated NRC evaluation, which is enclosed. The evaluation defines the basis for acceptance of the report.

We do not intend to repeat our review of the matters described in the report and found acceptable when the report appears as a reference in license applications, except to assure that the material presented is applicable to the specific plant involved. Our acceptance applies only to the matters described in the report.

In accordance with procedures established in NUREG-0390, it is requested that B&W publish accepted versions of this report, proprietary and non-proprietary, within three months of receipt of this letter. The accepted versions shall incorporate this letter and the enclosed evaluation between the title page and the abstract. The accepted versions shall include an -A (designating accepted) following the report identification symbol.

Should our criteria or regulations change such that our conclusions as to the acceptability of the report are invalidated, B&W and/or the applicants referencing the topical report will be expected to revise and resubmit their respective documentation, or submit justification for the continued effective applicability of the topical report without revision of their respective documentation.

Sincerely,

Cecil O. Thomas

Cecil O. Thomas, Chief
Standardization and Special
Projects Branch
Division of Licensing

Enclosure:
As stated

SAFETY EVALUATION
BABCOCK & WILCOX REPORT BAW-1543, REV. 2, DATED FEBRUARY 1984
"INTEGRATED REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM"
(TAC NO. 53244)

MATERIALS APPLICATION SECTION
MATERIALS ENGINEERING BRANCH

In 1976 several utilities with reactor vessels designed by Babcock & Wilcox requested exemptions from the Appendix H requirement for an in-vessel material surveillance program. The staff reviewed and evaluated each licensee's request and proposal for an integrated surveillance program; and granted the requested exemption for a period of five years.

A revised Appendix H, 10 CFR 50 was published in the Federal Register on May 27, 1983 and became effective on July 26, 1983. Section II.C of the revised Appendix H provides an integrated surveillance program provided it is approved by the Director, Office of Nuclear Reactor Regulation. This section of Appendix H provides the criteria to be used in evaluating the integrated surveillance program. The criteria are:

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

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

3. There must be an adequate dosimetry program for each reactor.

4. 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.

5. No reduction in the requirements for number of materials to be irradiated, specimen type, or number of specimens per reactor is permitted.

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

In a letter from J. H. Taylor to D. Moran, dated March 14, 1984, the B&W Owners Group submitted for staff review and approval a revised integrated surveillance program. The program was documented in Babcock & Wilcox Report BAW-1543, Rev. 2, February 1984, "Integrated Reactor Vessel Material Surveillance Program."

DISCUSSION

The B&W Owners Group initiated the B&W 177 FA integrated program as a result of flow induced vibration that caused cracking of capsule holder tubes in reactor vessels at Oconee, Units 1, 2, and 3; Arkansas Nuclear One, Unit 1; Rancho Seco; and Three Mile Island, Unit 1. The capsules were

removed from these vessels, redesigned, and installed in Crystal River, Unit 3, Davis-Besse, Unit 1, and Three Mile Island, Unit 2 reactors, which had, as yet, not achieved criticality. Installation of new holder tubes in operating reactors would have subjected plant personnel to significant radiation exposure -- up to about 100 man-rem per reactor. Thus, transferring capsules to unirradiated reactor vessels was an advantage for the utilities with cracked capsule holder tubes because it reduced personnel exposure.

The reactor vessels with capsules and the reactor vessels without capsules are Babcock & Wilcox designed vessels, which contain 177 Mark B (15 x 15) fuel assemblies, have core barrels and thermal shields of the same design and are constructed of similar materials. Since the neutron energy spectrum is a function of geometry, materials, and core loading, the relative neutron spectrum for all reactors participating in the program will be equivalent for equivalent core loadings. Hence, it is expected that differences in neutron irradiation dose rate between reactors would be only due to variations in power levels and core loadings.

The reactor vessel capsule dosimetry is used to estimate the past and future neutron fluence of the vessel and capsules. In this program the neutron fluence analysis uses a time weighted average (pin by pin) power distribution to account for variations in power levels and core loadings.

The flux calculated is compared to the flux measured from the analyses of the dosimeters to produce a normalizing factor to be used in estimating reactor vessel neutron fluence. This method of calculating neutron irradiation dose rate will account for variations in power levels and core loadings between reactors participating in the integrated surveillance program.

Uncertainties in neutron fluence estimates were discussed by the staff in its review of the B&W owners group request for exemptions to the requirements of Appendix H, 10 CFR 50. The dosimetry methodology and vessel fluence analysis have been reviewed and accepted by the staff in a memorandum dated December 5, 1984 from L. S. Rubenstein to W. V. Johnston, "Review of Response to the Request for Additional Information on Capsule RSI-B for Rancho Seco, Reported in BAW-1702."

In the staff's review of BAW-1702 it was reported that this methodology resulted in a maximum uncertainty in end-of-life vessel fluence of 34 percent. This uncertainty may be reduced for vessels not containing in-vessel dosimetry by inclusion of dosimetry devices in the reactor cavity. The B&W Owners Group has indicated that they have begun testing of these types of dosimeter devices. However, until these devices are installed, plants without dosimetry in the reactor vessel will have to rely on the methods of neutron fluence analysis documented in BAW 1702.

The method of predicting radiation damage recommended by the staff is documented in Regulatory Guide 1.99, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials." In this regulatory guide, the amount of radiation damage is estimated from the changes in fracture toughness of the material. The change in fracture toughness is dependent upon the neutron fluence and the residual elements in the reactor vessel beltline materials. The residual elements in all beltline materials, which are in Owners Group reactor vessels, have been documented by either a chemical analysis from base metal prolongations or from weld metal samples that were analyzed by the methods in Babcock & Wilcox Reports BAW-1500, "Chemistry of 177-FA B&W Owners' Group Reactor Vessel Beltline Welds" and BAW-10144, "Evaluation of Atypical Weldment." The use of these chemical analysis and the neutron fluence predicted for each reactor vessel will provide each utility with the capability of predicting the amount of neutron irradiation damage as a function of power output.

As previously discussed, all capsules would have been irradiated in the reactor vessels at Crystal River Unit 3, Davis Besse Unit 1, and Three Mile Island Unit 2. The capsules must receive sufficient neutron irradiation to meet the requirements of ASTM E 185, "Standard Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels." The capsules have been located within the reactor vessels at positions in which the lead

factor between the capsule and the one quarter thickness location is either 7 to 1 or 9.7 to 1. These lead factors will result in the capsules receiving their required neutron fluence, when the capsules are placed in the vessel locations and removed in accordance with the schedule documented in Table E-1 of BAW 1543, Rev. 2.

The integrated surveillance program contains six surveillance capsules from each reactor vessel that is participating in the integrated surveillance program. In addition, the Owners Group has provided six research capsules to the integrated surveillance program. Hence, there are 60 capsules being irradiated as part of the integrated surveillance program.

There are presently two research capsules and four surveillance capsules in the reactor vessel at Three Mile Island, Unit 2. Since this unit has been shut down since 1979, the status of these capsules are in doubt. There are 54 other capsules in the integrated surveillance program, which can be irradiated at either Davis Besse, Unit 1 or Crystal River, Unit 3. Since there are two vessels available for irradiation of the remaining 54 capsules, shuffling of capsules between vessels is possible, and the capsules have been located at positions that have high lead factors, an extended outage or operation at reduced power levels should not seriously jeopardize the integrated surveillance program.

Appendix H, 10 CFR 50 requires that the number of specimens, type of specimens and material being irradiated in surveillance capsules must be capable of monitoring the changes in fracture toughness properties of the ferritic materials in the reactor vessel beltline.

Using the method of estimating neutron irradiation damage in Regulatory Guide 1.99, the B&W Owners Group has estimated the changes in the fracture toughness (increase in reference temperature) of each ferritic material in the beltline of each reactor vessel that is participating in the integrated surveillance program. Using this method of analysis, the limiting materials in each beltline are the weld metals which were fabricated using the automatic submerged arc weld process, Linde 80 flux, and Mn-Mo-Ni filler wire. The amount of residual elements in each beltline weld of each vessel has been reported, as previously discussed, by the B&W Owners Group. The base metal in the beltlines are either SA 533 Gr.B, Cl.1 plate, SA 302 Gr.B plate or SA 508 Cl.2 forgings.

The type of specimens, number of specimens, materials and amounts of residual elements in each material that are in each capsule are identified in BAW 1543. The base metal samples in the surveillance capsules were prepared from either SA 533 Gr.B, Cl.1 plate, SA 302 Gr.B plate and SA 508 Cl.2 forgings. The weld metal samples in the surveillance capsules were fabricated using the automatic submerged arc weld process, Linde 80 flux and Mn-Mo-Ni filler wire. The amounts of residual elements in the weld metal samples in the surveillance capsules and the research capsules are representative of the amounts of residual elements in the beltline welds in the reactor vessels.

The number and types of specimens should be adequate for determining the effect of neutron irradiation on the fracture toughness of each material in each capsule. Hence, the specimens and materials placed in the capsules should be adequate for monitoring the effect that neutron irradiation has on the fracture toughness of the beltline materials in the B&W Owners Group reactor vessels.

To assure that data from the integrated surveillance program is shared between plants, all reactor vessel surveillance program test reports and relevant information is distributed to all plant owners through the B&W Owners Group Materials Committee. Each utility participating in the integrated surveillance program has a representative on the Materials Committee. The utility representatives meet regularly to discuss this program, monitor test results, and benchmark efforts.

CONCLUSIONS

1. The integrated surveillance program documented in Babcock & Wilcox Report BAW-1543, Rev. 2, February 1984 meets the evaluation criteria of Section II.C of Appendix H, 10 CFR 50.

2. Based on conclusion 1, we recommend that the Director, Office of Nuclear Reactor Regulation, approve the integrated surveillance program documented in B&W Report [illegible]3, Rev. 2 for Oconee, Units 1, 2 and 3; Arkansas Nuclear One, Unit 1; Rancho Seco; Three Mile Island, Unit 1; Crystal River, Unit 3;

and Davis Besse, Unit 1. Three Mile Island, Unit 2 and Midland, Units 1 and 2 are participating in the integrated surveillance program. Since the status of the surveillance capsules in these plants is in doubt, approval of the integrated surveillance program for these plants is not justified at this time.

3. After approval of this integrated surveillance program by the Director, Office of Nuclear Reactor Regulation, exemptions to Appendix H, 10 CFR 50 will no longer be required, and licensees may utilize BAW-1543, Rev. 2 and the resulting test data to support their licensing actions.

4. In-cavity dosimetry testing should continue in order to reduce uncertainties in neutron fluence for vessels that do not contain in-vessel dosimetry. If these test results provide an effective method of monitoring vessel neutron fluence, the in-cavity dosimetry should be incorporated in plants.

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

May 3, 1985

Mr. J. H. Taylor, Manager, Licensing
Babcock & Wilcox Company
3315 Old Forest Road
P. O. Box 1260
Lynchburg, Virginia 24505-1260

1543

Dear Mr. Taylor:

SUBJECT: ACCEPTANCE FOR REFERENCING OF REVISIONS TO TABLES 3-11 AND E-1 OF LICENSING TOPICAL REPORT BAW-1534, REV. 2 "INTEGRATED REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM"

Revision 2 was accepted by our letter dated March 13, 1985. On March 18, 1985, the Owners Group informed NRR that errors had been found and provided revisions to two tables correcting the error.

We have completed our review of the subject revisions to Tables 3-11 and E-1 submitted by Babcock & Wilcox Company (B&W) by letter dated March 18, 1985. We find the revised report to be acceptable for referencing in license applications to the extent specified and under the limitations delineated in the report and the associated NRC evaluation, which is enclosed. The evaluation defines the basis for acceptance of the report.

We do not intend to repeat our review of the matters described in the report and found acceptable when the report appears as a reference in license applications, except to assure that the material presented is applicable to the specific plant involved. Our acceptance applies only to the matters described in the report.

In accordance with procedures established in NUREG-0390, it is requested that B&W incorporate the revised Tables 3-11 and E-1 and publish accepted versions of this report, proprietary and non-proprietary, within three months of receipt of this letter. The accepted versions shall incorporate this letter and the enclosed evaluation between the title page and the abstract. The accepted versions shall include an -A (designating accepted) following the report identification symbol.

Should our criteria or regulations change such that our conclusions as to the acceptability of the report are invalidated, B&W and/or the applicants referencing the topical report will be expected to revise and resubmit their respective documentation, or submit justification for the continued effective applicability of the topical report without revision of their respective documentation.

Sincerely,

Herbert N. Berkow for

Cecil O. Thomas, Chief
Standardization and Special
Projects Branch
Division of Licensing

Enclosure:
As stated

SAFETY EVALUATION

B&W REPORT BAW 1543, REV. 2
(TAC NO. 55683)

MATERIALS APPLICATION SECTION
MATERIALS ENGINEERING BRANCH

In a letter from J.H. Taylor to J.F. Stolz dated March 18, 1985, the B&W Owners Group reported that the surveillance withdrawal schedule in Babcock & Wilcox Report BAW-1543, Rev. 2 "Integrated Reactor Vessel Material Surveillance Program," was incorrect because the Oconee capsule OCI-B had been inserted into the Crystal River 3 reactor vessel in lieu of Oconee capsule OC II-B. As a result of this error, the B&W Owners Group requested staff approval of a proposed withdrawal schedule, which was documented in revised BAW 1543 Tables 3-11 and E-1.

BAW 1543, Rev. 2 indicates that the desired neutron fluences for capsules OCI-B and OCII-B are 4.4×10^{18} n/cm^2 and 6.7×10^{18} n/cm^2, respectively. Revision 2 to BAW 1543 was approved by the staff in a letter from C.O. Thomas to J.H. Taylor dated March 13, 1985. The Owners Group indicates that the proposed capsule withdrawal schedule will result in capsules OCI-B and OCII-B receiving neutron fluences of 6.5×10^{18} n/cm^2 and 7.9×10^{18} n/cm^2, respectively. The expected neutron fluence to be received by each capsule is sufficiently close to the desired amount to ensure that the surveillance program provides meaningful reactor vessel materials surveillance data. Hence, the proposed withdrawal schedule should be incorporated into the next revision of BAW-1543 and plants technical specifications.

The Owners Group indicates that this situation arose because of difficulty in reading the serial number etched on the activated capsules. In order to minimize the possibility of this situation arising in the future, B&W proposes to take the following actions:

1. Request utility and field personnel involved with Integrated Reactor Vessel Surveillance Program (IRVSP) capsule handling to obtain independent verification of serial numbers and report the actual shuffle performed in writing.

2. Maintain IRVSP capsule inventory control at the Lynchburg Research Center's (LRC) storage pool with updates performed following each movement.

In addition, to ensure that there have been no other errors of this type, the serial numbers on each IRVSP capsule in the LRC storage pool should be verified within the next six months and the results reported to the staff.

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 the 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 eleven 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 three classes -- those from which specimens come (guests), those in which the specimen irradiations are performed (hosts), and plant-specific programs. The eleven power reactors involved (six guests, three hosts, two plant-specific) are similar in both design and operating conditions.. Specimens are enclosed in two different

types of specially designed cylindrical capsules -- the early design surveillance capsules and the larger plant-specific surveillance and research capsules.

A brief description of the federal guidelines and legislation is presented, 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; detailed descriptions of the types and properties of materials being investigated, the types of capsules used to contain the surveillance specimens, and, in the appendixes, the program and capsule designations are given.

The report has been revised to include the latest information on the surveillance capsule withdrawal schedules, to add new information on regulatory requirements, reactor vessel end-of-life reference temperatures, and neutron dosimetry, and to incorporate the Nuclear Regulatory Commission's letters of acceptance.

Babcock & Wilcox
Nuclear Power Division
Lynchburg, Virginia

Report BAW-1543A, Rev. 2
May 1985

Integrated Reactor Vessel Material Surveillance Program

A. L. Lowe, Jr., K. E. Moore, J. D. Aadland

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 that included 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, and two plant-specific programs from reactors soon to come on-line; 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. This revision incorporates revised surveillance capsule withdrawal schedules, new information on regulatory requirements, reactor vessel end-of-life reference temperatures, and neutron dosimetry, and the Nuclear Regulatory Commission's letters of acceptance.

RECORD OF REVISIONS

Date	Revision Number	Description
November 1981	0	Original Issue
October 1983	1	Summary - revision paragraph added
		Abstract - revision sentence added
		Section 1.1 - Midland 1 and 2 information added
		Tables 1-1, 1-2, 1-3 - Information updated, typographical errors corrected
		Table 1-4 added
		Section 2 - rewritten and updated
		Table 2-1 updated
		Section 3. - updated
		Section 3.1 - Midland 1 and 2 information added
		Section 3.1.4 - rewritten and expanded; Midland 1 and 2 information added.
		Section 3.2.3 - was Section 3.2.4
		Section 3.2.4 - was Section 3.2.5
		Section 3.2.5 - was Section 3.2.6
		Section 3.2.6 - was Section 3.2.7
		Section 3.3 - expanded and updated for TMI-2 complication and Midland 1 and 2
		Section 3.4 - was Section 3.2.3
		Table 3-1 - updated to ASTM E 185-79
		Table 3-2 - expanded
		Table 3-4 - updated; Midland 1 and 2 information added
		Table 3-5 - corrected
		Table 3-6 - corrected
		Table 3-8 - expanded
		Table 3-9 - corrected
		Table 3-11 - updated
		Table 3-12 - updated
		Table 3-13 - updated
		Table 3-14 - new table

RECORD OF REVISIONS (Cont'd)

Date	Revision Number	Description
		Table 3-15 - new table
		Figure 3-3 - new figure
		Figure 3-4 - new figure
		Figure 3-5 - was Figure 3-3
		Figure 3-6 - was Figure 3-4
		Figure 3-7 - was Figure 3-8
		Figure 3-8 - new Figure
		Figure 3-9 - was Figure 3-5
		Figure 3-10 - was Figure 3-6
		Figure 3-11 - was Figure 3-7
		Figure 3-12 - new figure
		Figure 3-13 - new figure
		Section 4 - new signature page
		Section 5 - expanded; was Appendix E
		Tables A-1 through A-9 - updated and corrected
		Table A-10 - new table
		Table A-11 - new table
		Table B-1 - updated
		Table B-2 - updated
		Table C-1 - corrected
		Table D-1 - expanded
		Table D-11 - new table
		Table D-12 - new table
		Appendix E - new Appendix
		Minor rewrites and corrections was made throughout Revision 1.
February 1984	2	Summary - last paragraph revised
		Abstract - last paragraph revised
		Table 3-1 - Title changed to reflect ASTM E 185-82
		Table 3-11 - updated
		Table 3-12 - updated
		Section 4 - new signature page

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Tables (Cont'd)

Tables (Cont'd)

List of Figures

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 Babcock & Wilcox (B&W) 177-fuel assembly (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.[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 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 damages and to eliminate the possibility of overall system damage if parts of the holder tube were dislodged, all of 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 (ANO-1); Rancho Seco; and Three Mile Island Unit 1 (TMI-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 RVSPs 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 eventually not meet current design requirements because of initial properties and

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*ASTM standards and B&W reports are not referenced specifically because of their frequent occurrence, but are included in the list of references under separate headings in numerical sequence.

unusual sensitivity to neutron embrittlement. This problem was further complicated by the fact that the capsules in the affected plants did not contain the 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 were 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 reactor owners of concern.

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, Nuclear Regulatory Commission-approved design were installed in Davis-Besse 1, Crystal River 3, and Three Mile Island 2 (TMI-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 would need to be performed remotely and underwater. 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.

Since the implementation of this program, the three host reactors have come on-line. Two additional 177-FA plants, Midland Units 1 and 2, will come on-line in the near future. While it is currently planned to irradiate the Midlands as plant-specific programs, i.e., each plant will irradiate its own surveillance capsules, the data generated will be available to the overall program, and so in this sense the two plants are part of the program. Additionally, the TMI-2 incident leaves some unanswered questions for the program. Only one TMI-1 surveillance capsule was in TMI-2 at the time of the accident, so TMI-1 still has five available capsules, the number required by ASTM specification E185-82. The capsules of TMI-1 will be irradiated in Crystal River 3. The status of the capsules in TMI-2 is undetermined, and will remain so until they are removed and evaluated following the TMI-2 head removal.

The current status of the integrated program is as follows:

Guest reactors	Owners	Host reactors
Oconee 1, 2, 3 Three Mile Island 1 (partial)	Duke, GPU(N)/Florida Power	Crystal River 3
Arkansas Nuclear One 1, Rancho Seco 1	AP&L, SMUD/Toledo Edison	Davis-Besse Unit 1
Three Mile Island 1 (partial)	GPU(N)/GPU(N)	Three Mile Island 2
Plant specific	Consumers Power	Midland 1 Midland 2

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 vessels and internals 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 through 1-4.

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 Tables 1-1 through 1-4, 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 not differ significantly from plant to plant, and will have no effect on surveillance results.

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 operational variations among the plants and power level changes due to maneuvering.

The variations due to design operational differences between the host and guest reactors are insignificant. Between partial- ($\sim$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 over time among plants; this is supported by the comparability of available reactor vessel surveillance results. In any case, the variation in inlet condition temperatures is considered too low to cause significant variation in self-annealing. The inlet temperature will also vary about 40F between hot shutdown (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, some of the individual RVSPs 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 for vessels sustaining extensive radiation degradation. 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. The integrated program eliminated the need to replace the failed specimen holder tubes on operating plants and thus eliminated the potential high exposure to personnel and provides 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 parameters	Host reactor: Davis-Besse 1	Guest reactors: Rancho Seco	Guest reactors: 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/h; gpm	143.8; 387,000	143.8; 387,000	139.7; 375,000
Coolant temperatures, F			
Nominal inlet	558	558	556
Avg rise in vessel	49	49	47
Avg in vessel	582	582	579
No. of fuel assemblies	177	177	177
Type of fuel assemblies	Mark B (15x15)	Mark B (15x15)	Mark B (15x15)
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.	143.2	141.8	141.8
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 Tp B Cl.1	SA533 Tp B 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 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

(a)Over cladding and instrumentation nozzles.

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

Plant parameters	Host reactor: Crystal River 3	Guest reactors: Oconee 1	Oconee 2	Oconee 3	TMI-1
Design heat output (core), MWt	2544	2568	2568	2568	2568
Design overpower, % des power	112	112	112	112	112
System pressure (nominal), psia	2200	2200	2200	2200	2200
Coolant flow rate, 10^6 lb/h;gpm	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
Avg rise in vessel	46	47	47	47	47
Avg in vessel	579	579	579	579	579
No. of fuel assemblies	177	177	177	177	177
Type of fuel assemblies	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
Thermal shield ID/OD, in.	147/151	147/151	147/151	147/151	147/151
Core structural characteristics					
Core equiv diameter, in.	128.9	128.9	128.9	128.9	128.9
Core active fuel height, in.	141.8	141.8	141.8	141.8	142.3
Reflector thickness, compos'n					
Top (water + steel), in.	12	12	12	12	12
Bottom (water + steel), in.	12	12	12	12	12
Side (water + steel), in.	18	18	18	18	18
Reactor vessel design parameters					
Principal material	SA533 Tp B C.1	SA302 GrB Cl.1[a]	SA508 Cl.2	SA508 Cl.2	SA302 GrB[a]
Design pressure, psig	2500	2500	2500	2500	2500
Design temperature, F	650	650	650	650	650
Shell ID, in.	171.375	171	171	171	171
OD across nozzles, in.	249	249	249	249	249
Overall vessel-closure head height[b]	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

[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/h; gpm	137.8;369,000	139.7;375,000
Coolant temperatures, F		
Nominal inlet	556	556
Avg rise in vessel	52	47
Avg in vessel	582	579 No. of fuel
assemblies	177	177 Type of fuel
assemblies	Mark B (15x15)	Mark B (15x15)
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	142.3 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, Tp 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 7/8"	40' 8 7/8"
Core barrel-thermal shield principal material	Type 304 SS	Type 304 SS

[a]Over cladding and instrumentation nozzles.

Table 1-4. Compilation of Plant Parameters for Midland Unit 1 and Midland Unit 2

Plant parameters	Midland 1	Midland 2
Design heat output (core), MWt	2452	2452
Design overpower, % design power	112	112
System pressure (nominal), psia	2200	2200
Coolant flow rate, 10^6 lbm/h; gpm	131.3;352,000	131.3;352,000
Coolant temperatures, F		
Nominal inlet	555	555
Avg rise in vessel	48	48
Avg in vessel	579	579
No. of fuel assemblies	177	177
Type of fuel assemblies	Mark B (15x15)	Mark B (15x15)
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	141.8	141.8
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	SA508 Cl. 2	SA508 Cl. 2
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)]	40' 8 7/8"	40' 8 7/8"
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

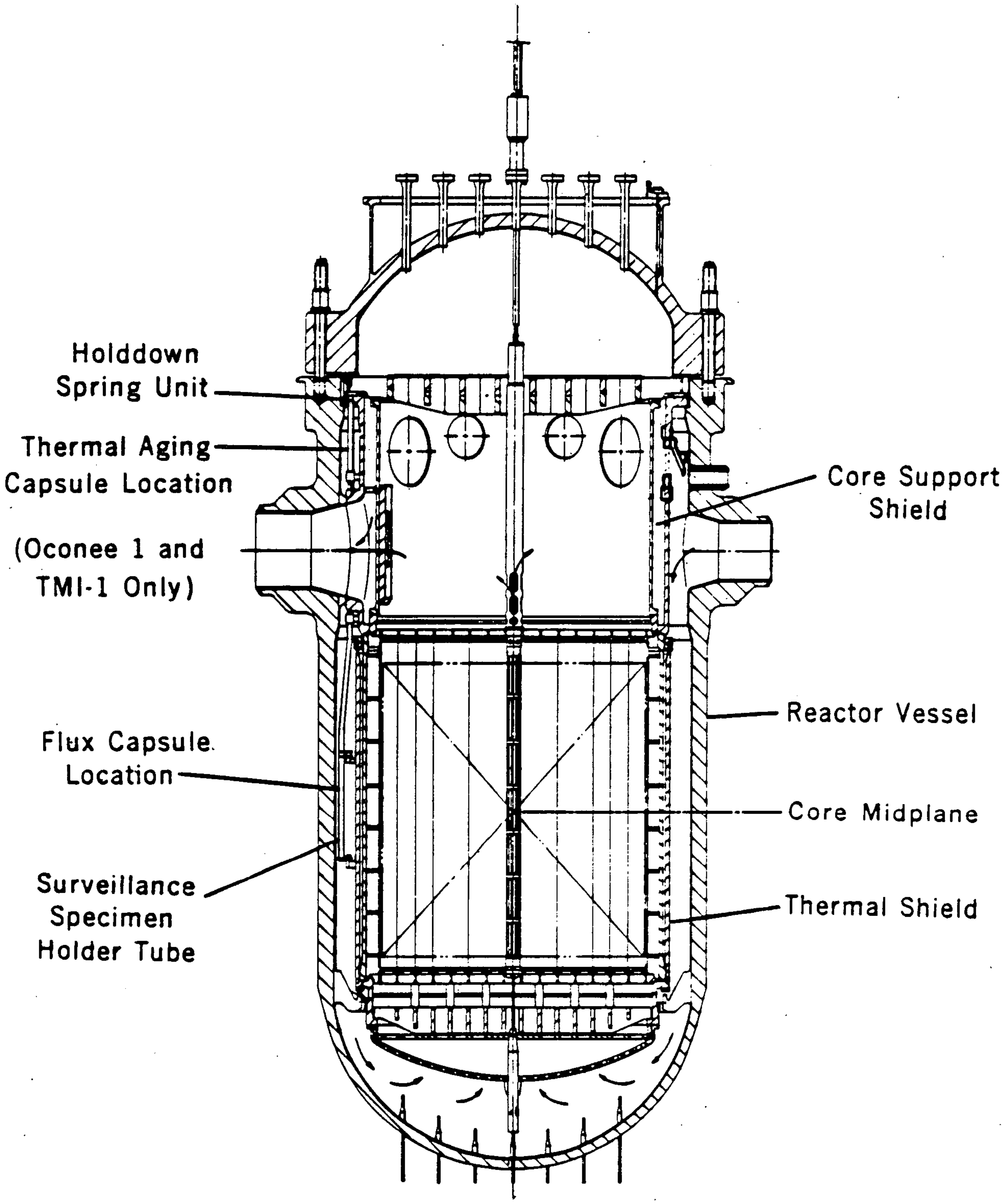
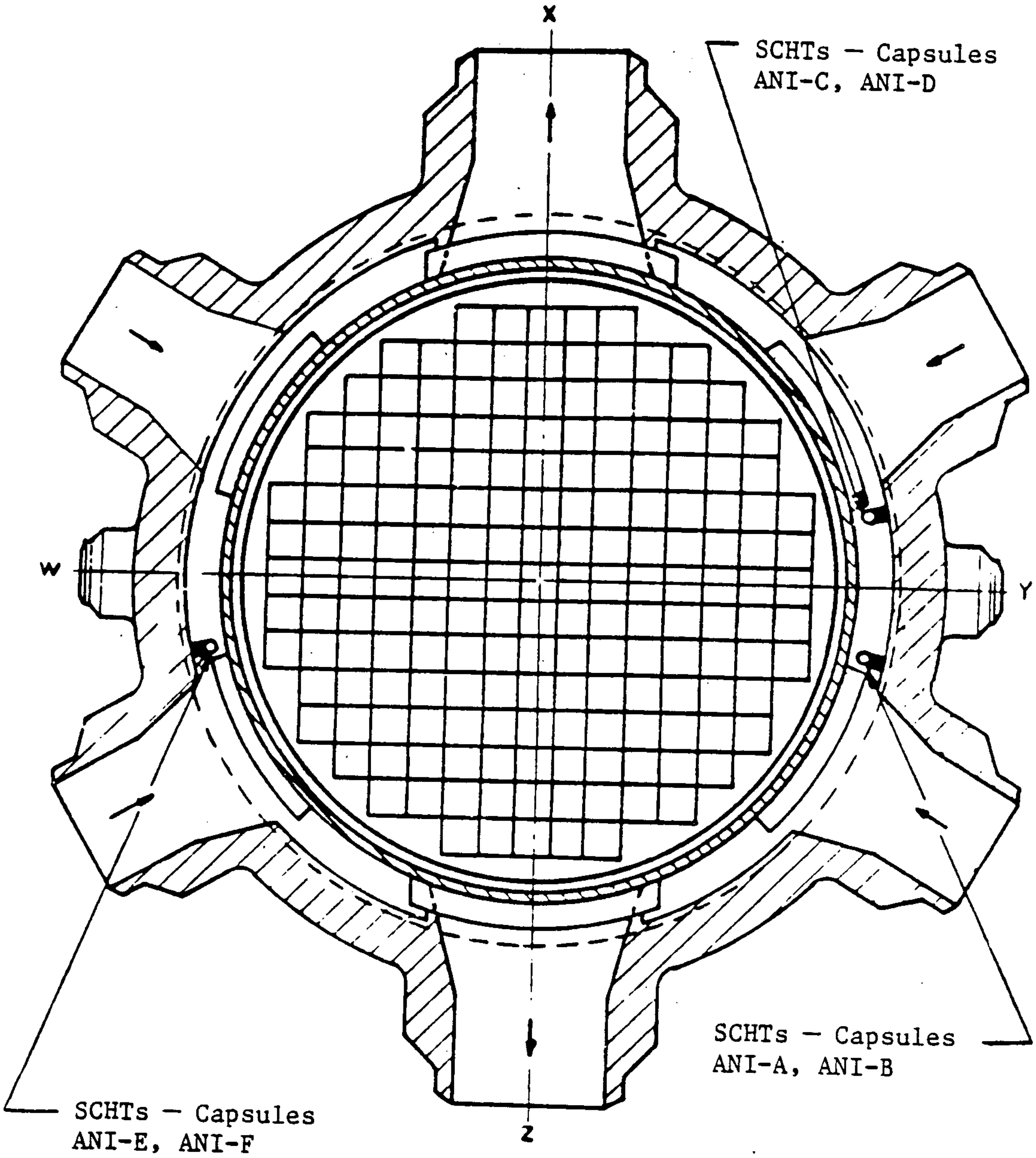


Figure 1-2. Reactor Vessel Cross Section Showing Original Surveillance Capsule Locations (Typical Plant)



2. BACKGROUND

It became apparent in the late 1950's that neutron embrittlement could seriously degrade the mechanical properties of steels used in reactor vessels. This was a phenomenon that varied significantly from one type of steel to another, from one heat to another, and from one weld to another. Accordingly, a number of research programs were conducted to evaluate the phenomenon. By the time the first commercial nuclear plants were designed, enough data were accumulated to confirm that the neutron irradiation damage to the reactor vessel materials could significantly degrade the properties. Not all of the first generation reactor vessels were equipped with surveillance capsules to monitor irradiation damage; however, out of this initial period of nuclear development the guidelines for establishing a RVSP were adopted. The ASTM standard E185-61, "Standard Practice for Conducting Surveillance Tests for Light Water-Cooled Nuclear Power Reactor Vessels"* was issued and conveyed the current state-of-the-art technology for designing such a program.

During the 1960's, the research on neutron irradiation damage to reactor vessel steels continued. Specific research during this time studied the effects of the principal parameters influencing neutron embrittlement sensitivity. The effects of differing neutron spectra, neutron flux rates, irradiation temperature, and chemical composition were studied as parts of different programs. These studies were conducted in both commercial power plants and test reactors with the primary objective of determining the sensitivity of commonly used reactor vessel steels to neutron irradiation.[3-6]

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*This document was revised in 1966, 1970, 1973, 1979 and 1982 to reflect knowledge gained. These revisions are compared in Table 2-1.

In the late 1960's, a significant discovery was made when the copper and phosphorus contents in pressure vessel materials were identified by the Naval Research Laboratory as principal parameters contributing to mechanical property degradation.[7-10] Further work in cooperation with The Babcock & Wilcox Company confirmed the role of these two elements and led to their restriction in the manufacture of steels and weld metal to be exposed to neutron fluences.

In 1973, in a concerted effort to improve the quality of reactor pressure vessel integrity and to base the assessment of vessel integrity on a theoretical rather than an empirical basis, the concept of fracture mechanics techniques was implemented through Nuclear Regulatory Commission (NRC) regulation. These requirements are included in 10 CFR 50, Appendix G, "Fracture Toughness Requirements."[11] Also included was a requirement for monitoring the neutron embrittlement of the reactor vessel beltline region, which is described in Appendix H of 10 CFR 50 and called, "Reactor Vessel Material Surveillance Program Requirements."[12] With the issue of Appendix H, a justification was established for a concerted effort to acquire the necessary information from surveillance capsules and to standardize the existing surveillance program to the extent possible. Also, these appendixes made the RVSP mandatory. Up to this point, the data gathered from the RVSP had received low priority and were diversified because the requirements were broadly defined and gave considerable latitude to the RVSP designer.

In addition, ASTM E185-73 was further revised to support the requirements of 10 CFR 50, Appendix H through a cooperative effort between the standard development committee and the regulators.

Today, the requirements of 10 CFR 50, Appendixes G and H, together with ASTM standard E185, are recognized throughout the nuclear industry as the standards and procedures for ensuring the integrity of nuclear reactor pressure vessels subjected to environmental degradation during service.

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

ASTM E185 Revision	Materials monitored by program	No. of capsules	No. of specimens/ capsule/material	No. of baseline specimens/mat's	Specimen orientation	Index temp for measuring T	Capsule withdrawal schedule	Dosimetry requirements	Temperature monitor requirements	Special requirements and recommendations
1966	1. Base metal with the highest trans temp. 2. Any weld metal 3. HAZ metal	13 or more	8 Charpy 2 tension	15 Charpy 3 tension	Parallel to major working direc'n	Charpy energy fix temp as identified by NDT unirr. drop wt tests (normally 30 ft-lb)	One at neutron fluence corresponding to EOL; others not specified	Refer to ASTM E261; selection given to designer	Low-melting-point elements or alloys may be employed	1. Desirable to include correlation monitor 2. Thermal control specimen desirable
1970	1. Base metal with the highest trans temp. 2. Representative weld metal (same wire or rod & flux as one of the high-flux region welds) 3. HAZ of base metal	3 or more	8 Charpy 2 tension	15 Charpy 3 tension	Parallel to major working direc'n	Same as above	One corresponding to 30% of design life; one to 100% life; others not specified	Determined per ASTM 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 E261; 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
1979	General guidance for selection controlling materials 1. Controlling base metal 2. Controlling weld metal (same heat of weld wire and lot of flux as beltline region controlling weld) 3. HAZ of base metal	5	12 Charpy 3 tension & fracture mechanics	18 Charpy 3 tension & fracture mechanics	Normal to major working direc'n	ΔT @ 30 ft-lb + RT_{NDT} ΔT @ 50 ft-lb & ΔT @ 35 mils for information only	5 capsules — first 4 capsules withdrawn at specific times; 5th capsule standby 4 capsules — first 3 capsules withdrawn at specific times; 4th capsule standby	Selected per ASTM E482 to measure integrated flux, fast neutron spectrum, & therm neutron spectrum	Same as above	1. Correlation monitor specimens are optional 2. Capsule neutron lead shall be between 1 and 3 3. Complete chemistry of test materials shall be determined 4. Add'l fracture toughness specimen per ASTM E636 shall be determined 5. Capsule and attachment design shall permit insertion of replacement capsules 6. Accelerated capsules optional 7. Test equipment shall be calibrated
1982	Same as above	5	Same as above	Same as above	Same as above	Same as above	Same as above	Same as above	Same as above	1. Charpy index temperatures, upper shelf energy determined from average curves

3. INTEGRATED REACTOR VESSEL SURVEILLANCE PROGRAM

The integrated surveillance program is more than a combination of eleven 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 RVSP correlates data from both power reactor surveillance monitoring and test reactor research programs. However, since the test reactor irradiations are not performed in B&W 177-FA 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 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; 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 materials 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 the same irradiation holder tubes as the regular plant-specific surveillance capsules. The data from these capsules will be compared with data obtained on the same material by 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 RVSP complies with ASTM E185-82 and also addresses the additional requirements of 10 CFR 50, Appendixes G and H. Among the additional requirements are those described below.

1. Test methods for fracture toughness tests must be submitted to and approved by the NRC prior to testing.

2. The effects of neutron radiation on the reference temperature and upper shelf energy of reactor vessel beltline materials are to be predicted from the results of pertinent radiation effects studies in addition to the results of the surveillance program.

3. A proposed withdrawal schedule must be submitted, along with a technical justification, to the NRC for approval prior to its implementation.

4. The design and location of holder tubes must permit the insertion of replacement surveillance capsules.

5. The withdrawal and test results of each capsule must be reported to the NRC within one year of capsule withdrawal.

6. 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.

7. Since the test materials in the capsules of the plant-specific RVSP were not selected in accordance with ASTM E185-73, the data to be generated are not necessarily applicable to any specific plant. That is, the materials monitored by the RVSP are not always the materials judged in 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 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.

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

3.1. Plant-Specific 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 that was designed to meet the requirements of the NRC and ASTM E185 at the time the programs were developed. The topical reports describing the appropriate RVSPs for B&W's 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
Midland Unit 1	None
Midland Unit 2	None

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

Each RVSP consists of six surveillance capsules, except for Midland 2, which has four capsules. There are two capsules in each holder tube. Four of the six capsules (Midland 2: three of four) are the prime data-collecting capsules, while the others 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 shows typical withdrawal schedules from E185-82.

For the first nine plants in the surveillance programs, 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, in varying combinations, are placed in these capsules: Charpy, tensile, and compact fracture toughness (CT). (Appendix C 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 toughness specimen is 0.5 inch thick, 1.25 by 1.20 inches, and conforms to the basic requirements of ASTM E399-81 and E813-81. The Midland 1 and 2 surveillance capsules are stainless steel cylinders approximately 2.6 feet long, 2.75 inches in outside diameter, and 2.5 inches in inside diameter. The capsules contain some combination of Charpy, tensile, and compact fracture toughness specimens. The Charpy V-notch specimens conform to ASTM E23-72. The tensile specimens are miniature (Charpy-sized), 2.165 inches in overall length. The compact fracture toughness specimens are 0.394 inch thick and 0.500 inch thick rectangular, and 0.936 round, and conform to the basic requirements of ASTM E399-81 and E813-81. The specimens are shown in Appendix C.

Specimen identity is maintained throughout the program by die-stamping the top and bottom of each specimen with an identification code (a combination of letters and numbers).

In addition to the specimens, each capsule contains dosimeters and temperature monitors. Figures 3-1 through 3-4 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 inert helium 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 eleven separate 177-FA reactors. Within this common network are five types of surveillance programs (types A, B, C, D, and E), in which nine capsule types (I-IX) are irradiated. Surveillance program A uses capsule types I and II; program B uses types III and IV; program C uses types V and VI; program D uses types VII, VIII, and IX; and program E uses types VII and IX.

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 five separate programs (A-E) and the capsule types (I-IX) 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. Therefore, a perforated design was selected for the holder tube.

The capsules are locked into the holder tube by a removal 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 (1/4T) 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 the first six 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 fluence gradient effects inside a capsule have not been discernable in material property correlations to date, rotation capability was not included in the new capsule holder tubes.)

3.1.2. Dosimeters

Dosimeters are placed in the specimen capsules to determine the actual neutron fluence levels experienced by the specimens. Each capsule of the first nine surveillance programs contains four dosimeter tubes, each tube accommodating six different flux wires; the capsules of the Midland surveillance programs contain seven dosimeter tubes, each tube accommodating five or seven different dosimeters. The dosimeter types are defined in Table 3-2.

Dosimeter tube placement within the capsules is shown in Figures 3-1 through 3-4. The tubes run the length of the specimen stacks so the actual fluence experienced by the specimens can be determined.

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

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 through 3-4.

3.1.4. Types of Surveillance Programs and Capsules

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

3.1.4.1. Surveillance Program A

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

Capsule Type I -- Capsule type I contains eight tensile specimens and 36 Charpy specimens. Tensile specimens are prepared from weld metal and base metal A in the longitudinal direction. Charpy specimens are 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 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 fracture toughness specimens 0.5 inch thick 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 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, base metal 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 CT 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 CT 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 were originally the upper and lower capsules in the holder tubes, respectively.

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.

3.1.4.4. Surveillance Program D

Surveillance program D consists of capsule types VII, VIII, and IX. The capsules of this program, specific to Midland 1, have not yet been constructed.

Capsule Type VII -- Capsule type VII contains twelve tensile specimens, 54 Charpy specimens, six 0.394TCT specimens, twelve 0.500TCT specimens, and nine 0.936-inch thick round compact fracture toughness (0.936TRCT) specimens. Tensile specimens are prepared for weld metal A and the base metal in the transverse direction. Charpy specimens are prepared for weld metal A, the HAZ in the transverse direction, and the base metal in the transverse direction. All CT specimens were prepared for weld metal A.

Capsule Type VIII -- Capsule type VIII contains twelve tensile specimens, 54 Charpy specimens, six 0.394TCT specimens, twelve 0.500TCT specimens, and nine 0.936TRCT specimens. Tensile specimens are prepared for weld metals A and B, and the base metal in the transverse direction. Charpy specimens are prepared for weld metals A and B, the HAZ in the transverse direction, and the base metal in the transverse direction. CT specimens of all three sizes were prepared for weld metals A and B.

Capsule Type IX -- Capsule type IX contains twelve tensile specimens, 54 Charpy specimens, six 0.394TCT specimens, twelve 0.500TCT specimens, and nine 0.936TRCT specimens. Tensile specimens are prepared for weld metal A and the base metal in the transverse direction. Charpy specimens are prepared for weld metal A, the HAZ in the transverse direction, and the base metal in the transverse direction. All CT specimens were prepared for the base metal in the transverse direction.

3.1.4.5. Surveillance Program E

Surveillance program E consists of capsule types VII and IX. The capsules of this program, specific to Midland 2, have been constructed but not yet installed in the holder tubes. The makeup of the capsule types has been discussed in section 3.1.4.4; only one weld metal is included.

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 (TMI-2), Crystal River Unit 3 (CR-3), and Davis-Besse Unit 1 (DB-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, like the Midland capsules, have an inside diameter of 2.5 inches and an effective length of 22.0 inches. Each research capsule contains Charpy, tensile, and CT 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 3-5 and 3-6. The type R-2 capsule represents an improved design over type R-1 since it utilizes miniature (Charpy size) tensile specimens. The miniature specimens allow the addition of five more tensile and three more CT specimens per capsule than 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 R-2. Table 3-5 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 E399 and E813 specimen geometry. The number of

Charpy and tensile specimens per weld per capsule is adequate to characterize the impact 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 CT specimens should be tested. The combination of CT specimens is believed to be adequate to confirm the toughness curves. The information generated by the research capsules and the Heavy Section Steel Technology (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 CT 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 original surveillance capsules, and are comparable to the Midland capsules. The research capsules have a larger diameter because they are used to irradiate relatively thick compact fracture specimens. The original surveillance capsules hold only eight 0.500-inch thick CT specimens. When the research capsules were designed, many uncertainties were associated with the measuring capacity of the smaller CT 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 for both the shell and end fitting is type 304 stainless steel. Aluminium spacers hold the specimens, dosimeters, and temperature monitors in place and fill the gaps within the capsule. The remaining spaces are filled with inert 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.

Whenever necessary, the individual capsules can be withdrawn and replaced with other capsules. The capsules are normally withdrawn during plant refueling. The capsule holder tube assembly permits remote removal and replacement of the capsules from the fuel handling bridge without removing the plenum assembly. When no capsules remain to be inserted, a dummy capsule, containing no specimens, will be inserted into a holder tube, or else the tube will be left empty. The tubes must contain either two capsules or none.

The type R-1 capsules were designed before the R-2s 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 five more tensile specimens and three additional CT specimens, which are the most important specimens included in the capsules. 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; the tensile properties also provide key input to fracture toughness tests and fracture analysis.

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 E185 as the minimum number required to obtain a full Charpy V-notch data curve. The size and number of CT specimens per weld are adequate for determination of the fracture toughness properties of the corresponding welds (based on current state-of-the-art). The tension, Charpy, and CT specimens are described in Appendix C. The materials contained in the capsules are described in Appendix B.

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 3-5 and 3-6 for capsule types R-1 and R-2.

3.2.3. 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 minimizes 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 3-7 illustrates the calculated vessel wall temperature distribution for steady-state normal operation. The temperature gradient is caused by gamma heating within the vessel wall and by the insulation of the reactor vessel outside wall. 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.

3.2.4. Dosimetry

Each capsule contains dosimeter tubes, which in turn contain neutron-detecting element wires of a sufficient variety to measure fast neutron fluence (time integrated flux), fast neutron spectrum, and thermal neutron fluence. A variety of neutron detecting elements was chosen in accordance with ASTM Standard Recommended Practice E419-73 and E482-82. The dosimeters are distributed throughout the capsule to measure the neutron fluence at various locations.

3.2.4.1. Dosimeters

Table 3-8 lists the neutron detecting elements and provides their energy range and shielding requirements. The gadolinium or shield thickness of 15 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 detecting elements, along with their shielding, are then stacked in aluminum holder tubes, which may contain from 5 to 10 elements.

3.2.4.2. 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 degree and 180 degree 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 RCT specimen stacks. Figures 3-4 and 3-5 show the exact locations of the dosimeters in the capsules.

3.2.4.3. 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 3-3 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 3-4 and 3-5 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 RCT specimen stack.

3.2.5. Insertion and Withdrawal Schedules

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

3.2.6. 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 larger 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 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 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. However, since most of the reactors have had at least one capsule withdrawn before the initiation of the integrated RVSP, this forms a basis for evaluating any abnormal behavior that may be attributable to the higher lead factor.

Tables 3-11 through 3-15 define the insertion and withdrawal schedules for each host and plant-specific 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 withdrawal schedules for the guest and host surveillance capsules and the research capsules are shown in Tables 3-11, 3-12, and 3-13; more information is presented in Appendix E. The withdrawal schedules for the two plant-specific programs are shown in Tables 3-14 and 3-15.

The TMI-2 incident has raised some unanswered questions for the integrated program. Six capsules are currently in TMI-2 holder tubes: one from the TMI-1 surveillance program, three from the TMI-2 surveillance program, and the two TMI2-LG research capsules. The TMI-2 surveillance program is dependent on the ultimate resolution of the TMI-2 plant status, although nothing prevents irradiation and testing of the TMI-2 capsules in support of the integrated RVSP. The condition of the specimens in the six capsules in TMI-2 will be determined after TMI-2 head removal. If the capsules are suitable for use, the TMI-1 capsule and the TMI2-LG capsules will probably be irradiated at Crystal River 3, although Davis-Besse 1 and the Midland plants are also possibilities.

3.4. 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, as shown in Figure 3-8.

The azimuthal locations of the holder tubes are shown in Figures 3-9 through 3-13 for the host and plant-specific reactors. Table 3-7 provides a list of the locations of the six research capsules and their azimuthal locations.

3.5. Supplementary Information

This report revision includes information requested by the NRC which was not in previous revisions. Appendix F details regulatory requirements of 10 CFR 50, Appendix H, and the B&W Owner's Group compliance. Appendix G shows estimated end-of-life RT_{NDT} values for the beltline weld metals in the reactor vessels. Appendix H provides information pertaining to the neutron dosimetry of the IRVSP.

Table 3-1. Recommended Withdrawal Schedules as per ASTM Specification E 185-82

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

Table 3-2. Surveillance Capsule Dosimeters

Neutron-sensitive element	Shield	Reaction cross-section threshold energy	Product Isotope
First nine programs[13-15]			
^{59}Co	Cd-Ag	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
Type DB (Midland 1,2)[16]			
^{237}Np	Gd	0.5 MeV	Appropriate fission products
^{238}U	Gd	1.1 MeV	Appropriate fission products
^{58}Ni	Gd	2.3 MeV	71d ^{58}Co
^{54}Fe	Gd	2.5 MeV	314d ^{54}Mn
^{63}Cu	Gd	6.1 MeV	5.3 yr ^{60}Co
Type DC (Midland 1,2)[16]			
^{59}Co	Gd	0.4 eV	5.3 yr ^{60}Co
^{237}Np	Gd	0.5 MeV	Appropriate fission products
^{238}U	Gd	1.1 MeV	Appropriate fission products
^{58}Ni	Gd	2.3 MeV	71d ^{58}Co
^{54}Fe	Gd	2.5 MeV	314d ^{54}Mn
^{63}Cu	Gd	6.1 MeV	5.3 yr ^{60}Co
^{59}Co	None	All levels	5.3 yr ^{60}Co

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 has been verified by the fabricator from its final form and reported and documented.

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

Capsule ID	type	Table of mat'l specs	Table of capsule specs	Date tested	Applicable report
Oconee Unit 1					
A	I	A-1	D-2	Aug 84	BAW-1837
B	II	A-1	D-2	--	
C	I	A-1	D-2	--	
D	II	A-1	D-2	--	
E	I	A-1	D-2	June 76	BAW-1436
F	II	A-1	D-2	July 75	BAW-1421, Rev. 1
Topical Report BAW-10006A, Rev 3					
Three Mile Island Unit 1					
A	I	A-4	D-5	--	
B	II	A-4	D-5	--	
C	I	A-4	D-5	--	
D	II	A-4	D-5	--	
E	I	A-4	D-5	June 76	BAW-1439
F	II	A-4	D-5	--	
Topical Report BAW-10006A, Rev 3					
Crystal River Unit 3					
A	III	A-7	D-7	--	
B	IV	A-7	D-7	June 81	BAW-1679
C	III	A-7	D-7	--	
D	IV	A-7	D-7	--	
E	III	A-7	D-7	--	
F	IV	A-7	D-7	--	
Topical Report BAW-10100A					
Rancho Seco Unit 1					
A	III	A-8	D-9	--	
B	IV	A-8	D-9	Feb 82	BAW-1702
C	III	A-8	D-9	--	
D	IV	A-8	D-9	Oct 83	BAW-1792
E	III	A-8	D-9	--	
F	IV	A-8	D-9	--	
Topical Report BAW-10100A					

Table 3-4. (Cont'd)

Capsule ID	type	Table of mat'l specs	Table of capsule specs	Date tested	Applicable report
Oconee Unit 3					
A	V	A-3	D-4	July 76	BAW-1438
B	VI	A-3	D-4	Oct 81	BAW-1697
C	V	A-3	D-4	--	
D	VI	A-3	D-4	--	
E	V	A-3	D-4	--	
F	VI	A-3	D-4	--	
Topical Report BAW-10100A*					
Oconee Unit 2					
A	I	A-2	D-3	Dec 81	BAW-1699
B	II	A-2	D-3	--	
C	I	A-2	D-3	June 76	BAW-1437
D	II	A-2	D-3	--	
E	I	A-2	D-3	--	
F	II	A-2	D-3	--	
Topical Report BAW-10006A, Rev 3					
Three Mile Island Unit 2					
A	III	A-5	D-6	--	
B	IV	A-5	D-6	--	
C	III	A-5	D-6	--	
D	IV	A-5	D-6	--	
E	III	A-5	D-6	--	
F	IV	A-5	D-6	--	
Topical Report BAW-10100A					
Arkansas Nuclear One Unit 1					
A	I	A-6	D-8	July 84	BAW-1836
B	II	A-6	D-8	Nov 81	BAW-1698
C	I	A-6	D-8	--	
D	II	A-6	D-8	--	
E	I	A-6	D-8	Aug 76	BAW-1440
F	II	A-6	D-8	--	
Topical Report BAW-10006A. Rev 3					

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

Table 3-4. (Cont'd)

Capsule ID	type	Table of mat'l specs	Table of capsule specs	Date tested	Applicable report
Davis-Besse Unit 1					
A	III	A-9	D-10	--	
B	IV	A-9	D-10	May 1984	BAW-1834
C	III	A-9	D-10	--	
D	IV	A-9	D-10	--	
E	III	A-9	D-10	--	
F	IV	A-9	D-10	Jan 82	BAW-1701
Topical Report BAW-10100A					
Midland Unit 1					
A	VII	A-10	D-11	--	
B	VII	A-10	D-11	--	
C	VIII	A-10	D-11	--	
D	IX	A-10	D-11	--	
E	IX	A-10	D-11	--	
F	IX	A-10	D-11	--	
No applicable topical report					
Midland Unit 2					
A	VII	A-11	D-12	--	
B	VII	A-11	D-12	--	
C	IX	A-11	D-12	--	
D	IX	A-11	D-12	--	
No applicable topical report					

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

Weld metal ID per capsule	Specimens: 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					
W3	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
0.500 TCT	12	12
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
DB1-LG1	R-2	Davis-Besse 1	Upper	10.9° off W
DB1-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. Research Capsule Dosimeters

Neutron-sensitive element	Shield	Reaction cross-section threshold energy	Product Isotope
Long tube (TMI2-LG1, -LG2 only)(17)			
^{59}Co coated with Cd	none	0.4 eV	5.3 yr ^{60}Co
^{237}Np	Gd	0.5 MeV	Appropriate fission products
^{238}U	Gd	1.1 MeV	Appropriate fission products
^{58}Ni	Gd	2.3 MeV	71d ^{58}Co
^{54}Fe	Gd	2.5 MeV	314d ^{54}Mn
^{63}Cu	Gd	6.1 MeV	5.3 yr ^{60}Co
^{59}Co	none	All levels	5.3 yr ^{60}Co
Short tube (TMI2-LG1, -LG2 only)(17)			
^{237}Np	Gd	0.5 MeV	Appropriate fission products
^{238}U	Gd	1.1 MeV	Appropriate fission products
^{54}Fe	Gd	2.5 MeV	314d ^{54}Mn
Type DA (CR3-LG1, -LG2; DB1-LG1, -LG2)(18,19)			
^{59}Co	Gd	0.4 eV	5.3 yr ^{60}Co
^{237}Np	Gd	0.5 MeV	Appropriate fission products
^{238}U	Gd	1.1 MeV	Appropriate fission products
^{58}Ni	Gd	2.3 MeV	71d ^{58}Co
^{54}Fe	none	2.5 MeV	314d ^{54}Mn
^{59}Co	none	All levels	5.3 yr ^{60}Co
Type DB (CR3-LG1, -LG2; DB1-LG1, -LG2)(18,19)			
^{237}Np	Gd	0.5 MeV	Appropriate fission products
^{238}U	Gd	1.1 Mev	Appropriate fission products
^{58}Ni	Gd	2.3 MeV	71d ^{58}Co
^{54}Fe	Gd	2.5 MeV	314d ^{54}Mn
^{63}Cu	Gd	6.1 MeV	5.3 yr ^{60}Co

Table 3-8. (Cont'd)

Neutron-sensitive element	Shield	Reaction cross-section threshold energy	Product Isotope
Type DC (CR3-LG1, -LG2; DB1-LG1, -LG2)(18,19)			
^{59}Co	Gd	0.4 eV	5.3 yr ^{60}Co
^{237}Np	Gd	0.5 MeV	Appropriate fission products
^{238}U	Gd	1.1 MeV	Appropriate fission products
^{58}Ni	Gd	2.3 MeV	71d ^{58}Co
^{54}Fe	Gd	2.5 MeV	314d ^{54}Mn
^{63}Cu	Gd	6.1 MeV	5.3 yr ^{60}Co
^{59}Co	none	All levels	5.3 yr ^{60}Co

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	0.936 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	6	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 Schedule for Crystal River Unit 3

Holder tube	Location in holder tube	Remove	Insert
Installed at Initial Fuel Load			
XW	Top		CR3-B (WC)
XW	Bottom		CR3-D(WC)
End of First Fuel Cycle (1A)			
WZ	Top		CR3-LG1(WC)
WZ	Bottom		CR3-LG2(WC)
ZY	Top		CR3-C(W)
ZY	Bottom		CR3-A(W)
YZ	Top		OCII-A(W)
YZ	Bottom		OCI-A(W)
YX	Top		OCII-E(W)
YX	Bottom		OCIII-D(W)
XW	Top	CR3-B(WC)	CR3-E(W)
WX	Top		OCIII-B(W)
WX	Bottom		CR3-F(WC)
End of Second Cycle			
YZ	Top	OCII-A(W)	OCI-C(W)
WX	Top	OCIII-B(W)	TMI1-C(W)
End of Fourth Cycle			
YZ	Bottom	OCI-A(W)	OCI-B
WZ	Top	CR3-LG1(WC)	None
WZ	Bottom	CR3-LG2(WC) (WZ now empty)	None
End of Fifth Cycle			
WX	Top	TMI1-C(W)	OCIII-C(W)
XW	Bottom	CR3-D(WC)	TMI1-B
ZY	Top	CR3-C(W)	OCIII-F(W)
WZ	Top	None	OCII-B
WZ	Bottom	None (WZ no longer empty)	CR3-LG2(WC)
End of Sixth Cycle			
YX	Top	OCII-E(W)	TMI1-D
YX	Bottom	OCIII-D(W)	OCI-D
YZ	Bottom	OCI-B[a]	OCIII-E(W)

Table 3-11. (Cont'd)

Holder tube	Location in holder tube	Remove	Insert
End of Seventh Cycle			
XW	Bottom	TMI1-B[a]	OCII-D
WX	Bottom	CR3-F(WC)	TMI1-F
YZ	Top	OCI-C(W)	OCII-F
End of Eighth Cycle			
WX	Top	OCIII-C(W)[a]	Dummy
WZ	Top	OCII-B[a]	None
WZ	Bottom	CR3-LG2(WC) (WZ now empty)	None
End of Ninth Cycle			
XW	Top	CR3-E(W)[b]	OCIII-F from ZY top
ZY	Top	OCIII-F to XW top	None
ZY	Bottom	CR3-A(W)[a] (ZY now empty)	None
End of Eleventh Cycle			
WX	Top	Dummy	None
WX	Bottom	TMI1-F to YZ bottom	None
YX	Top	TMI1-D	None
YX	Bottom	OCI-D[a]	None
YZ	Bottom	OCIII-E(W)[a] (WX and YX now empty)	TMI1-F from YZ bottom
End of Twelfth Cycle			
XW	Top	OCIII-F(W)[a]	Dummy
YZ	Bottom	TMI1-F[a]	None
YZ	Top	OCII-F[a] (YZ now empty)	None
End of Fourteenth Cycle			
XW	Top	Dummy	None
XW	Bottom	OCII-D[a] (XW now empty)	None

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Table 3-11. (Cont'd)

(a) Capsule may not be tested upon removal.

(b) Capsule may remain in place until fluence $\approx$5E19 n/cm^2. This will require a schedule change.

(W) Capsule contains weld metal specimens.

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

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

Holder tube	Location in holder tube	Remove	Insert
Installed at Initial Fuel Load			
WZ	Top		AN1-B
WZ	Bottom		RS1-B(WC)
ZY	Top		TE1-B(WC)
ZY	Bottom		TE1-F(WC)
YZ	Top		AN1-A(W)
YZ	Bottom		AN1-C(W)
YX	Top		RS1-D(WC)
YX	Bottom		TE1-C(W)
XW	Top		TE1-D(WC)
XW	Bottom		RS1-C(W)
WX	Top		TE1-A(W)
WX	Bottom		RS1-F(WC)
End of First Cycle			
WZ	Top	AN1-B	DB1-LG1(WC)
WZ	Bottom	RS1-B(WC)	RS1-E(W)
ZY	Bottom	TE1-F(WC)	DB1-LG2(WC)
End of Second Cycle			
YX	Top	RS1-D(WC)	RS1-A(W)
End of Third Cycle			
YZ	Top	AN1-A(W)	AN1-D
ZY	Top	TE1-B(WC)	TE1-E(W)

Table 3-12. (Cont'd)

Holder tube	Location in holder tube	Remove	Insert
End of Fourth Cycle			
YX	Top	RS1-A(W)[a]	AN1-F
WZ	Top	DB1-LG1(WC)	RS1-F from WX bottom
WX	Top	TE1-A(W)	None
WX	Bottom	RS1-F to WZ top (WX now empty)	None
End of Fifth Cycle			
WZ	Top	RS1-F(WC)	AN1-D from YZ top
YZ	Top	AN1-D to WZ top	None
YZ	Bottom	AN1-C(W) (YZ now empty)	None
End of Sixth Cycle			
WZ	Bottom	RS1-E(W)[a]	AN1-F from YX top
YX	Top	AN1-F to WZ bottom	None
YX	Bottom	TE1-C(W)[a] (YX now empty)	None
End of Seventh Cycle			
XW	Bottom	RS1-C(W)[a]	Dummy
End of Eighth Cycle			
WZ	Top	AN1-D[a]	Dummy from XW bottom
XW	Top	TE1-D(WC)	None
XW	Bottom	Dummy to WZ top (XW now empty)	None
End of Ninth Cycle			
ZY	Bottom	DB1-LG2(WC)	Dummy from WZ top
WZ	Top	Dummy to ZY bottom	None
WZ	Bottom	AN1-F[a] (WZ now empty)	None
End of Eleventh Cycle			
ZY	Top	TE1-E(W)[a]	None
ZY	Bottom	Dummy (ZY now empty)	None

Table 3-12. (Cont'd)

(a) Capsule may not be tested upon removal.

(W) Capsule contains weld metal specimens.

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

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

Holder tube	Location in holder tube	Remove	Insert
At Initial Fuel Load			
WX	Top		TMI2-LG1(WC)
WX	Bottom		TMI2-B(WC)
XW	Top		TMI2-A(W)
XW	Bottom		TMI2-D(WC)
ZY	Top		TMI1-A(W)
ZY	Bottom		TMI2-LG2(WC)

(W) Capsule contains weld metal specimens.

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

Note: At this time the capsules inside TMI-2 and the remaining capsules of the TMI-2 surveillance program (TMI2-C, E, and F) are of uncertain status. The six capsules listed above will be evaluated at the earliest possible date. Should they prove suitable for further use, TMI1-A, TMI2-LG1, and TMI2-LG2 will be worked into the withdrawal schedule of another host reactor. The six TMI-2 surveillance program capsules' status will be determined at a later date.

Table 3-14. Surveillance Capsule Withdrawal Schedule for Midland Unit 1

All capsules inserted prior to first cycle/plant startup

Capsule ID	Holder tube	Location in holder tube	Time of removal
MD1-A	WX	Top	End of first cycle
MD1-D	WX	Bottom	End of third cycle
MD1-B	XW	Top	End of seventh cycle
MD1-E	XW	Bottom	End of ninth cycle
MD1-C	ZY	Top	Standby
MD1-F	ZY	Bottom	Standby

Table 3-15. Surveillance Capsule Withdrawal Schedule for Midland Unit 2

All capsules inserted prior to first cycle/plant startup

Capsule ID	Holder tube	Location in holder tube	Time of removal
MD2-A	XW	Top	End of first cycle
MD2-C	XW	Bottom	End of fifth cycle
MD2-B	ZY	Top	End of ninth cycle
MD2-D	ZY	Bottom	Standby

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

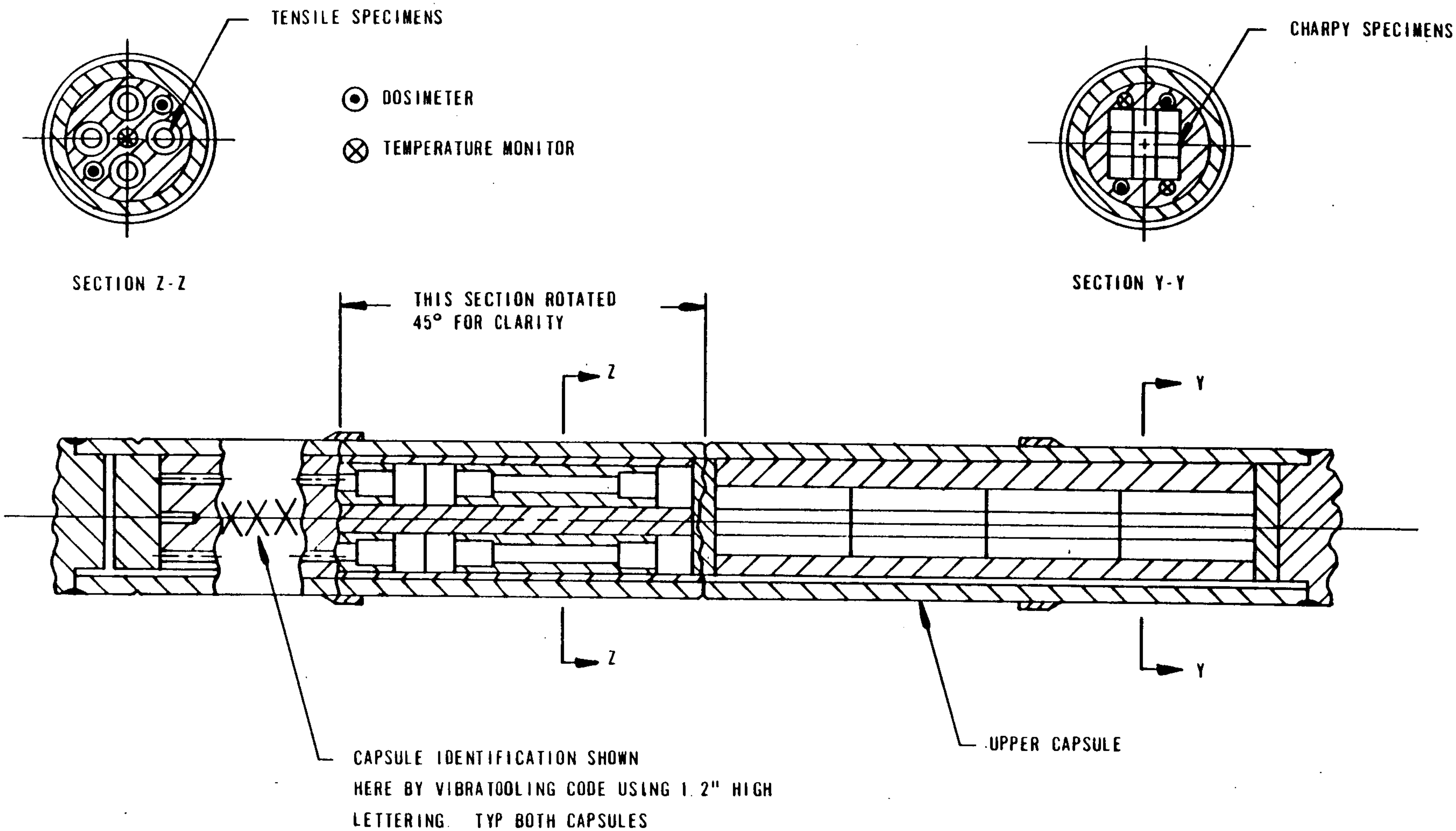


Figure 3-2. Surveillance Capsule Arrangement -- Type IV

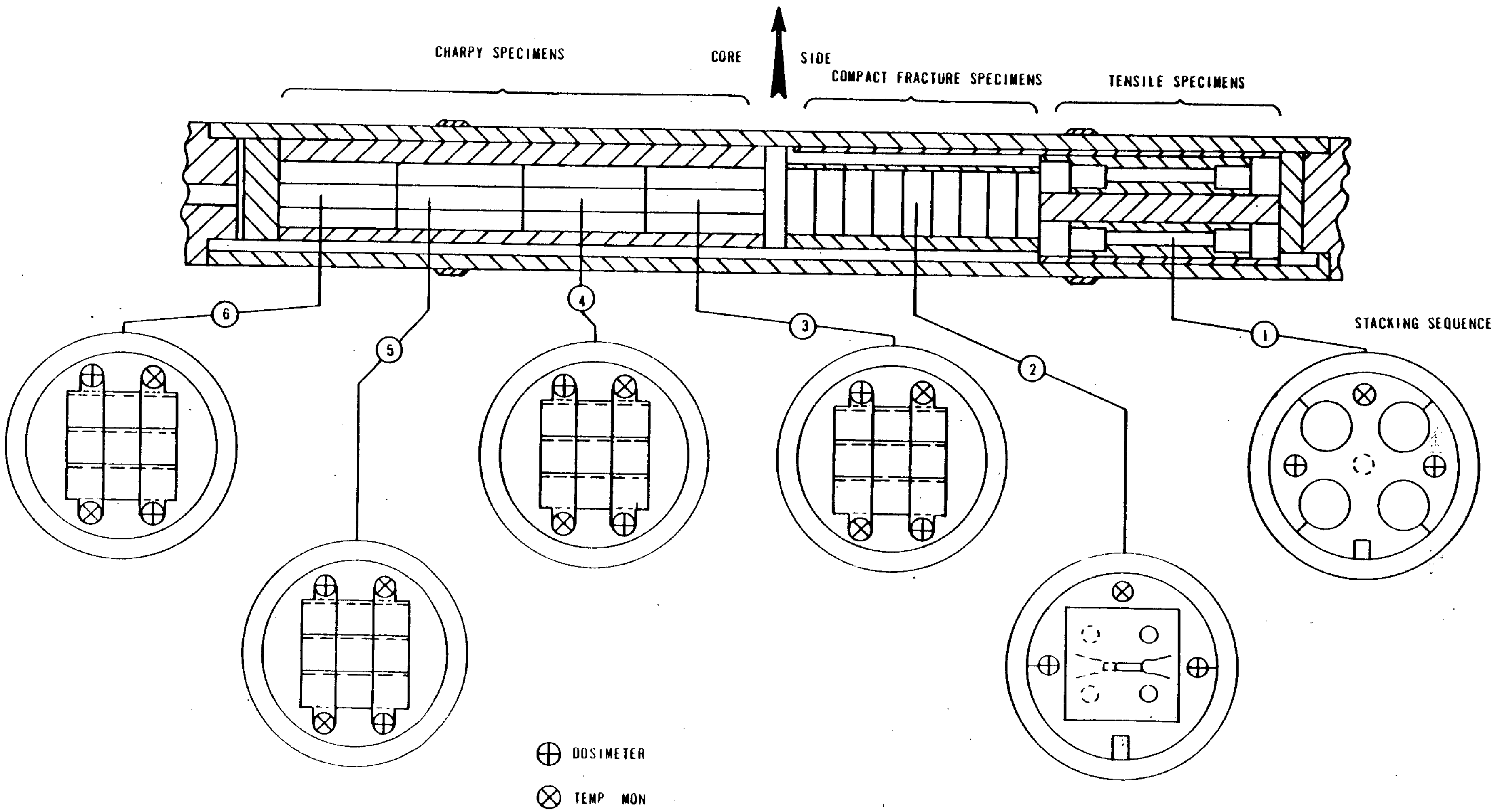


Figure 3-3. Surveillance Capsule Arrangement -- Types III, V, and VI

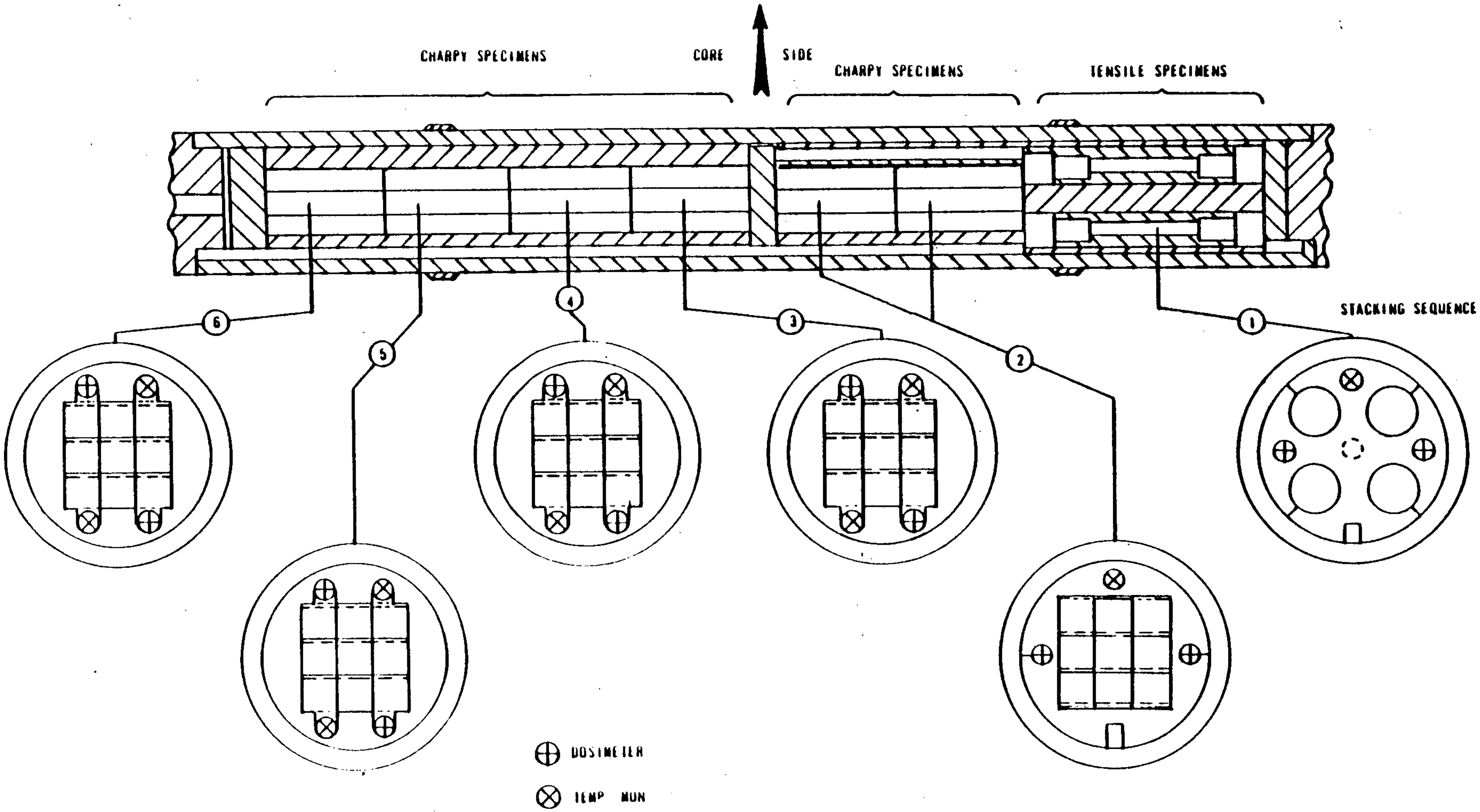
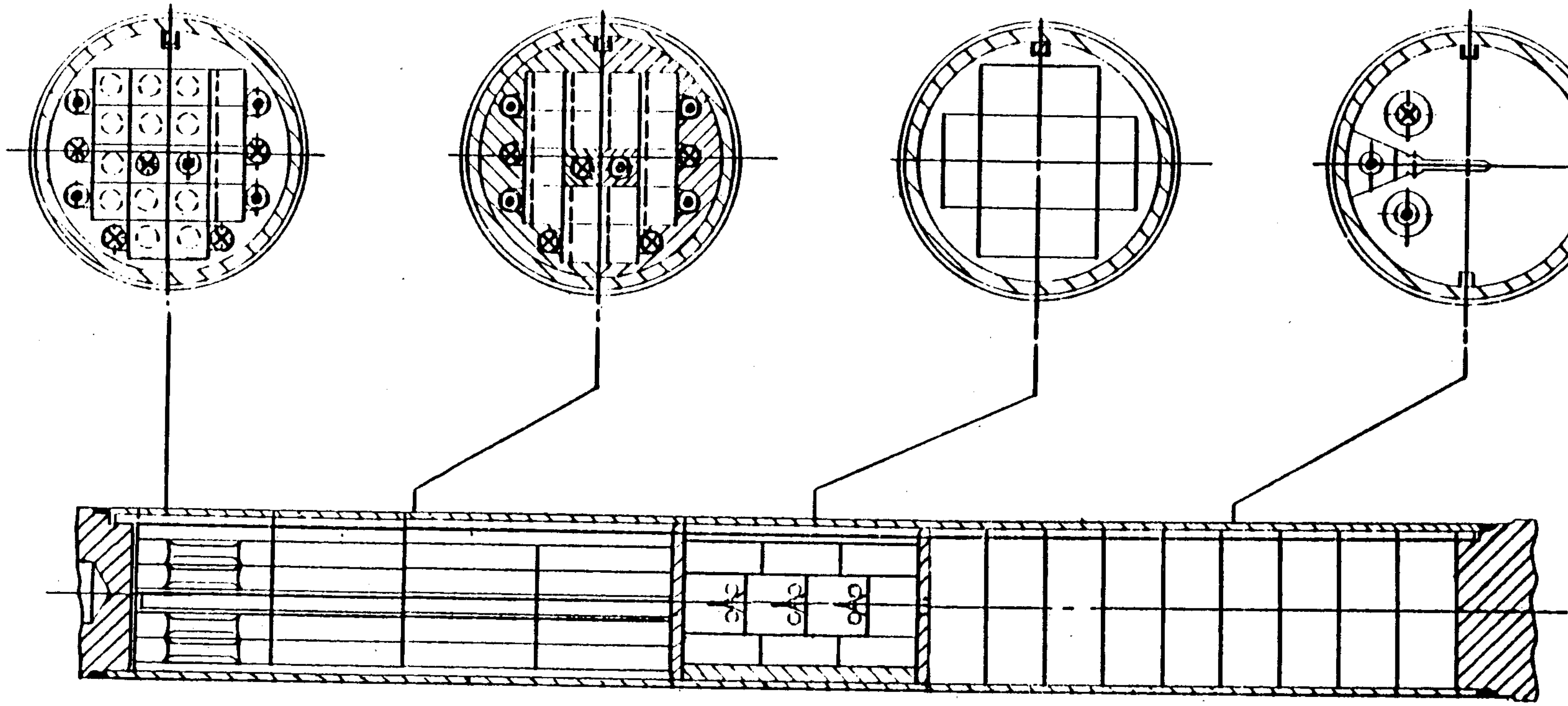


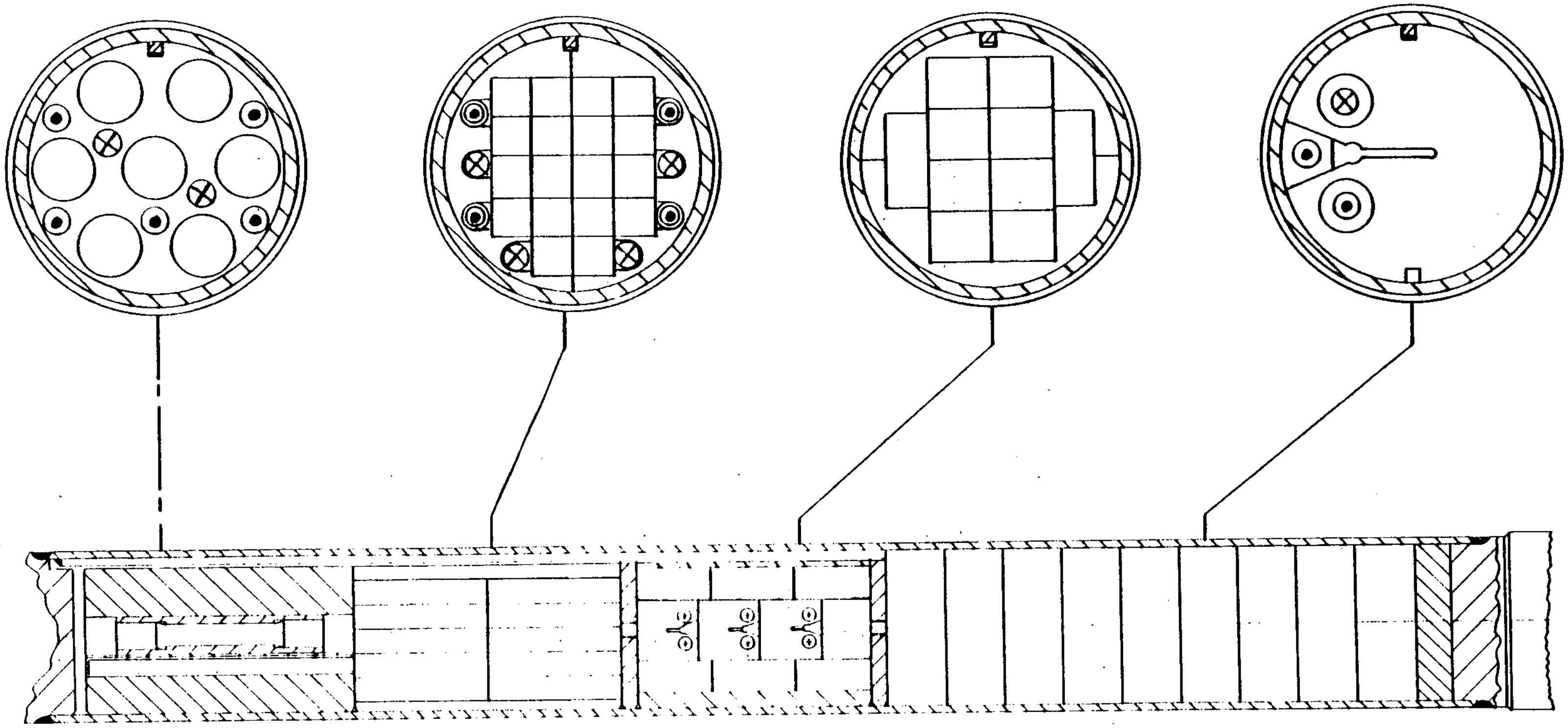
Figure 3-4. Midland 1 and 2 Surveillance Capsule Arrangement -- Types VII, VIII, and IX



DOSIMETER

TEMPERATURE MONITOR

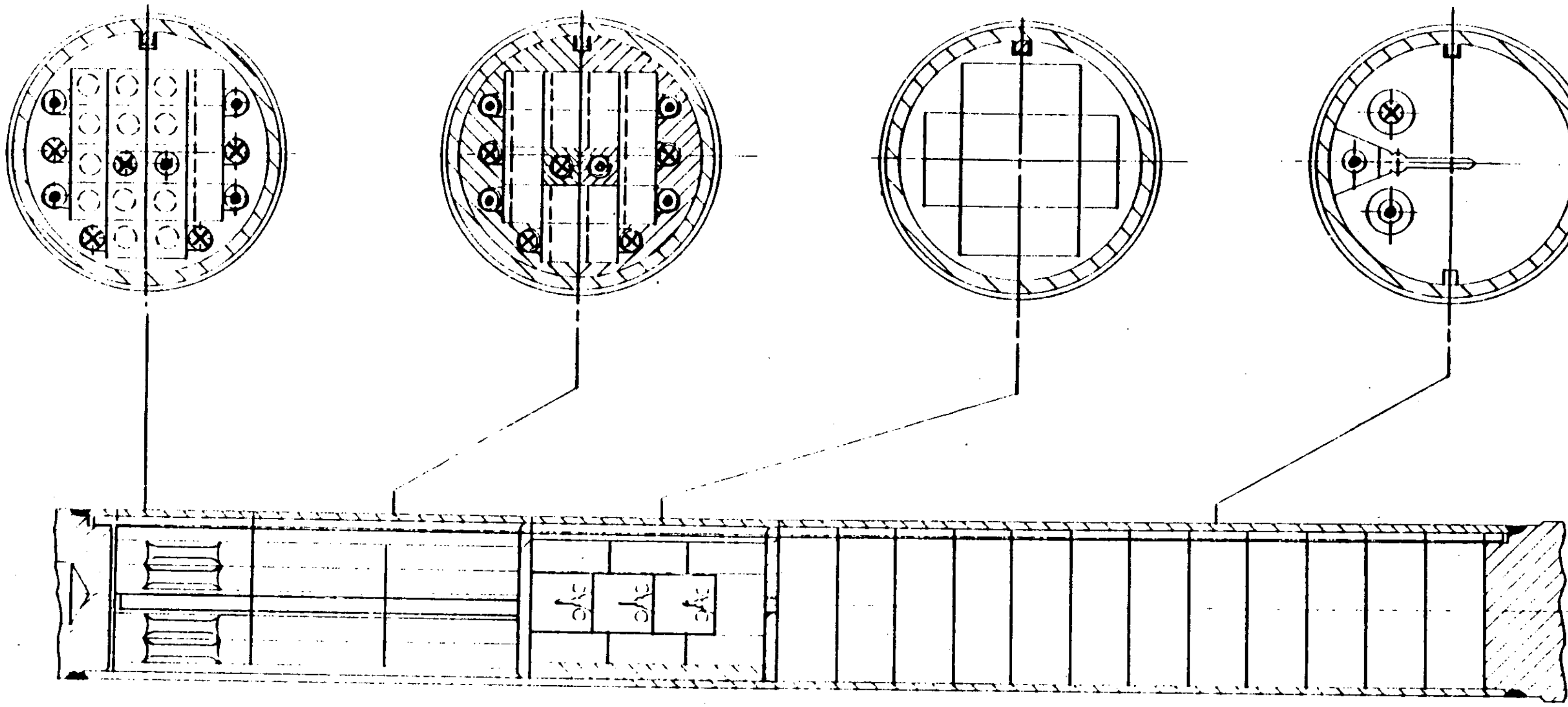
Figure 3-5. Research Capsule, Type R-1



⊙ DOSIMETER

⊗ TEMPERATURE MONITOR

Figure 3-6. Research Capsule, Type R-2



⊙ DOSIMETER

⊗ TEMPERATURE MONITOR

Figure 3-7. 177-FA Reactor Vessel Through-Thickness Temperature Distribution at Steady-State Normal Operation

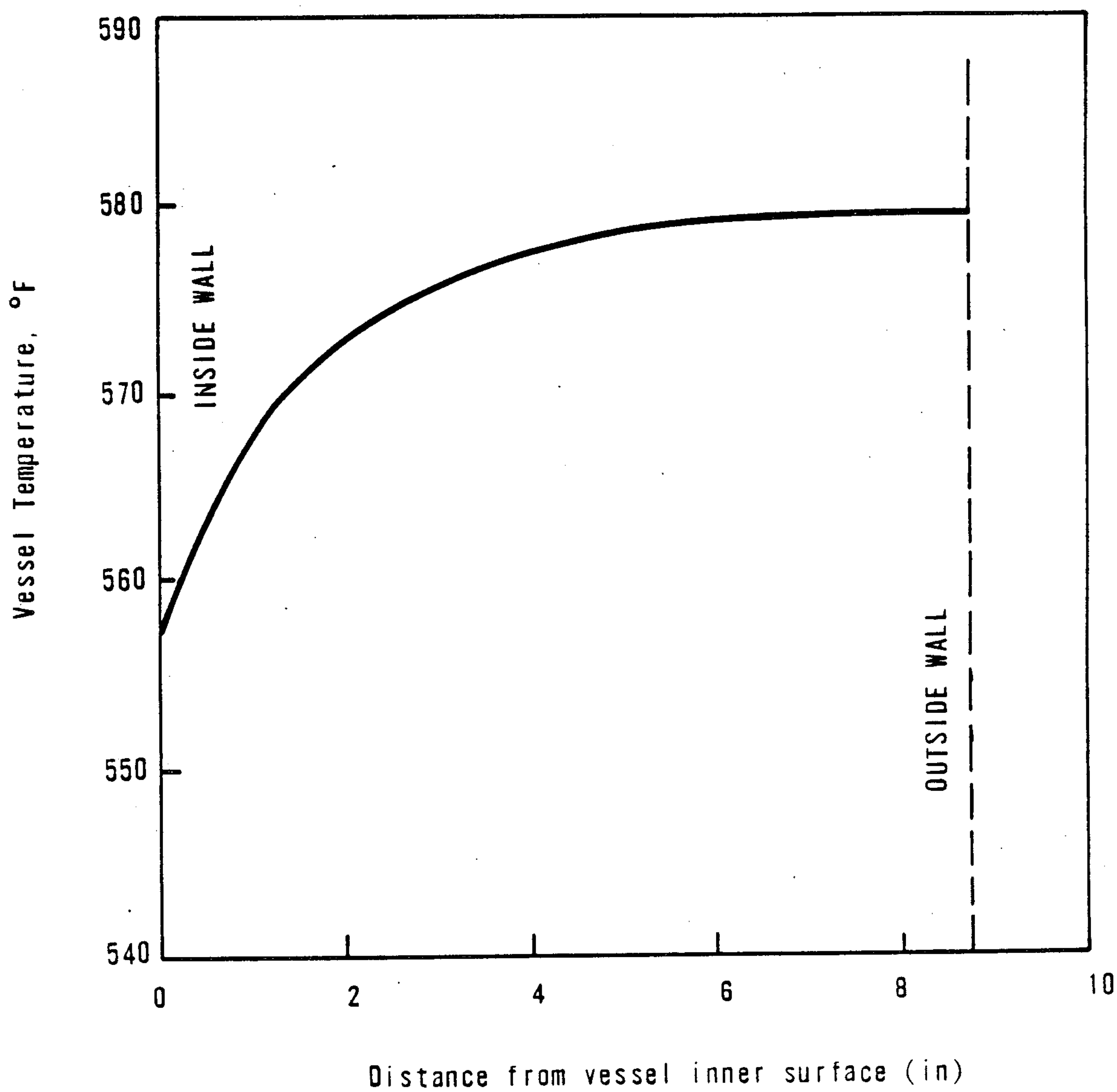


Figure 3-8. Reactor Vessel Arrangement Showing Current Surveillance Capsule Holder Tube Locations

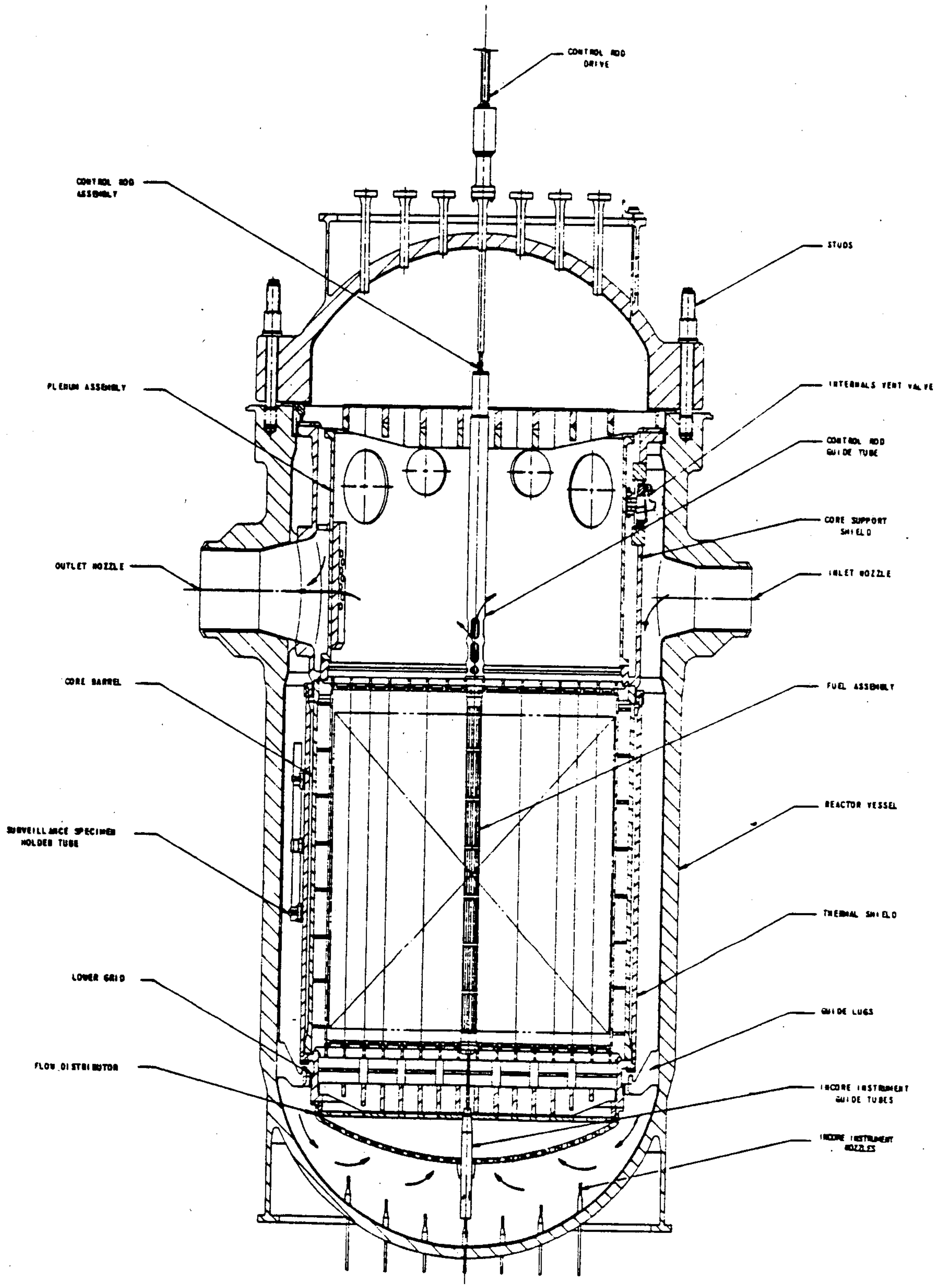


Figure 3-9. Surveillance Capsule Holder Tube Location and Identification for Davis Besse Unit 1

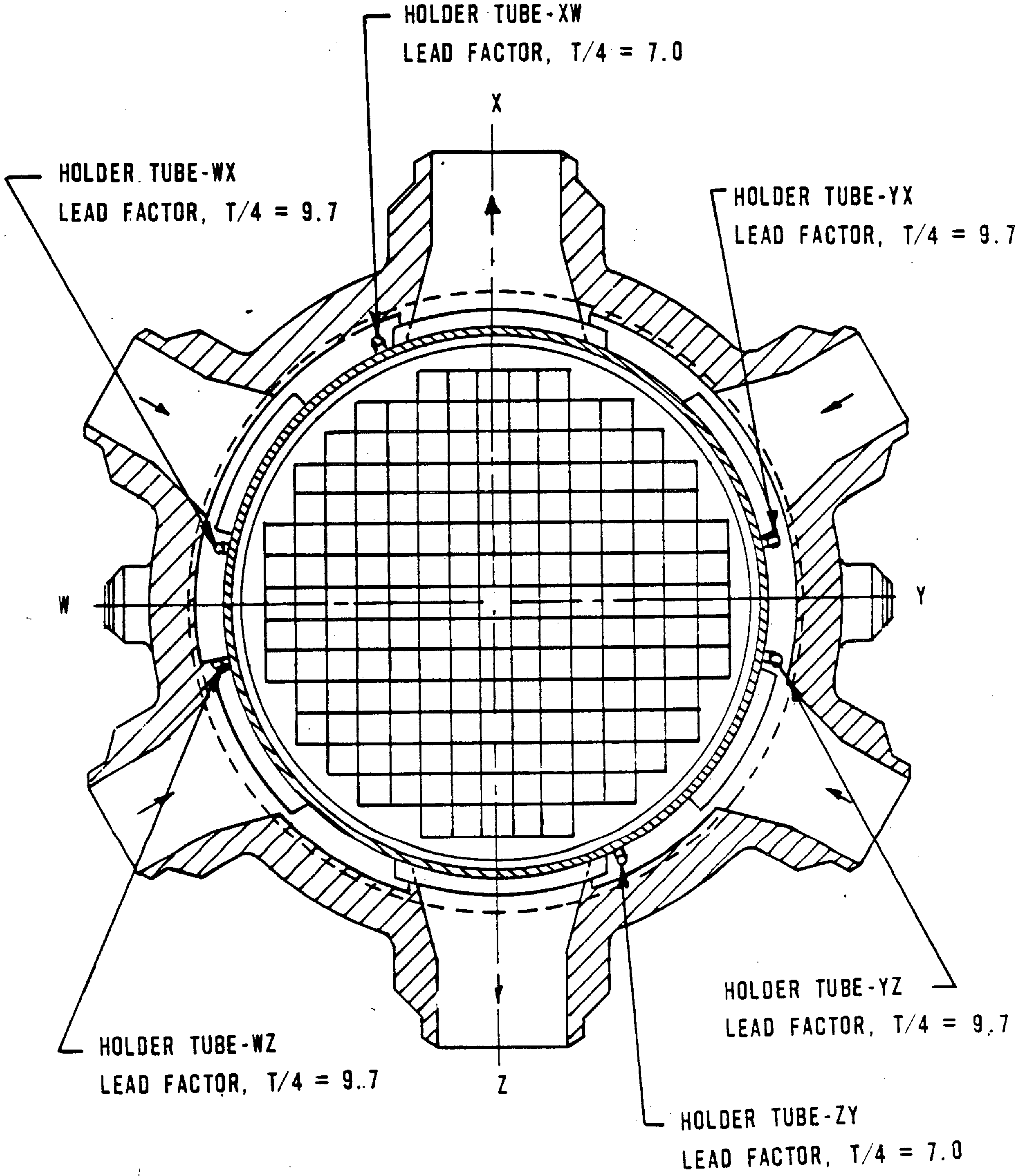


Figure 3-10. Surveillance Capsule Holder Tube Location and Identification for Crystal River Unit 3

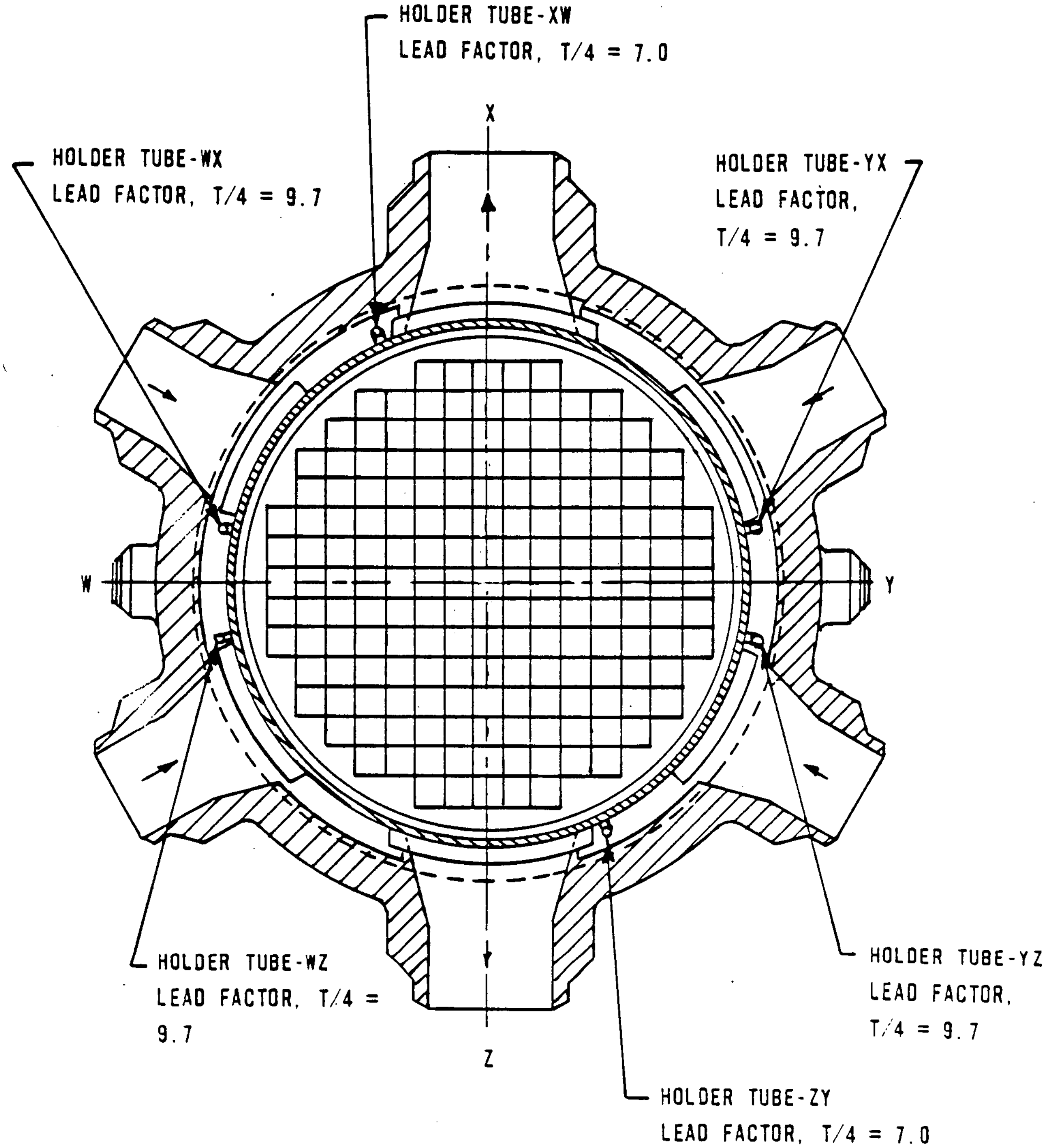


Figure 3-11. Surveillance Capsule Holder Tube Location and Identification for Three Mile Island Unit 2

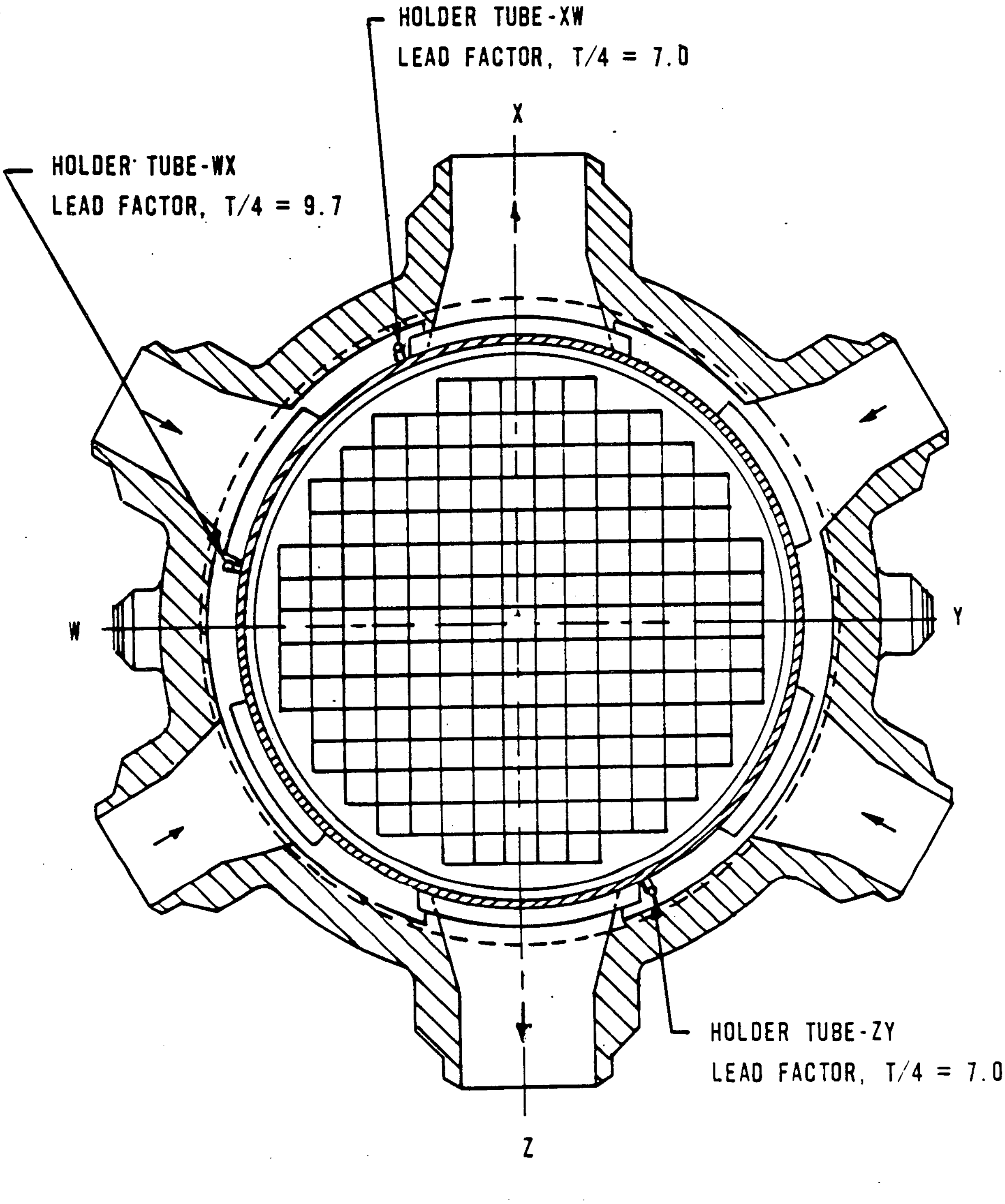


Figure 3-12. Surveillance Capsule Holder Tube Location and Identification for Midland Unit 1

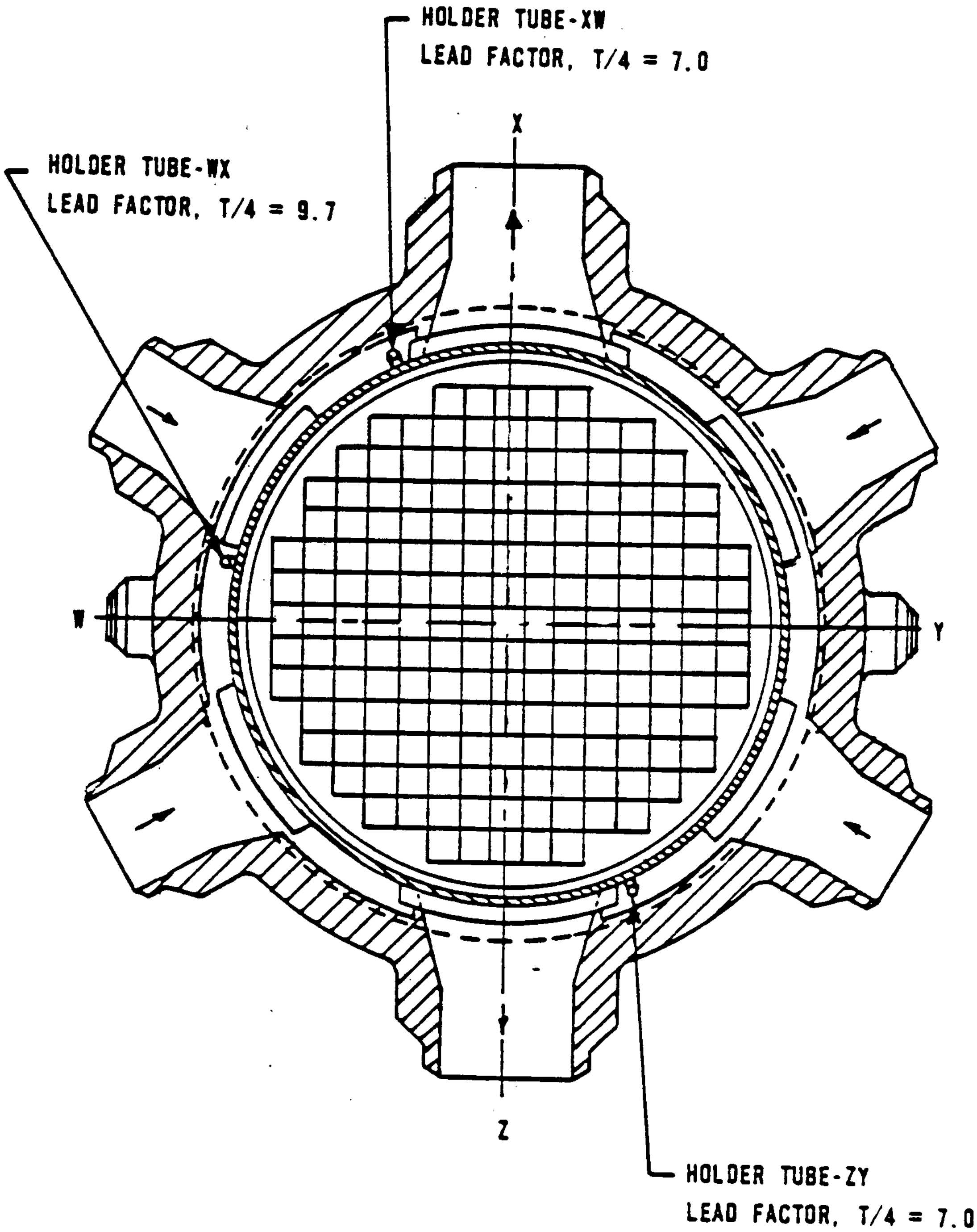
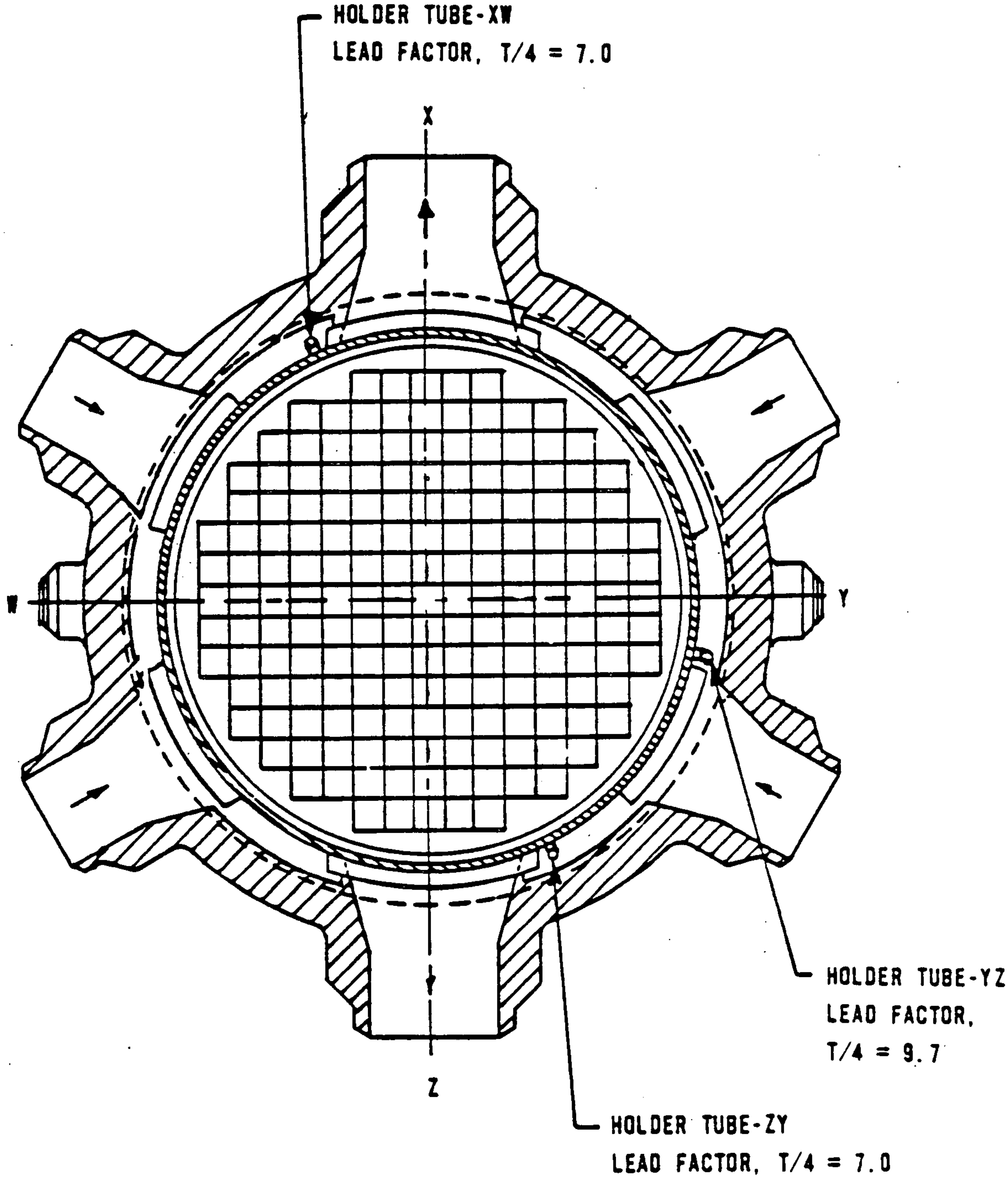


Figure 3-13. Surveillance Capsule Holder Tube Location and Identification for Midland Unit 2



4. CERTIFICATION

This report is an accurate description of the integrated reactor vessel materials surveillance program designed in accordance with the requirements of 10 CFR 50, Appendixes G and H.

29 May 1985

A. L. Lowe, Jr., P.E. — Date

Project Technical Manager

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

5/29/85

L. B. Gross, P.E. — Date

Materials and Chemistry

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APPENDIX A

Description and Properties of RVSP Materials

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

	Chemical composition, %									Impact properties				
Material ID	C	Mn	P	S	Si	Ni	Cr	Mo	Cu	T_{NDT}, F	RT_{NDT}, F	C_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[a]	0.079	1.47	0.016	0.015	0.54	0.59	0.066	0.40	0.32	-50	0	65	83.0	66.0

				Heat treatment[b]		
Material ID	Heat No.	Spec No.	Supplier	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

(a) See BAW-1500 listed under General References -- Babcock & Wilcox reports in section 5.

(b) 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.

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Table A-2. Oconee 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	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[a]	0.11	1.55	0.022	0.010	0.65	0.58	0.089	0.39	0.36	-20	10	68	87.5	72.5

Material ID	Heat No.	Spec No.	Supplier	Heat treatment[b] Austenitizing	Tempering	Stress relief
Base metal A	3P-2359	SA 508 Cl.2	LADISH	1640F ± 20F held at color 4 h, 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

[a]See BAW-1500 listed under General References -- Babcock & Wilcox reports in section 5.

[b]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.

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	40	151	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	40	116	84.8	60.8
Weld metal[(a)]	0.083	1.63	0.017	0.012	0.61	0.58	0.096	0.39	0.30	10	10	66	87.5	72.5

Material ID	Heat No.	Spec No.	Supplier	Heat treatment[(b)]: 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 ± 25F 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	1100-1150F for 30 h, furnace-cooled

(a) See BAW-1500 listed under General References -- Babcock & Wilcox reports in section 5.

(b) 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.

Table A-4. TMI-1 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	1.36	0.010	0.017	0.23	0.57	--	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[a]	0.090	1.62	0.014	0.015	0.46	0.66	0.10	0.40	0.33	-20	-14	81	80.75	66.5

Material ID	Heat No.	Spec No.	Supplier	Heat treatment[b]: Austenitizing	Tempering	Stress relief
Base metal A	C2789-2	SA 302 Gr B	Lukens	1600-1650F for 9.5 h, brine quench 1200-1225F for 9.5 h, brine quench 1600-1650F for 9.5 h, brine quench 1600-1650F for 9.5 h, brine quench 1510-1535F for 5 h, brine quench 1200-1225F for 5 h, brine quench		1100-1150F for 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

(a) See BAW-1500 listed under General References -- Babcock & Wilcox reports in section 5.

(b) 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.

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Table A-5. TMI-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_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[a]	0.075	1.68	0.015	0.013	0.49	0.63	0.13	0.39	0.28	-30	19	84	81.0	NA

				Heat treatment[b]		
Material ID	Heat No.	Spec No.	Supplier	Austenitizing	Tempering	Stress relief
Base metal A	C1946-2	SA 533 Tp B Cl. 1	Lukens	1600-1650F for 9.5 h, brine quench	1180-1200F for 4.5 h, brine quench	1100-1150F for 40 h, furnace-cooled
Base metal B	C1937-2	SA 533 Tp 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 50 h, furnace-cooled

(a) See BAW-1500 listed under General References -- Babcock & Wilcox reports in section 5.

(b) 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.

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	65.7
Weld metal(a)	0.090	1.49	0.016	0.016	0.51	0.59	0.060	0.39	0.28	-20	30	73	82.0	65.0

Material ID	Heat No.	Spec No.	Supplier	Heat treatment(b)		
				Austenitizing	Tempering	Stress relief
Base metal A	C5114-1	SA 533 Tp 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 Tp B Cl. 1	Lukens	Same as above	Same as above	Same as above
Weld metal	WF-193	NA	NA	NA	NA	1100-1150F for 29 h, furnace-cooled

(a)See BAW-1500 listed under General References -- Babcock & Wilcox reports in section 5.

(b)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.

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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 A[(a)]	0.08	1.65	0.021	0.013	1.00	0.10	0.073	0.45	0.39	-20	52	79	93.8	77.1
Weld metal B[(b)]	0.10	1.57	0.018	0.009	0.54	0.60	0.094	0.43	0.35	-50	43	63	87.5	72.5

Material ID	Heat No.	Spec No.	Supplier	Heat treatment[(c)]		
				Austenitizing	Tempering	Stress relief
Base metal A	C4344-1	SA 533 Tp B Cl. 1	Lukens	1650F-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 Tp B Cl. 1	Lukens	Same as above	1100F, min, held 0.5 h/in./min. air cooled	Same as above
Weld metal A	Atypical	NA	NA	NA	NA	1100-1150F for 27 h, furnace-cooled
Weld metal B	WF-209-1	NA	NA	NA	NA	1100-1150F for 27 h, furnace-cooled

(a) See BAW-10144 listed under General References -- Babcock & Wilcox reports in section 5.

(b) See BAW-1500 listed under General References -- Babcock & Wilcox reports in section 5.

(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.

Table A-8. Rancho Seco 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	0.20	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[a]	0.090	1.49	0.016	0.016	0.51	0.59	0.060	0.39	0.28	-100	15	66	82.0	65.0

Material ID	Heat No.	Spec No.	Supplier	Heat treatment[b] Austenitizing	Tempering	Stress relief
Base metal A	C5070-1	SA 533 Tp 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 Tp B Cl. 1	Lukens	Same as above	Same as above	Same as above
Weld metal	WF-193	NA	NA	NA	NA	1100-1150F for 28 h, furnace-cooled

(a)See BAW-1500 listed under General References -- Babcock & Wilcox reports in section 5.

(b)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.

Table A-9. Davis-Besse 1 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	0.22	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[a]	0.088	1.69	0.014	0.013	0.41	0.63	0.15	0.40	0.21	-20	2	81	81.0	NA

Material ID	Heat No.	Spec No.	Supplier	Heat treatment[b]: Austenitizing	Tempering	Stress relief
Base metal A	5P4086	SA 508 Cl.2	LADISH	1640F ± 10F held at color 4 h, cold water quenched Reaustenitized 1590F ± 10F held at color 4 h, cold water quenched	1240F ± 10F held at color 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 6 h, then air cooled	Same as above
Weld metal	WF-182-1	NA	NA	NA	NA	1100-1150F for 15.5 h, furnace-cooled

(a)See BAW-1500 listed under General References -- Babcock & Wilcox reports in section 5.

(b)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.

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Table A-10. Midland 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	0.20	0.66	0.008	0.011	0.19	0.74	0.42	0.60	0.02	+20	+20	140	86.3	60.5
Weld metal A[(a)]	0.09	1.62	0.018	0.011	0.59	0.59	0.10	0.40	0.35	+20	+20	76	87.5	72.5
Weld metal B[(b)]	0.091	1.63	0.018	0.009	0.54	0.59	0.11	0.40	0.42	--	[+20]	[66]	85.5	69.0

Material ID	Heat No.	Spec No.	Supplier	Heat treatment[(c)]		
				Austenitizing	Tempering	Stress relief
Base metal A	ABZ196	SA 508 Cl. 2	Ladish	1640 ± 10F - 4 hrs, water quenched; 1590 ± 10F - 4 hrs, water quenched	1240 ± 10F - 10 hrs, water quenched	1125 ± 25F - 40 hrs, furnace-cooled below 600F.
Weld metal A	WF 209-1	NA	NA			1100-1150F - 22.5 h
Weld metal B	WF 70	NA	NA			1100-1150F - 48 h

(a) Chemistry not yet verified. Average values from similar weldments shown.

(b) See BAW-1500 listed under General References -- Babcock & Wilcox reports in section 5.

(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.

[ ] Estimated per BAW-10046A.

Table A-11. Midland 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_v-USE, ft-lb	UTS, ksi	YS, ksi
Base metal A	0.18	0.61	0.016	0.010	0.25	0.75	0.32	0.52	0.03	+10	+10	96	90.6	62.4
Weld metal A	0.12	1.20	0.004	0.016	0.32	0.46	0.09	0.30	0.03	-30	-30	77	80.5	63.0

				Heat treatment[a]		
Material ID	Heat No.	Spec No.	Supplier	Austenitizing	Tempering	Stress relief
Base metal A	BZB243	SA 508 Cl.2	Ladish	1640F ±10 - 3-1/2 h, water quench 1590F ± 10 - 3-1/2 h, water quench	1250F - 6 h, water quench	1125F - 40 h, furnace-cooled below 600F
Weld metal A	WF 336					1100-1150 - 16.5 h, furnace-cooled below 600F

[a] 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.

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APPENDIX B

Description and Properties of
Research Capsule Program Materials

Table B-1. Chemical Composition and Unirradiated Mechanical Properties of Research Capsule Surveillance Weld Metals

Ident No.	Chemical composition, %									Impact properties				
	C	Mn	P	S	Si	Ni	Cr	Mo	Cu	T_{NDT}, F	RT_{NDT}, F	C_v-USE, ft-lb	UTS, ksi	YS, ksi
Weld metal-W1	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
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	-50	-8	78	81.0	NA
Weld metal-W4	0.09	1.53	0.013	0.017	0.53	0.70	0.08	0.42	0.37	-40	-20	74	88.0	NA
Weld metal-W5	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
Weld metal-W6	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
Weld metal-W8	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
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 Research Capsule 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	48 h at 1100-1150F
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	48 h at 1100-1150F
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 toughness specimens included in the research capsule portion of this program.

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 Toughness Specimens

There are two configurations of compact fracture toughness 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 Toughness Specimens

The rectangular compact fracture toughness specimens are modifications of those in ASTM E399 and E813. 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. Four sizes of this type of specimen are included. The specimen sizes (in terms of thickness) are 0.394, 0.500, 0.936, and 2.00 inches. The dimensions of these specimens are listed in Table C-1.

3.2. Round Compact Fracture Toughness 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 toughness specimen with its corresponding dimensions.

3.3. Side-Grooved Specimens

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

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

	Dimensions, in.					
Specimen ID	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
0.500 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-5.

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

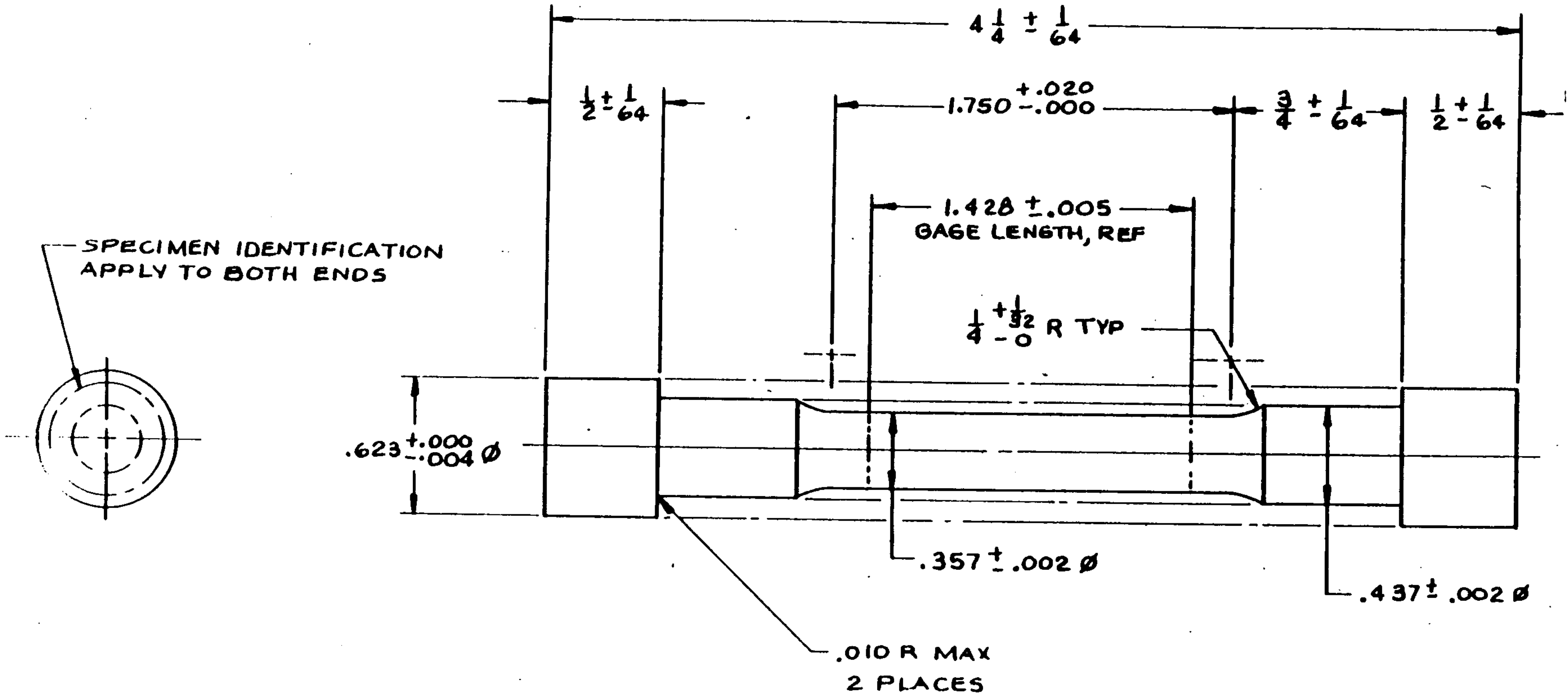

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Figure C-2. Miniature Size Tensile Specimens — Used on Type B Capsules

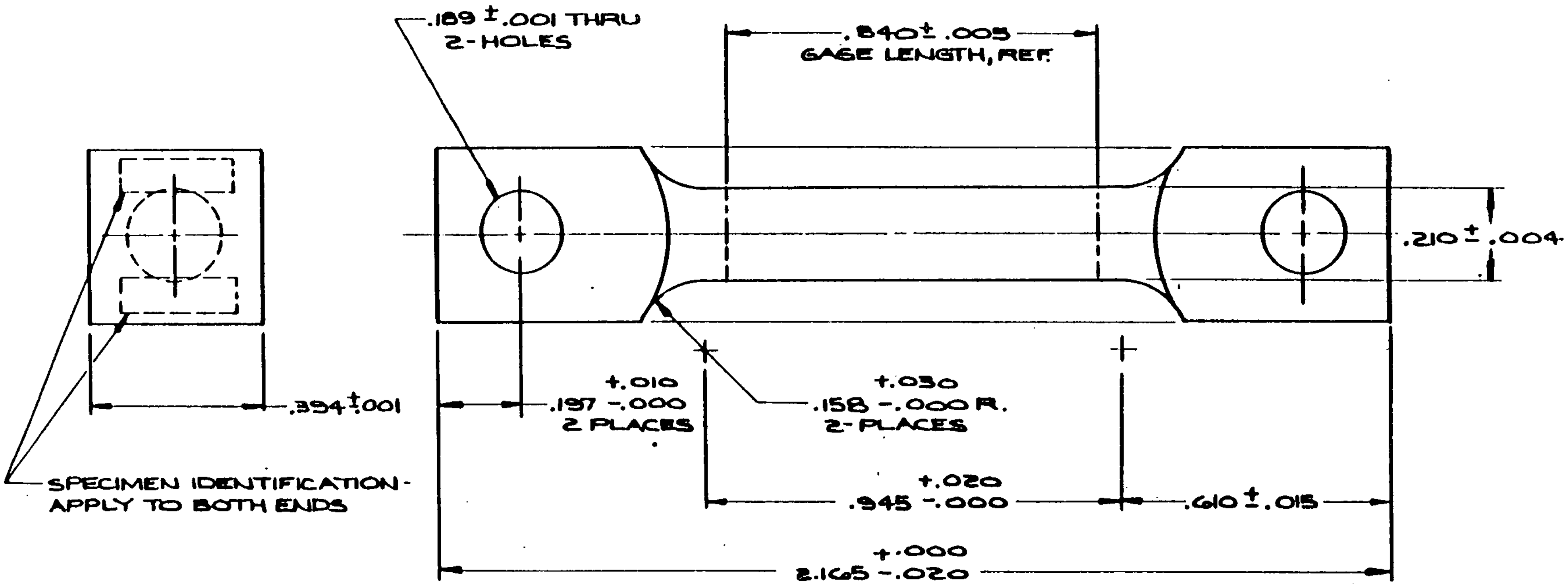

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Figure C-3. Charpy V-Notch Specimen

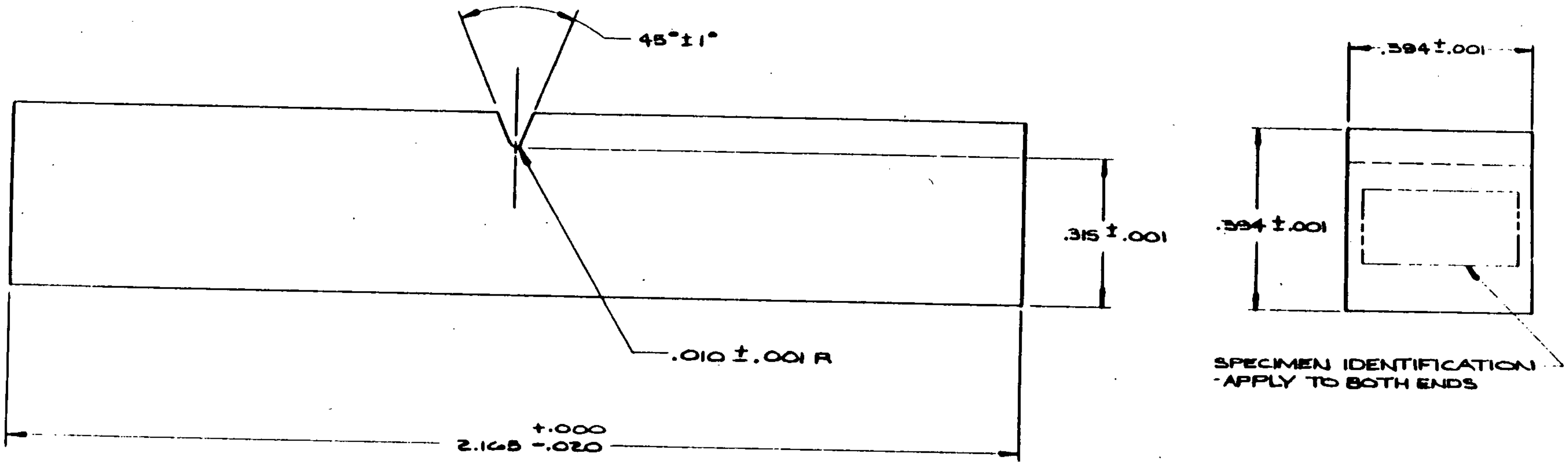

Figure C-4. Rectangular Compact Fracture Toughness Specimen -- Standard Proportions and Modifications for Measurement of Displacement at Load Line

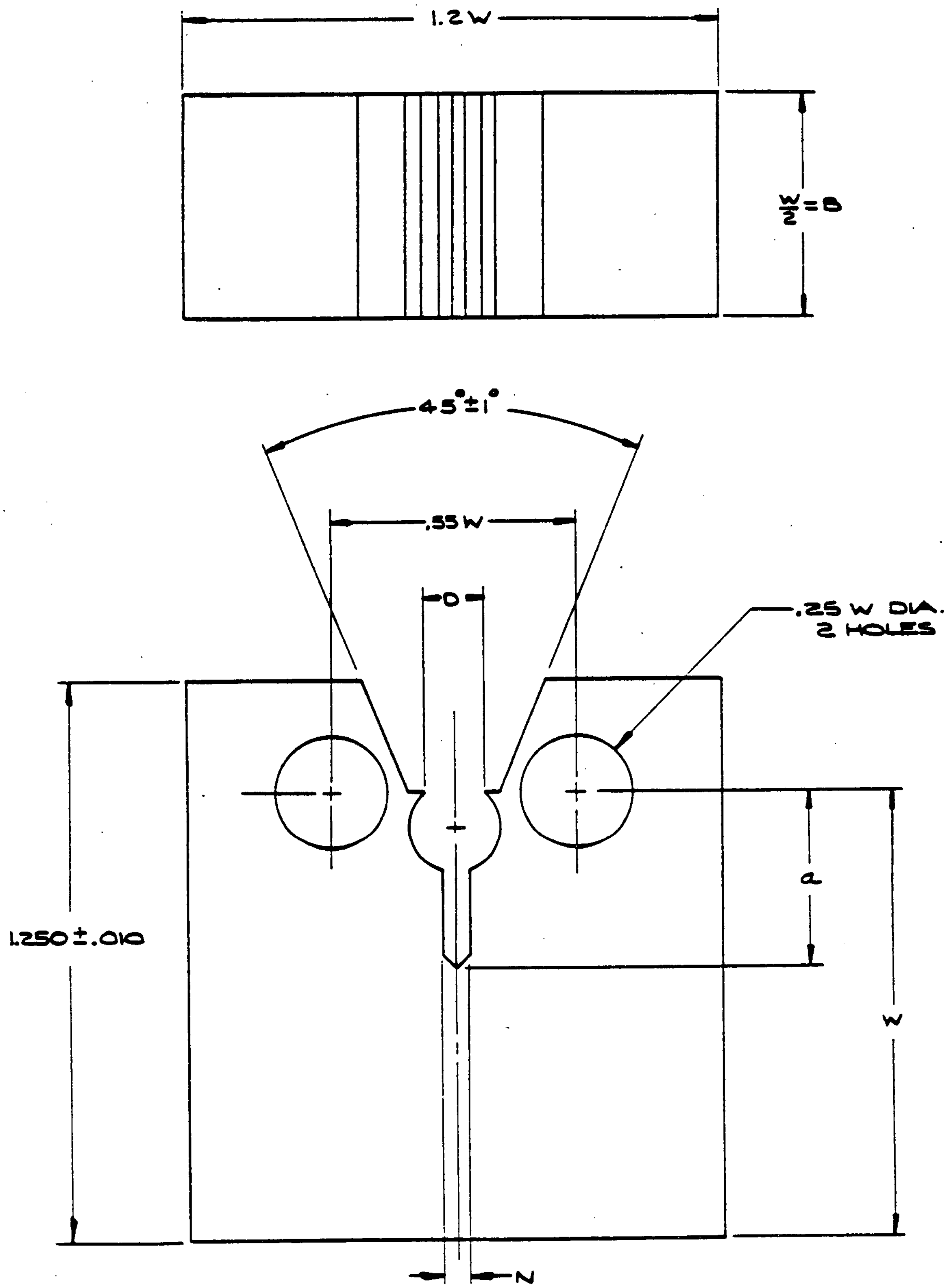

Figure C-5. Round Compact Fracture Toughness Specimen -- Dimensions and Modification for Measurement of Displacement at Load Line

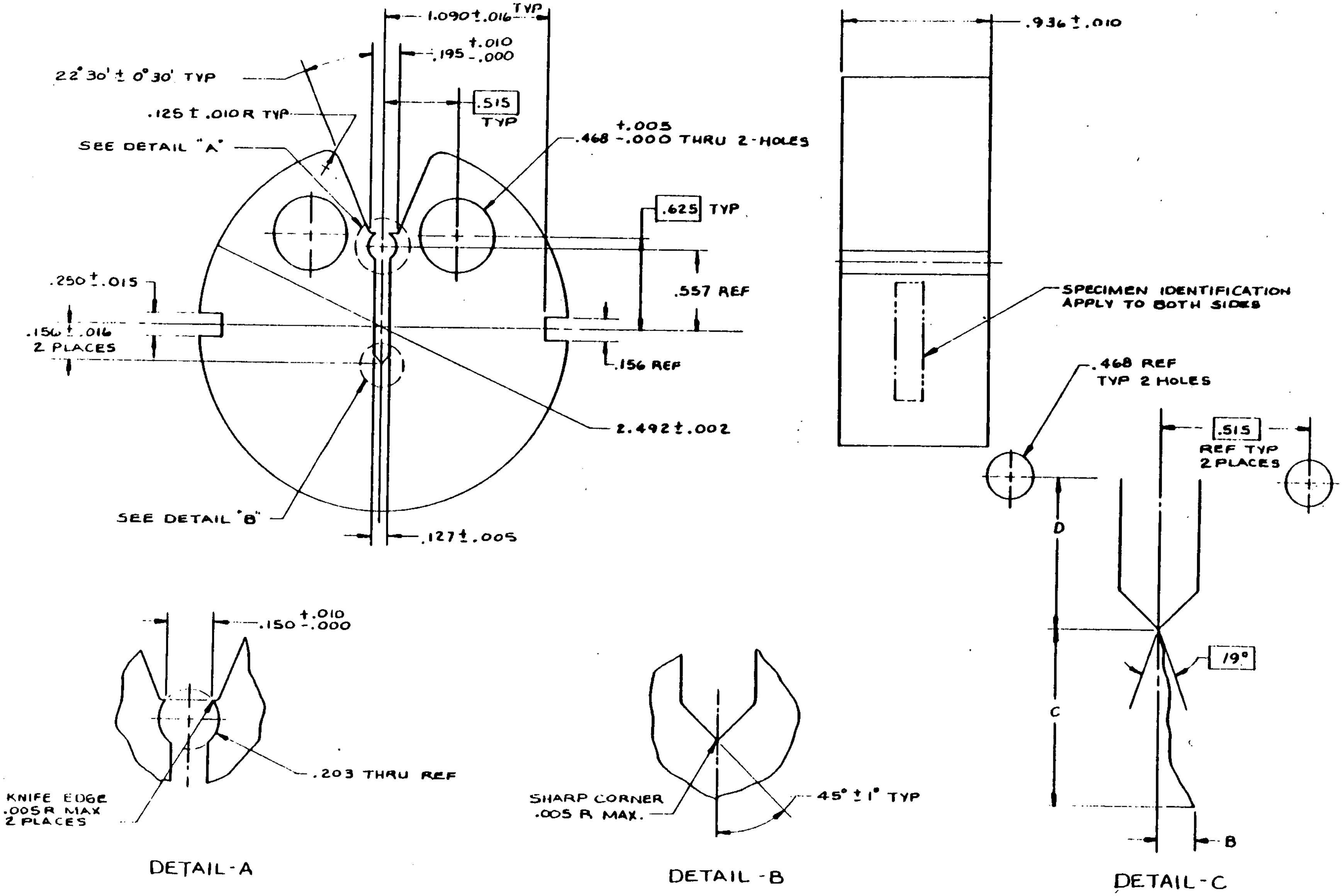

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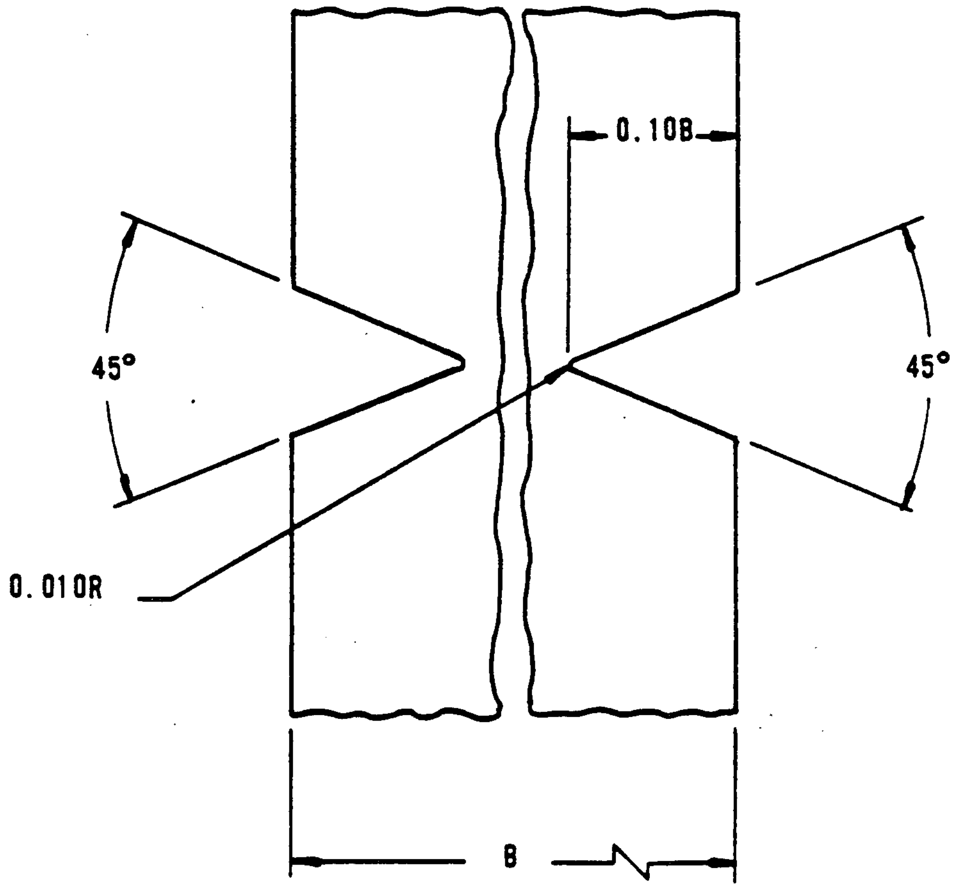
Figure C-6. Geometry of Side Grooves for 0.936 TRCT



APPENDIX D

Program and Capsule Type Designations

Table D-1. RVSP Capsule Types

Material description	No. of specimens: Tensile	No. of specimens: 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	No. of specimens		
	Tensile	Charpy	0.5 TCT
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-1. (Cont'd)

Material description	No. of specimens: Tensile	Charpy	0.394 TCT	0.5 TCT	0.936 TRCT
Capsule Type VII					
Weld metal A	6	18	6	12	9
Weld-HAZ, heat A, transverse	0	18	0	0	0
Base metal, heat A, transverse	6	18	0	0	0
Total per capsule	12	54	6	12	9
Capsule Type VIII					
Weld metal A	3	15	3	6	4
Weld metal B	3	15	3	6	5
Weld-HAZ, heat A, transverse	0	12	0	0	0
Base metal, heat A, transverse	6	12	0	0	0
Total per capsule	12	54	6	12	9
Capsule Type IX					
Weld metal A	6	18	0	0	0
Weld-HAZ, heat A, transverse	0	18	0	0	0
Base metal, heat A, transverse	6	18	6	12	9
Total per capsule	12	54	6	12	9

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

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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 209-1	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	No. of specimens Charpy
Capsules OCIII-A, -C, -E		
Weld metal, WF 209-1	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 209-1		
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	No. of specimens Charpy
Capsules TMI1-A, C, E		
Weld metal, WF 25	4	8
HAZ		
Heat C2789-2, longitudinal	0	8
Baseline material, plate		
Heat C2789-2, longitudinal	4	8
transverse	0	4
Correlation, HSST plate 02	0	8
Total per capsule	8	36
Capsules TMI1-B, D, F		
HAZ		
Heat C3307-1, longitudinal	4	10
Baseline material, plate		
Heat C3307-1, 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 TMI2-A, 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 TMI2-B, 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

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

Material description	No. of specimens		
	Tensile	Charpy	0.5 TCT
Capsules CR3-A, -C, -E			
Weld metal, WF 209-1	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 209-1	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 AN1-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 AN1-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

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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 RS1-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 RS1-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

Material description	Tensile	Charpy	0.5 TCT
	No. of specimens		
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

Table D-11. Materials and Specimens in Surveillance Capsules of Midland Unit 1

Materials description	Tensile	Charpy	0.394 TCT	0.5 TCT	0.936 TRCT
	No. of specimens				
Capsules MD1-A, B					
Weld metal, WF 209-1	6	18	6	12	9
Weld-HAZ Heat 532598, transverse	0	18	0	0	0
Base metal Heat 532598, transverse	6	18	0	0	0
Total per capsule	12	54	6	12	9
Capsule MD1-C					
Weld metal, WF 209-1	3	15	3	6	4
Weld metal, WF 70	3	15	3	6	5
Weld-HAZ Heat 532598, transverse	0	12	0	0	0
Base metal Heat 532598, transverse	6	12	0	0	0
Total per capsule	12	54	6	12	9
Capsules MD1-D, E, F					
Weld metal, WF 209-1	6	18	0	0	0
Weld-HAZ Heat 532598, transverse	0	18	0	0	0
Base metal Heat 532598, transverse	6	18	6	12	9
Total per capsule	12	54	6	12	9

Table D-12. Materials and Specimens in Surveillance Capsules of Midland Unit 2

Materials description	Tensile	Charpy	0.394 TCT	0.5 TCT	0.936 TRCT
	No. of specimens				
Capsules MD2-A, B					
Weld metal, WF 336	6	18	6	12	9
Weld-HAZ Heat BZB243, transverse	0	18	0	0	0
Base metal Heat BZB243, transverse	6	18	0	0	0
Total per capsule	12	54	6	12	9
Capsules MD2-C, D					
Weld metal, WF 336	6	18	0	0	0
Weld-HAZ Heat BZB243, transverse	0	18	0	0	0
Base metal Heat BZB243, transverse	6	18	6	12	9
Total per capsule	12	54	6	12	9

APPENDIX F
REGULATORY REQUIREMENTS

10CFR50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements," provides criteria for integrated surveillance programs. Paragraph II.C 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 personnel 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 the B&W Owners Group program. Each of the above criteria and considerations are discussed below.

A. All of the reactors in the program are of the Babcock & Wilcox 177 FA design, having similar design and operating features.

B. The B&W Owners Group is engaged in a program to provide for all the 177 FA reactors with cavity dosimeters. After an initial development and benchmarking effort, the reactor will be modified to accept removable (and replaceable) dosimeters.

C. All of the irradiation capsules originally prepared for all of the reactors in the program are scheduled to be irradiated (with the exception of those rendered questionable by being in the Three Mile Island-2 reactor at the time of the accident). However, as stated in this report, it is planned that some of those capsules will not be tested, it being our judgement that they will not provide enough relevant information to be worth the effort. It should be noted, though, that six additional capsules are being irradiated (two of which are in TMI-2) that contain test material of the greatest pertinence, improving the quality of the program beyond that of the original capsules.

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

E. Although, as stated in (A) above, the B&W 177 FA reactors are of similar design, their power levels are different in a number of instances. This is accounted for by positioning the capsules at various lead factors and analytically determining the effect of the reactor's power level upon vessel irradiation. Benchmarking experiments have confirmed the validity of this approach.

F. Through the B&W Owners Group Materials Committee, to which all the 177 FA plant owners subscribe, all RVSP reports and relevant information aredistributed to all plant owners. The Owners representatives meet regularly to discuss this program and monitor progress of development and benchmarking efforts.

G. Operation of the host reactor at reduced power level or their experienceing 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.

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.

APPENDIX G

ESTIMATED REFERENCE TEMPERATURE INFORMATION

The estimated reference temperature information is a vital part of the plant licensing process, since it is an input to the reactor pressure-temperature limitation calculations. All specific information and calculational results are presented on Tables G1-G8. The bases for the tabular information are explained below.

Material Identification - All weld numbers and locations were verified by reviewing B&W Mt. Vernon QA records. The materials included conform to the beltline definition of 10CFR50, Appendix G.[1]

Chemical Composition - Weld metal compositions were otained from (in order of priority) BAW-1500,[2] Table 10, BAW-1500, Table 6, or the weld qualification test reports. The atypical weld composition was obtained from BAW-10144A.[3]

Inside Surface Neutron Fluence - The Peak IS fluences are listed below; some are drawn from capsule reports, others are estimates. All are projections, based on fluence analyses of the maximum inside surface vessel fluence at EOL.

Plant	Peak IS fluence, n/cm^2 (E>1MEV)
Oconee 1	1.2E19
Oconee 2	1.2E19
Oconee 3	1.3E19
TMI-1	1.7E19
Crystal River 3	1.3E19
ANO-1	1.1E19
Rancho Seco 1	1.29E19
Davis-Besse 1	1.6E19

The fluences shown for the various welds on Tables G1-G8 are obtained by multiplying the peak IS fluence by the spatial fluence factors given in Table F-1 of B&W Document 51-1142595-00 (BAW-1485).[4] The sole exception is the Davis-Besse reactor vessel, which is not covered by that table. Davis-Besse is similar to Oconee 2 and 3, so the factors of Oconee 2 were used for Davis-Besse 1, with the exception of the nozzle belt to upper shell weld; this weld is at a higher elevation in Davis-Besse than in Oconee 2. Examination of construction drawings shows it on a level with

the nozzle belt to intermediate shell weld of Oconee 1. Therefore, the nozzle belt to intermediate shell weld fluence factors of Oconee 1 were used for the Davs-Besse 1 nozzle belt to upper shell weld.

Initial RT_{NDT} - These values were obtained from BAW-10046P[5] when no specific data were available, and are shown in parentheses. All other values were obtained from test reports.

Estimated Adjusted RT_{NDT} - The values shown for most weld metal were calculated according to Reg. Guide 1.99, Rev. 1.[6] The fluence for 1/4T locations was obtained by dividing the inside surface neutron fluence by 1.8, as described in Table 5-2 of Reference 4. All calculations were done by a TI59 programmable calculator. When the shift between initial and adjusted RT_{NDT} is less than 50°F, the calculation is not considered valid; however, the result is shown in these cases as an approximation.

The values for the atypical weld were calculated according to an NRC recommendation, which is presented in BAW-10144A. The calculation can be stated as follows:

$$\text{Adjusted } RT_{NDT} = \text{Initial } RT_{NDT} + 105.53(f/10^{19})^{1/2}$$

where f = fluence

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Table G-1. Material Properties and Estimated Reference Temperatures for Oconee Unit 1 Reactor Vessel Beltline Materials - 32 EFPY

Material Identification			Chemical Composition, wt %			Inside Surface Neutron Fluence	Initial RT_{NDT}	Estimated Adjusted RT_{NDT}	
								Inside surface	1/4T
Heat/Weld Number	Type	Location	Cu	P	Ni	n/cm^2	°F	°F	°F
SA1135	Weld	LNB/IS	0.25	0.011	0.54	1.9E18	(+20)	118	93
SA1229	Weld	IS/US (Inside 61%)	0.26	0.021	0.61	9.1E18	(+20)	292	223
WF25	Weld	IS/US (Outside 39%)	0.35	0.015	0.68	--	(+20)	--	--
SA1585	Weld	US/LS	0.21	0.016	0.59	1.2E19	(+20)	250	191
WF9	Weld	LS/Dutchman	0.21	0.016	0.57	6.7E16	(+20)	37	33
SA1073	Weld	IS-Longitudinal (both)	0.29	0.025	0.64	7.2E18	(+20)	286	229
SA1493	Weld	US-Longitudinal (both)	0.29	0.017	0.55	8.8E18	(+20)	296	226
SA1430	Weld	LS-Longitudinal (both)	0.29	0.017	0.55	1.1E19	(+20)	308	251

Table G-2. Material Properties and Estimated Reference Temperatures for Oconee Unit 2 Reactor Vessel Beltline Materials - 32 EFPY

Material Identification			Chemical Composition, wt %			Inside Surface Neutron Fluence	Initial RT_{NDT}	Estimated Adjusted RT_{NDT}	
								Inside surface	1/4T
Heat/Weld Number	Type	Location	Cu	P	Ni	n/cm^2	°F	°F	°F
WF154	Weld	LNB/US	0.31	0.013	0.59	9.1E18	(+20)	298	230
WF25	Weld	US/LS	0.35	0.015	0.68	1.2E19	-14[a]	279	248
WF112	Weld	LS/Dutchman	0.31	0.016	0.59	6.7E16	0	25	19

(a) - Based on TMI-1 surveillance data

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Table G-3. Material Properties and Estimated Reference Temperatures for Oconee Unit 3 Reactor Vessel Beltline Materials - 32 EFPY

Material Identification			Chemical Composition, wt %			Inside Surface Neutron Fluence	Initial RT_{NDT}	Estimated Adjusted RT_{NDT}	
								Inside surface	1/4T
Heat/Weld Number	Type	Location	Cu	P	Ni	n/cm^2	°F	°F	°F
WF200	Weld	LNB/US	0.24	0.010	0.63	9.9E18	(+20)	229	176
WF67	Weld	US/LS (Inside 75%)	0.24	0.021	0.60	1.3E19	(+20)	318	245
WF70	Weld	US/LS (Outside 25%)	0.35	0.018	0.59	--	(+20)	--	--
WF169	Weld	LS/Dutchman	0.18	0.016	0.63	7.3E16	(+20)	35	31

Table G-4. Material Properties and Estimated Reference Temperatures for TMI Unit 1 Reactor Vessel Beltline Materials - 32 EFPY

Material Identification			Chemical Composition, wt %			Inside Surface Neutron Fluence	Initial RT_{NDT}	Estimated Adjusted RT_{NDT}	
								Inside surface	1/4T
Heat/Weld Number	Type	Location	Cu	P	Ni	n/cm^2	°F	°F	°F
WF70	Weld	LNB/US	0.35	0.018	0.59	1.3E19	(+20)	318	286
WF25	WEld	US/LS	0.35	0.015	0.68	1.7E19	-14[a]	300	266
WF70	WEld	LS/Dutchman (Outside 50%)	0.35	0.018	0.59	--	(+20)	--	--
WF67	Weld	LS/Dutchman (Inside 50%)	0.24	0.021	0.60	9.5E16	(+20)	46	39
WF8	Weld	US-Longitudinal (both)	0.29	0.010	0.55	1.7E19	(+20)	334	273
SA1494	Weld	LS-Longitudinal (OD 63%)	0.18	0.015	0.63	--	(+20)	--	--
SA1526	Weld	LS-Longitudinal (ID 37%)	0.35	0.013	0.68	1.4E19	(+20)	322	290
Atypical	Weld	US/LS	0.39	0.021	0.10	1.7E19	+90	228	193

(a) - Based on TMI-1 surveillance data

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Table G-5. Material Properties and Estimated Reference Temperatures for Crystal River Unit 1 Reactor Vessel Beltline Materials - 32 EFPY

Material Identification			Chemical Composition, wt %			Inside Surface Neutron Fluence	Initial RT_{NDT}	Estimated Adjusted RT_{NDT}	
								Inside surface	1/4T
Heat/Weld Number	Type	Location	Cu	P	Ni	n/cm^2	°F	°F	°F
WF169	Weld	LNB/US (Outside 60%)	0.18	0.016	0.63	--	(+20)	--	--
SA1769	Weld	LNB/US (Inside 40%)	0.26	0.020	0.61	9.9E18	(+20)	299	228
WF70	Weld	US/LS	0.35	0.018	0.59	1.3E19	(+20)	318	286
WF154	Weld	LS/Dutchman	0.31	0.013	0.59	7.3E16	(+20)	45	39
WF8	Weld	US-Longitudinal	0.29	0.010	0.55	1.2E19	(+20)	305	232
WF18	Weld	US-Longitudinal	0.29	0.010	0.55	1.2E19	(+20)	305	232
SA1580	Weld	LS-Longitudinal (both)	0.29	0.015	0.55	1.1E19	(+20)	308	243
Atypical	Weld	US/LS	0.39	0.021	0.10	1.3E19	+90	210	180

Table G-6. Material Properties and Estimated Reference Temperatures for ANO Unit 1 Reactor Vessel Beltline Materials - 32 EFPY

	Material Identification		Chemical Composition, wt %			Inside Surface Neutron Fluence	Initial RT_{NDT}	Estimated Adjusted RT_{NDT} Inside surface	Estimated Adjusted RT_{NDT} 1/4T
Heat/Weld Number	Type	Location	Cu	P	Ni	n/cm^2	°F	°F	°F
WF182	Weld	NB/US	0.24	0.014	0.63	8.4E18	+5	216	162
WF112	Weld	US/LS	0.31	0.016	0.59	1.1E19	0	288	242
WF18	Weld	US and LS-Longitudinal	0.29	0.010	0.55	8.6E18	(+20)	261	200
SA1788	Weld	LS/Dutchman	0.25	0.017	0.54	6.2E16	(+20)	40	35

Table G-7. Material Properties and Estimated Reference Temperatures for Rancho Seco Unit 1 Reactor Vessel Beltline Materials - 32 EFPY

Material Identification			Chemical Composition, wt %			Inside Surface Neutron Fluence	Initial RT_{NDT}	Estimated Adjusted RT_{NDT}	
								Inside surface	1/4T
Heat/Weld Number	Type	Location	Cu	P	Ni	n/cm^2	°F	°F	°F
WF233	Weld	LNB/US	0.29	0.021	0.68	9.8E18	(+20)	302	252
WF154	Weld	US/LS	0.31	0.013	0.59	1.29E19	(+20)	317	270
WF233	Weld	LS/Dutchman	0.29	0.021	0.68	7.2E16	(+20)	47	40
WF29	Weld	US-Longitudinal (both)	0.23	0.015	0.63	1.15E19	(+20)	261	200
WF29	Weld	LS-Longtiduinal (100%)	0.23	0.015	0.63	1.28E19	(+20)	275	210
WF70	Weld	LS-Longitudinal (ID 73%)	0.35	0.018	0.59	1.28E19	(+20)	317	285
WF29	Weld	LS-Longitudinal (OD 27%)	0.23	0.015	0.63	--			
Atypical	Weld	LS-Longitudinal (ID 73%)	0.39	0.021	0.10	1.28E19	+90	209	179

Table G-8. Material Properties and Estimated Reference Temperatures for Davis-Besse Unit 1 Reactor Vessel Beltline Materials - 32 EFPY

Heat/Weld Number	Material Identification: Type	Material Identification: Location	Chemical Composition, wt %: Cu	P	Ni	Inside Surface Neutron Fluence, n/cm²	Initial RT_{NDT}, °F	Estimated Adjusted RT_{NDT}: Inside surface, °F	1/4T, °F
WF232	Weld	NB/US (Inside 9%)	0.14	0.011	0.69	2.6E18	(+20)	79	--
WF233	Weld	NB/US (Outside 91%)	0.29	0.021	0.68	--	(+20)	--	140
WF182	Weld	US/LS	0.24	0.014	0.63	1.6E19	+5	296	222
WF232	Weld	LS/Duchman (Inside 12%)	0.14	0.011	0.69	1.0E17	(+20)	32	--
WF233	Weld	LS/Dutchman (Outside 88%)	0.29	0.021	0.68	--	(+20)	--	43

APPENDIX G REFERENCES

1. Federal Regulation 10CFR Part 50, Fracture Toughness Requirements for Light-Water Nuclear Power Reactors. Appendix G, Fracture Toughness Requirements, Federal Register, Vol. 48, No. 104, May 27, 1983, pps. 24008-24011.

2. B&W Document BAW-1500, "Chemistry of 177-FA B&W Owners Group Reactor Vessel Beltline Welds."

3. B&W Topical Report BAW-10144, "Evaluation of the Atypical Weldment," February 1980.

4. B&W Document 51-1142595-00, "Pressure Vessel Fluence Analysis for 177-FA Reactors." (BAW-1485)

5. B&W Topical Report BAW-10046P, "Methods of Compliance With Fracture Toughness and Operational Requirements of 10CFR50, Appendix G," March 1976.

6. U. S. NRC Regulatory Guide 1.99, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials," (Rev. 1, April 1977).

APPENDIX H

REACTOR VESSEL NEUTRON DOSIMETRY INFORMATION

I. Description of the Dosimetry Portion of the RVSP

The two principal objectives of the Integrated RSVP are (1) to determine the change(s) 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 empirically determined from the measurement and subsequent analysis of in-capsule dosimetry. Mechanical testing is then used 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.

It is anticipated that the last surveillance capsule of the integrated RVSP will be removed by 1995 (with capsules being removed periodically over the next ten year period). At that time, the primary objectives of the integrated RVSP will be satisfied, but there will be no in-place method of fluence verification since the dosimeters are contained in the RVSP capsules.

Fluence determinations over the lifetime of the reactor vessels in the integrated RVSP are divided into three distinct time periods described as follows:

Initial period: Reactor startup to removal of last capsule in reactor vessel.

Intermediate period: Removal of last capsule from the original plant to installation of cavity dosimetry.

Final period: Installation of cavity dosimetry to end of plant life.

Only two of the eight operable 177FA B&W reactors are "host reactors," Davis-Besse 1 and Crystal River 3, and therefore are the only two which have a direct fluence measurement capability. All other operable B&W 177FA reactors are "guest reactors" and do not have a fluence verification capability per se.

The "guest reactors" are Oconee Units 1, 2, and 3, ANO-1, Rancho Seco 1, and TMI-1. The only reactors still in the initial period are the two host

reactors. All other reactors are in the intermediate period. The host reactor for the proposed in-out experiment (either CR-3 or D-B) will probably not have an intermediate period since cavity dosimetry will be installed as part of the cavity dosimetry benchmark experiment before all of the capsules are removed. The procedures used to determine vessel fluence for each of the three time periods are described below.

I. A. 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 (pin by pin) power distribution is calculated and is used as input to the DOT-4 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 the fact that many long half life isotopes have not reached 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 between 5 and 15% different from 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 is based on PDQ baffle flux comparisons.

The initial period fluences are considered to be very accurate, with uncertainties conservatively estimated to be in the 10-22% range. The projection to end-of-life fluence is more uncertain due to additional uncertainty associated with future fuel cycle designs, and ranges from 20 to 35%.

3.3.I.B. Intermediate Period (Capsule Removal to Cavity Dosimetry Installation)

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

The DOT code is used to calculate the reactor pressure vessel fluence 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 either the average normalization factor for all (Initial Period) capsule analyses or the normalization factor for the host reactor.

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

I.C. Final Period (Cavity Dosimetry Installation to End of Plant Life)

The vessel fluence will be calculated from a knowledge of the cavity flux using the semi-empirical method described below. The procedure is identical to the method described in Paragraph I.A. except for the inclusion of a correction factor which accounts for cavity streaming. This correction factor will be generated as part of the analyses of the results of the cavity dosimetry benchmark experiment (out-in) experiment.

The uncertainty associated with the final period fluence determination cannot be firmly established until the results of the Out-In Experiment have been evaluated.

II. SUMMARY

As discussed in Section I there are two principal objectives of the IRVSP: (1) determine fluence induced change(s) in the mechanical properties of RV steel and (2) monitor the vessel fluence.

The first objective will be satisfied as each reactor's RVSP ends (as discussed in Section I).

The dosimetry program described in Sections I.A., I.B., and I.C. provides a knowledge of the vessel fluence within a determinable uncertainty range for the past, present and future. This knowledge satisfies the second objective of the RVSP (continuous monitoring of vessel fluence).

The fulfillment of these two objectives demonstrates that the dosimetry program is adequate and that the B&W 177 FA reactors are in compliance with 10CFR50 Appendix H.