WESTINGHOUSE ADVANCED REACTORS DIVISION

CRBRP ENGINEERING STUDY REPORT

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Title: Adequacy of the CRBRP Reactor Vessel NDE Inspections

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SUMMARY

Differences between in-service inspection requirements for LMFBR and PWR vessels are the result of technical considerations associated with the materials of construction and operating conditions to which each system is exposed. The primary concern and reason for volumetric inspection in PWR vessels is brittle fracture. Similar abrupt type failure mechanisms do not exist in the stainless steel CRBRP vessel.

The CRBRP vessel welds have been thoroughly examined and meet all requirements of the ASME Code. Multiple angle, broad spectrum X-rays have been used extensively to provide high assurance that all flaws, including planar cracks have been detected. The bi-metallic welds, which are made using a high nickel content filler material and which sees only a low operating temperature, is not subjected to the conditions which have historically caused bi-metallic weld failure.

An assessment of current state of the art capability to ultrasonically (UT) examine austenitic stainless steel welds showed considerable progress has been made in overcoming problems encountered at the time of the FFTF fabrication. However, it was concluded that although interpretable information might be obtained from such examinations, application of UT to the CRBRP vessel is unlikely to make any significant contribution in finding additional flaws not identified by the required ASME Code non-destructive examination.

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TABLE OF CONTENTS

	Page
SUMMARY	i
TABLE OF CONTENTS	ii
1.0 INTRODUCTION	1-1
1.1 VESSEL DESCRIPTION	1-3
1.2 EXAMINATION RATIONALE	1-6
2.0 COMPARISON OF CRBRP AND PWR OPERATING CONDITIONS AND MATERIALS	2-1
2.1 DIFFERENCES IN CRBRP AND PWR OPERATING CONDITIONS	2-1
2.2 MATERIALS DIFFERENCES	2-3
3.0 NON-DESTRUCTIVE EXAMINATIONS (NDE) PERFORMED AND THEIR ADEQUACY	3-1
3.1 ASME CODE, SECTION III AND RDT STANDARD NDE REQUIREMENTS	3-1
3.1.1 REACTOR VESSEL SPECIFICATION REQUIREMENTS	3-1
3.1.2 NDE REQUIREMENTS FOR THE REACTOR VESSEL WELDS	3-1
3.1.3 NDE ACCEPTANCE CRITERIA FOR THE REACTOR VESSEL WELDS	3-3
3.2 DESCRIPTION OF APPLIED NDE METHODS	3-4
3.2.1 LIQUID PENETRANT	3-4
3.2.2 RADIOGRAPHY	3-5
3.3 NDE AND QUALITY ASSURANCE ACTIVITIES PERFORMED ON THE REACTOR VESSEL	3-7
3.3.1 REACTOR VESSEL NDE	3-7
3.3.2 QUALITY ASSURANCE	3-8
3.4 FACTORS INFLUENCING WELD STRENGTH AND BI-METALLIC WELD OPERATING EXPERIENCE	3-8
4.0 ASSESSMENT OF ULTRASONIC INSPECTION CAPABILITY OF THE CRBRP REACTOR VESSEL	4-1
4.1 DESCRIPTION OF ULTRASONIC METHODOLOGY	4-1
4.1.1 REVIEW OF ULTRASONIC EXAMINATION PRINCIPLES	4-1
4.1.2 EFFECTS OF MATERIALS ON ULTRASONIC INSPECTION	4-3
4.2 STATE OF THE ART ULTRASONIC TECHNOLOGY	4-4

TABLE OF CONTENTS (Continued)

	Page
4.3 TECHNICAL FEASIBILITY OF PERFORMING ULTRASONIC EXAMINATION ON THE REACTOR VESSEL	4-6
5.0 PHYSICAL CONSTRAINTS ON PERFORMING PRE-SERVICE ULTRASONIC EXAMINATIONS	5-1
5.1 FABRICATION STATUS AND ACCESSIBILITY	5-1
5.1.1 FABRICATION STATUS	5-1
5.1.2 ACCESSIBILITY FOR FURTHER INSPECTION	5-1
5.2 CONSEQUENCES OF FALSE INDICATIONS	5-2
6.0 CONCLUSIONS	6-1
7.0 REFERENCES	7-1

LIST OF TABLES

Table No.	Title	Page
2-1	LMFBR-PWR Comparison of Design Conditions	2-6
3-1	Reactor Vessel NDE Summary	3-12
3-2	Transition Joints in PWR Plants in Operation in Excess of Ten Years	3-14
3-3	Transition Joints in BWR Plants in Operation in Excess of Ten Years	3-15
3-4	Transition Joints in LMFBR Demonstration Plant Superheaters Operating in Excess of Ten Years	3-16

LIST OF FIGURES

Figure No.	Title	Page
1.1	Reactor Vessel	1-9
1.2	Reactor Vessel Weld Joint Configuration	1-10
3.1	Reactor Vessel Weld Development	3-17

1.0 INTRODUCTION

The purpose of this report is as follows:

A. To document the Non-Destructive Examinations (NDE) performed on the Clinch River Breeder Reactor Plant (CRBRP) reactor vessel (RV).

B. To provide rationale for differences between LMFBR and PWR inspection requirements.

C. To assess the adequacy of these examinations in assuring an acceptably low probability for undiscovered flaws which could result in vessel degradation during service.

D. To assess the technical feasibility and net benefit accrued from performing a pre-service ultrasonic examination of the reactor vessel welds.

E. To identify physical constraints which influence performance of additional pre-service examinations.

Additionally, although the focus of the report is the reactor vessel, it is anticipated that because of the similarity of design requirements, materials, configuration, NDE methodology and acceptance criteria, conclusions reached can be shown to be applicable to other vessels in the Primary Heat Transport System (PHTS), Intermediate Heat Transport System (IHTS), and the Reactor Residual Heat Extraction System (RRHES).

Non-Destructive Examinations which are performed on reactor components have one objective: to assess the structural integrity of the component for confirmation of its ability to perform as intended. Generally, examinations are performed throughout the fabrication process, before introduction into service and then periodically throughout the service life. Since the objective of NDE can be achieved in many ways, component physical characteristics, operating environment, and examination techniques available

often dictate the approach used. Each of these elements has had a strong influence on the NDE already performed on the reactor vessel and in selecting the examinations that will be performed throughout its life.

Defects which are found in welds have a wide range of configurations which are directly related to the initiating cause. Most often they are associated with departure from the intended weld process parameters such as heat input, cover gas purity, and deposition speed. They can also be the result of operator techniques, including incomplete removal of slag between passes, hot cracks, fissures, moisture and preheat. Commonly found defects include entrapped gas (porosity), slag inclusion (improper inter-pass cleaning) and lack of fusion at the weld metal/base metal junction or interpass regions. Defects therefore can take the form of small spherical pockets, long thin stringers, or planar type cracks. Examination methods most often used to detect flaws that penetrate the surface are liquid penetrant and magnetic particle. Because LMFBR vessels are primarily constructed from austenitic stainless steel (non-magnetic), surface inspection is usually liquid penetrant. However, proper selection of the type of liquid penetrant (based on material and weld geometry considerations) provides a high degree of assurance that all surface flaws will be detected. The liquid penetrant methodology used for the reactor vessel is discussed further in Section 3.

For flaw detection within the material, two primary methods are used: radiography and ultrasonic examination. Radiography utilizes the principle that the ability of radiation, usually gamma or X-ray, to penetrate a material is dependent on the mean free path of the radiation, i.e., the average distance traveled between interactions within the material. The mean free path is density dependent and thus increases if the radiation passes through a flaw in the material. Larger amounts of radiation penetrate flawed areas and thus show up on film as darker spots or in the case of tungsten inclusions, light spots, revealing the presence of a flaw. Clearly, different types of flaws have varying effects on the ability of radiation to penetrate the material. Flaws aligned with the direction of the radiation produce large increases in the mean free path and very thin (planar) flaws aligned perpendicular to the direction of radiation produce small increases in the mean free path. This latter effect necessitates proper alignment of the

source or multiple exposures at different angles to assure that planar type defects are detected. Multiple angle radiography was the means employed in examination of the reactor vessel to assure detection of planar flaws. Radiographic methods, calibration techniques and detectable flaw sizes are discussed in Section 3.

Ultrasonic examination techniques use high frequency sound waves to examine the material for flaws. Reflection of the sound wave indicates a discontinuity in the material. Since reflection of the sound can be the result of grain structure and other inherent material characteristics, interpretation of the results of ultrasonic examination is usually more subjective than interpretation of radiographic films. Ultrasonic examination is most sensitive to planar flaws perpendicular to the motion of the sound wave. Radiography is most sensitive to flaws parallel to the path of the radiation. Thus the two examination methods complement each other and where practical both may be used. Ultrasonic examination methodology is discussed further in Section 4.

The examination rationale for the CRBRP reactor vessel will be reviewed after a brief description of the vessel.

1.1 Vessel Description

The CRBRP reactor vessel was fabricated and is currently in storage at an inactive Babcock and Wilcox facility in Mt. Vernon, Indiana. An isometric cutaway view of the vessel is shown in Figure 1.1. The vessel is approximately 59 feet high x 20 feet in diameter. The upper portion is fabricated of SA-508 carbon steel to match the thermal expansion characteristics of the closure head. The support ring (SA-508) is welded to the top flange (SA-508). Both of these elements are fabricated from ring forgings. At a location approximately 4 feet below the bottom of the support ring, a 14 inch Inconel 600[1] (SB-168) shell is welded to the SA-508 support ring. This transition shell serves two purposes:

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[1] Inconel is a registered tradename of Huntingdon Alloys.

A. Because the coefficient of thermal expansion for Inconel 600 is approximately halfway between that of SA-508 and 304 SS, thermal stresses due to differential expansion between the flange and stainless steel vessel shell are reduced.

B. Lower thermal conductivity of the Inconel 600 (relative to the SA-508) reduces the amount of heat conducted into the flange area.

The lower transition joint (Inconel 600 - 304 SS) is approximately 2 feet above the sodium level and is thus protected from rapid changes in temperature in the outlet plenum sodium. The upper transition joint (SA-508 - Inconel 600) operates at a temperature of approximately 450°F. This joint is maintained at a relatively constant temperature since it is under the influence of the closure head and vessel flange heating system which controls the temperature of those components to within a design temperature range of 400 $\pm$ 50°F. The lower transition joint (Inconel 600 - 304 SS) has an operating temperature of approximately 620°F and is well below 800°F where high temperature design considerations must be addressed. Weld filler material used in both upper and lower transition joints is Inconel 82 (SFA - 5.14), a high nickel alloy which is the industry standard for joints of this type, and for which an extensive data base exists. Below the Inconel 600 transition joint, the remainder of the vessel pressure boundary is fabricated of 304 SS joined by 308 SS filler material.

Reactor vessel shell courses were fabricated from plate which had a 20° included angle, V-Bevel joint configuration for the longitudinal joints between plates and a ship lap with a J-Prep joint for joining shell courses. (See Figure 1.2) The snip lap joint positions the shells and provides an accurate fitup.

The shell courses were each fabricated from three plates. The plates, in general, used the 20° V-Bevel joint with a 3/4 inch nominal root gap at the edge of the weld preps shown on Figure 1-2. The gap was closed off by a backing bar for welding. The backing bar was removed after welding and the root area was back-grooved. The back-grooved area was then welded and ground.

The shell courses were joined with a ship lap type joint as shown on Figure 1.2. The joint positioned the shells and provided an accurate fitup. The J-Prep portion of the joint was welded and then the ship lap and root areas were removed by machining or arc air gouging followed by grinding. The weld was then completed from the opposite side.

Since potential defects such as slag inclusions, porosity, and lack of fusion have historically been found to be more likely in the root pass, removal of the root pass in each of the above welding operations considerably enhances assurance of a defect-free weld.

The design temperature for the upper stainless steel shells of the reactor vessel is 900°F. The actual maximum operating temperature is approximately 25°F to 30°F below the design temperature and thus assures that all portions of the vessel are effectively outside the regime where creep effects are significant. The ability to maintain low vessel wall temperatures is provided by inclusion of a 316 SS liner in the vessel which separates the hot outlet plenum sodium from the vessel wall. The annulus between the liner and vessel wall is cooled with vessel inlet sodium which is channeled into the annulus below the outlet plenum. Two rings (fabricated from forged bars) provide support for the vessel liner and core support structure and are welded into the vessel shell at locations approximately 30 and 44 feet below the top flange. The temperature of these forgings is controlled by the cool inlet sodium.

Three 36 inch outlet nozzles and three 24 inch inlet nozzles are located approximately 24 and 49 feet below the top flange respectively. The reactor vessel also has cover gas inlet and outlet, sodium makeup and overflow nozzles located in the upper shell course.

An additional feature provided in the CRBRP design to assure maintenance of safe levels of sodium at all times, is a guard vessel which surrounds the vessel and inlet/outlet piping. The annulus between the vessel and guard vessel is sized in conjunction with the outlet plenum volume to assure safe levels of sodium in the event an unanticipated leak occurs in the vessel or

piping. The annulus also provides access for visual examination of the vessel and guard vessel to satisfy inspection requirements of the ASME Code Section XI, Division 3.

1.2 Examination Rationale

A primary design objective for the CRBRP is the assurance of the public health and safety for all potential operating conditions. To accomplish this objective, a two pronged approach has been taken.

A. Design

Development of a design that is inherently tolerant to a broad range of conditions that could be imposed over the plant operational lifetime.

B. Fabrication and Operation

A comprehensive program for assuring that when the components are put into service they will be free of flaws that affect their ability to perform as designed and an in-service monitoring and inspection program to identify component degradation that could lead to leakage into the guard vessel.

Design considerations which utilize diversity and redundancy features, such as guard vessels, are well documented in the PSAR and will not be discussed further in this report.

Before further discussion of the vessel NDE rationale, it is important to define three terms that will be used throughout this report: 1) fabrication inspection, 2) pre-service inspection, and 3) in-service inspection. Fabrication inspection has the purpose of assuring that the product put into service meets all drawing requirements and is free of flaws (to a level specified by the appropriate codes and standards) that could degrade performance or lead to in-service performance degradation. Pre-service

inspection is performed to provide an initial description of some characteristic of a component, related to its integrity, which will be monitored throughout life (for changes from the baseline). It is possible that in the process of performing fabrication inspections, a baseline will be established for comparison with data planned to be obtained during service. In that case, the inspection will be both 'fabrication' and 'pre-service'. However, this dual role exists only if the fabrication inspection data taken is also planned for comparison with in-service inspection data.

In-service inspection relates to inspections performed after a component has been used for some period of time in its intended function. Data from in-service inspection is compared with the pre-service baseline to identify changes which could degrade the components ability to function as designed. As an example, since there are no plans to perform in-service radiography or liquid penetrant examination of the vessel welds, these examinations are classified as 'fabrication inspections'.

The reactor vessel is designed as a Class 1 component as defined by Section III and appropriate Code Cases of the ASME Code. It meets all fabrication requirements set forth by those documents. Section XI, Division 3 of the ASME Code sets forth requirements for the pre-service and in-service inspection of LMFBR vessels. With the exception of the Inconel 600 transition joint welds, Section XI, Table IMB-2500-1 (Item B1.10) requires visual, VTM-2[1], examination of the reactor vessel welds during service. Table IMB-2500-1 (Item B1.11) also requires continuous monitoring by leak detection. The reactor guard vessel provides tubing for continuous leak detection monitoring. A 'pre-service' visual examination of these welds (meeting the requirements of VTM-2) will be performed at site prior to placement of the vessel into service. No other 'pre-service' examination of these welds is

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(1) Visual Examination, VTM-2. a) The VTM-2 visual examination shall be conducted on exterior surfaces in such a way that accumulation of liquids, liquid streams, liquid drops, and smoke are discernable. b) The VTM-2 visual examination may be direct or remote using aids. c) The removal of external covering, such as insulation, is not required for VTM-2 visual examinations.

currently planned, nor are any required by the ASME Code. A discussion of the 'fabrication inspections' performed on these welds (which were in excess of the code requirements) and an assessment of their adequacy is presented in other sections of this report.

Table IMB-2500-1 also addresses 'Dissimilar Metal Welds'. A footnote to the table clarifies the word 'Dissimilar' as follows: 'includes welds joining carbon or low-alloy steel to high alloy or high nickel steels'. From this definition, an interpretation could be made that the ASME Code Section XI, only considers the top transition weld to be a 'Dissimilar Metal Weld' and thus the bottom weld would only require a VTM-2 examination. Dissimilar metal welds are required to have volumetric in-service inspection per Table IMB-2500-1 (Item 5.10). However, the CRBRP project has taken the position that in-service volumetric inspection of both top and bottom transition welds is unnecessary and thus the distinction between categorizations becomes irrelevant.

The exception taken by the project is based on considerations of joint design, loading conditions and low joint temperature operation (450°F) in a non-sodium environment. Details supporting this position are provided in Appendix G of the PSAR and are also reviewed in Section 3.4 of this report.

REACTOR VESSEL

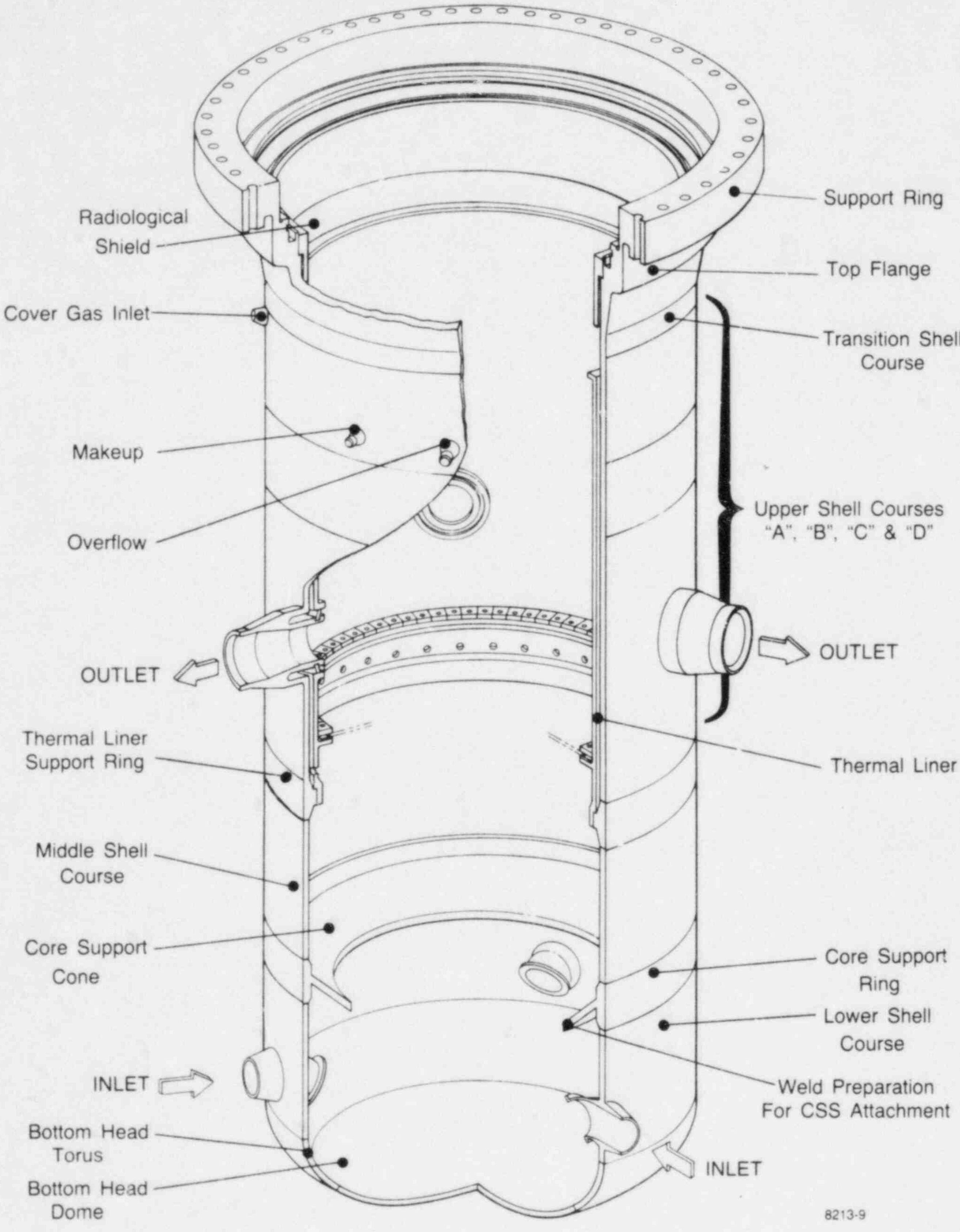
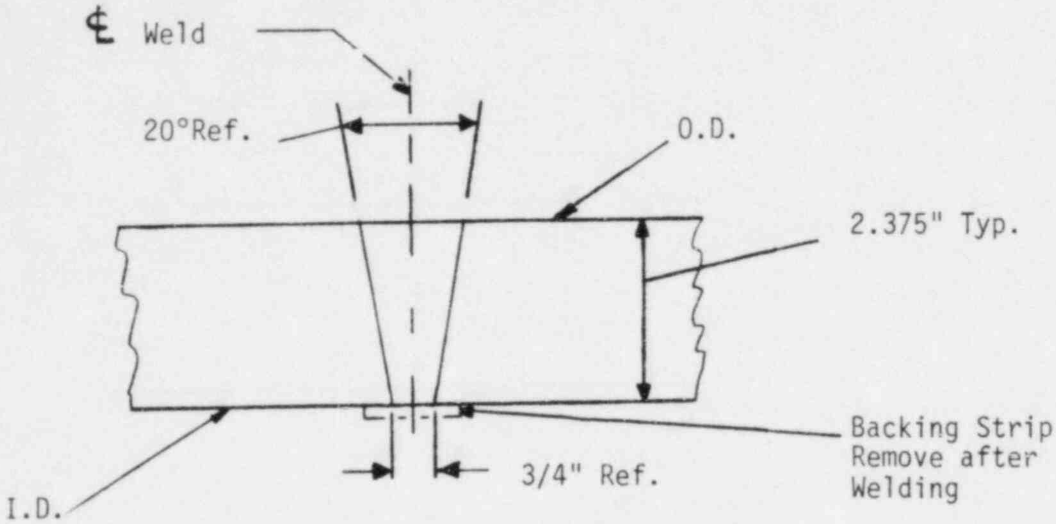
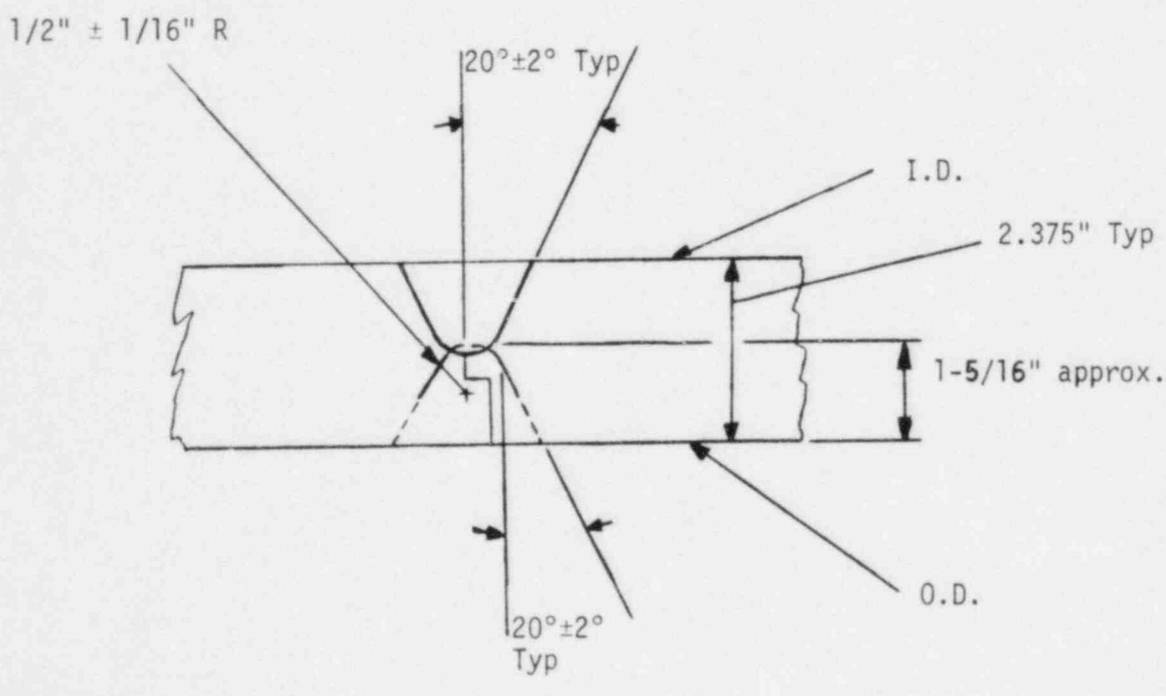


Figure 1.1



a. 20° V-Bevel Weld Joint



b. Ship Lap Weld Joint

Figure 1.2 Reactor Vessel Weld Joint Configurations

2.0 COMPARISON OF CRBRP AND PWR OPERATING CONDITIONS AND MATERIALS

This section of the report provides an overview of operating conditions and materials encountered in reactor vessels for CRBRP and PWR's.

2.1 Differences in CRBRP and PWR Operating Conditions

The CRBRP primary loop vessels contain liquid sodium in the temperature range of 750°F to 1015°F at full power conditions. The maximum pressure is in the cold leg or inlet plenum side of the reactor vessel. The design pressure is 200 psig while the normal pressure is actually 140 psig. The outlet plenum or hot leg has a 15 psig design pressure with the normal outlet plenum pressure expected to be six inches of water. All areas not wetted by sodium contain argon cover gas which maintains the area inert.

The reactor vessel is surrounded by a guard vessel, in the reactor cavity. The reactor cavity is maintained inerted by nitrogen with 0.5% to 2% oxygen to prevent nitriding. The guard vessel and reactor vessel outlet plenum have been sized to maintain outlet nozzle and core submergence in the event of a reactor vessel or contained primary piping leak.

The reactor vessel wall will be maintained below 900°F by use of a thermal liner which also mitigates thermal transient effects on the vessel wall. The vessel wall also is shielded from high levels of neutron radiation, which could induce changes in material properties, by the sodium and both fixed and removable radial shields in the core.

Due to the low vapor pressure of sodium at high temperatures, system pressures can be and are maintained low and therefore, the reactor vessel shells are typically 2-3/8 and 2-3/4 inches thick and made of 304 stainless steel. The LMFBR has a high core ΔT and high temperature compared to the PWR's (See Table 2-1). The high temperatures dictate the use of materials which retain adequate static and creep strength. Type 304 and 316 stainless steels provide the strength required. The high ΔT values and temperature ramp rates at the core outlet require a structural configuration which has a low thermal inertia

in order to minimize thermal stresses and the associated low cycle fatigue damage. The thin wall CRBRP reactor vessel and internal thermal liners mitigate the effects of the high core outlet ΔT and ramp rates. The upper end of the vessel is made of two large SA-508 carbon steel ring forgings (the vessel temperature is sufficiently low to allow use of the carbon steel at this elevation).

PWR primary loop vessels contain borated water at temperatures in the range of approximately 500°F to 575°F at full power conditions. The maximum pressure is approximately 2400 psi with a relatively small difference between the hot leg and cold leg piping. Air is used in the cell surrounding the exterior of the reactor vessel.

The PWR vessels have been sized to contain the high pressures at the operating temperatures. The vessels are carbon steel with internal stainless steel cladding to prevent corrosion. Typical vessel thickness, including the cladding, is 8-11 inches. The PWR vessel shells experience radiation induced embrittlement due to high fluence levels on the carbon steel. This is due to the close proximity of the PWR vessel wall to the core (PWR vessels are smaller than the LMFBR type vessels) and the lower threshold for onset of ductility loss in carbon steel versus stainless steel.

PWR vessels are routinely inspected in-service by removal of the reactor head and having direct access to the inside of the vessel. Closed circuit TV can also be used to view the inspection since the water is clear. The CRBRP closure head is not removable for inspection and the sodium is opaque. The radiation levels in the CRBRP reactor cavity are high at shutdown and prevent any personnel entry into the cavity for inspection. All inspection on the CRBRP RV must be done remotely without direct visual access to the exterior of the vessel. Unlike PWR vessels, which are supported from the nozzle areas, the vessel is supported from the support forging in a low temperature area (~400°F).

2.2 Materials Differences

The CRBRP vessel and primary loop piping is primarily fabricated from austenitic stainless steel, either 304 (SA-240) or 316 (SA-240). These materials are used due to their compatibility with sodium, their properties at the temperatures involved, and the resistance to radiation induced damage. The stainless steel is also easily fabricated and welded.

Carbon steel (SA-508 and SA-533) is typically used on PWR's with internal stainless steel cladding due to the large amount of material used (thick vessels and piping) and the lower cost of the carbon steel. Also, carbon steel can be used due to the lower temperatures involved. The carbon steel is subject to radiation damage which results in embrittlement. As a result of embrittlement the first third of the shell thickness must be closely examined for possible cracking. The stainless steel CRBRP vessel exhibits a ductile material behavior (leak-before break) than carbon steel PWR vessels.

Ferritic carbon steels used for PWR and BWR reactor vessels exhibit an abrupt decrease in fracture toughness when the temperature drops below a value termed the ductile-brittle transition temperature (DBTT). As long as the material temperature exceeds the DBTT, brittle fracture is virtually impossible; at temperatures below DBTT, brittle fracture becomes a credible phenomenon when a small crack of critical crack size is present. Consequently, it is required that the operating temperature of a PWR or BWR reactor pressure vessel exceed the DBTT by a minimum margin. This requirement assures that the vessel material is in the high-toughness regime whenever the vessel is subjected to operating loads.

The difficulty which arises is that the cumulative effect of service irradiation causes the DBTT to shift upwards. The original margin between operating temperature and the DBTT may thus be slowly eroded, or even eliminated. The margin of protection against brittle fracture is accordingly reduced. This chain of events has led to a strong emphasis on materials surveillance and in-service inspection for PWR and BWR reactor vessels.

Physical considerations suggest that less emphasis on material surveillance and in-service inspection is required to assure the long-term adequacy of the CRBRP reactor vessel. The CRBRP vessel is subjected much to lower levels of irradiation than are water-cooled reactor vessels, and experiences essentially no irradiation-induced change in material properties. The other salient factor is that the austenitic stainless steel of the CRBRP does not exhibit a significant change in toughness over a small temperature regime. The toughness of an austenitic stainless steel does vary with temperature, but the variation is gradual with no abrupt reduction in the toughness-temperature relation. Consequently, there is no mechanism by which brittle fracture of the CRBRP reactor vessel can be abruptly rendered credible, as there is in the water-cooled reactor vessel. Therefore, there is less need for in-service inspection of the CRBRP reactor vessel.

In summary, there is a strong technical basis for differences in PWR and LMFBR in-service inspection practice. The primary considerations are:

- o PWR vessels are constructed of carbon steel and LMFBR vessels of stainless steel.

- o Carbon steel has a significantly lower threshold for radiation damage than stainless steel. Because of the close proximity of the PWR vessel to its core, radiation damage causes a reduction in material ductility which may make it susceptable to brittle fracture. Radiation damage thresholds are not reached in the CRBRP vessel and brittle fracture is not a potential failure mechanism.

- o Small cracks in irradiated carbon steel can precipitate brittle fracture. It is therefore important to conduct periodic examinations of carbon steel to preclude cracks growing to a size which could precipitate sudden brittle fracture.

- o Cracks, if they exist in stainless steel vessels, will grow gradually by the action of cyclic straining until they penetrate the wall of the vessel and create a leak.

Leak detection in a stainless steel vessel therefore serves the same function as periodic ultrasonic examination of a PWR vessel.

TABLE 2-1

LMFBR-PWR COMPARISON OF DESIGN CONDITIONS

Parameter		LMFBR	PWR
Temperatures:	Core Outlet Reactor Outlet	1100°F 1050°F	650°F 650°F
Pressures:	Reactor Vessel Outlet Pump Discharge	15 psia 190 psia	2250 psia 2370 psia
Transients:	Core Outlet ΔT Rate Vessel Outlet ΔT Rate	350°F 35°F/Sec 320°F 3°F/Sec	80°F 5°F/Sec 80°F 5°F/Sec
Coolant		Sodium	Water

3.0 NON-DESTRUCTIVE EXAMINATIONS (NDE) PERFORMED AND THEIR ADEQUACY

3.1 ASME Code, Section III and RDT Standard NDE Requirements

3.1.1 Reactor Vessel Specification Requirements

The Reactor Vessel Specification cites the following documents to provide requirements relative to Reactor Vessel inspection during fabrication:

A. ASME Boiler and Pressure Vessel Code, 1974 Edition with Addenda through Winter 1974.

B. ASME Code Case 1594-1, June 30, 1975, Examination of Elevated Temperature Nuclear Components, Section III, Class 1.

C. RDT Standard E15-2NB-T, November 1974 with Amendment 1, January 1975, Class 1 Nuclear Components.

D. RDT Standard F3-6T, December 1974, Nondestructive Examination.

Additionally, the RDT Standards for all materials used on the Reactor Vessel pressure boundary contain additional requirements for examination of base metal material in excess of the ASME code.

3.1.2 NDE Requirements for the Reactor Vessel Welds

The Reactor Vessel contains weld joints which are classified as A, B, C, and D Joints in accordance with Section III. Section III, Division NB specifies acceptable materials, design rules, fabrication and installation requirements, examination requirements, testing requirements and protection against overpressure for nuclear pressure vessels. The joint categories as applied to the Reactor Vessel are as follows: Category A, are longitudinal joints between plates or forged bars which make up shell courses or rings: Category B, are circumferential joints or circle seams between shell courses or rings; Category C, is used only for the joint of the vessel flange to the Inconel 600

transition shell; Category D, are all nozzle to shell course welds. As part of Section III Examination Requirements, it is stated that the inspection of the above welds be conducted in accordance with ASME Section V which details methods for all types of inspection used by the ASME Code. Section III further modifies the methods of Section V for Radiography. The modifications include prohibition of fluorescent screens, requires geometric unsharpness[(1)] not exceed the limits defined in Paragraph T-251 of Section V, and requires penetrameters listed in Table NB-5311-1 be used.

Section III also requires that all weld edge preparation surfaces for joint categories A-D, as applicable to the Reactor Vessel, be examined by Magnetic Particle (MT) or Liquid Penetrant (PT).

Code requirements for inspection on deposited weld metal are radiography (RT) and liquid penetrant or magnetic particle inspection on the external and accessible internal weld surfaces and adjacent base material. Category C welds may include the use of multiple exposures where special radiographic techniques are required to show that corner weld joints are acceptable. (The Reactor Vessel does not contain any corner welds.)

ASME Code Case 1594-1 further modifies Section III of the Code for designated high temperature areas of the Reactor Vessel. Code Case 1594 requires that Category A-D welds be examined volumetrically by one of three methods: (1) Radiography plus ultrasonic; (2) Radiography plus eddy current examination provided the portion of weld examined is less than 1/4 inch thick; or (3) Radiography at two different angles (one within 15° of perpendicular and the other at an appropriate angle to reveal any lack of fusion and cracking at the weld/base material interface near the root pass). The multiple angle shots assure that planar type defects parallel to the heat affected zone are detected. The third option was chosen for the CRBRP reactor vessel.

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(1) Geometric unsharpness is a measure of focus for radiography. Section V provides a formula for the definition of geometric unsharpness and RDT F3-6T gives acceptable values in the form of a graph.

RDT Standard F3-6T also modifies radiographic, liquid penetrant, magnetic particle and ultrasonic inspection requirements called out by Section C of the Code. For radiography, the modifications deal with geometric unsharpness and the use of penetrameters. Liquid penetrant modifications delete water washable penetrant, specify surface preparation and penetration time. Magnetic particle modifications modify the examination coverage. Ultrasonic inspection requirements have more extensive modification. This modification was used only on the carbon steel weld at the top of the vessel.

As can be seen from the above requirements, the basic ASME Code fabrication inspection requirements are satisfied by radiography and either PT or MT on the I.D. and O.D. of welds. The additional requirements imposed by the ASME Code Case and RDT Standards only modify the basic two requirements of RT and PT or MT by imposing multiple volumetric examination of which one option is double angle RT. Ultrasonic (UT) inspection of welds is not required to satisfy any Code or RDT standard fabrication requirements.

3.1.3 NDE Acceptance Criteria for the Reactor Vessel Welds

The acceptance standards for nondestructive examinations are contained in the ASME Code Section III, Paragraph NB-5300. The acceptance criteria are further modified by RDT E15-2NB-T Paragraphs NB-5320, 5330, 5340 and 5350. The following paragraphs will detail the acceptance standards for radiography and liquid penetrant examinations since these were the predominant inspection methods for the reactor vessel.

Radiographic acceptance standards in Section III define the following as being unacceptable: (a) any crack, incomplete fusion or lack of penetration; (b) elongated indications which have a length greater than 3/4 inch for the Reactor Vessel welds; (c) a group of indications in line with dimensions given for lengths and distances between indications; and (d) porosity in excess of that defined by Appendix VI. RDT E15-2NB-T adds additional definition that defines Tungsten inclusions as porosity. Tungsten inclusions are also limited to a maximum of 5 in any 6 inches of weld length with size limited to medium

pore size as shown in Table VI-1132-1 of Appendix VI. Since Tungsten Inert Gas (TIG) welding was not employed on the vessel, this requirement is not applicable.

Liquid penetrant acceptance standards in Section III define the following indications as unacceptable: (a) any cracks or linear indications; (b) rounded indications with dimensions greater than 3/16 inch; (c) four or more rounded indications in a line separated by 1/16 inch or less edge-to-edge; and (d) 10 or more rounded indications in any 6 square inches of surface with the major dimension of this area not to exceed 6 inches with the area taken in the most unfavorable location. The section also defines indications with major dimensions greater than 1/16 inch as being relevant.

RDT E15-2NB-T adds one other acceptance criterion for PT on the reactor vessel. The additional criteria is that aligned indications are unacceptable when the average or center-to-center distance between any one indication and two adjacent indications is less than 3/16 inch.

3.2 Description of Applied NDE Methods

As described in the previous section, reactor vessel austenitic and bi-metallic welds were required to be inspected by radiography and liquid penetrant or magnetic particle. Since the reactor vessel is predominantly austenitic stainless steel, PT was used except on the SA-508 forging weld. The radiographic source used for reactor vessel pressure boundary welds (with the exception of some repair welds) was the 7.5 or 8.0 Mev Linatron. The following paragraphs will discuss PT and RT relative to the reactor vessel.

3.2.1 Liquid Penetrant

The type of liquid penetrant used for the reactor vessel inspection was the solvent removable method. The type of liquid penetrant is required by RDT standard F3-6T. The following sequence of operations was performed:

A. The surface was ground (if necessary) and made acceptable for PT;

B. The weld was thoroughly cleaned and dried;

C. A red dye was brushed onto the weld and allowed to penetrate approximately 10 minutes;

D. The red dye was removed from the surface of the weld by carefully wiping with a lint free cloth;

E. A wet developer was sprayed onto the weld surface to draw out any dye that penetrated;

F. After waiting a minimum of 7 minutes and a maximum of 30 minutes, an evaluation of the surface area was made.

G. The surface area was then cleaned with solvent.

Indications in this test appeared as a red mark on a white backgroud. This method detected tight surface discontinuities such as cracks or porosity. The weld edge preparation, root-pass, the backgouged side and both the outer and inner surfaces of the welds were examined. Any linear indication as well as certain aligned porosity was cause for rejection. A relevant indication was considered to be larger than 1/16 inch. The detailed acceptance criteria were given in Section 3.1.3. This method of detection could readily detect these discontinuities.

3.2.2 Radiography

The two radioactive isotope sources specified for reactor vessel radiography were Cobalt 60 and Iridium 192. Other isotopes may be used for thin materials but are not applicable to the CRBRP vessel. The energy level of gamma rays emitted from an isotope is constant. For Cobalt 60 two energy levels are produced, 1.17 Mev and 1.33 Mev. For Iridium 192 the energy level is .4 Mev.

Because of these energy levels (expressed in million electron volts) the minimum thickness of material that might be penetrated is limited due to adequate radiation attenuation to produce an image. The maximum thickness is limited only by the time required for exposure.

The ASME Code specifies that the minimum thickness for radiography using Cobalt 60 is 1.5 inches and for Iridium 192 the minimum thickness is .75 inches.

X-rays are electromagnetic rays produced when high energy electrons interact with the target of the X-ray machine. The intensity of the beam depends on the number of electrons striking the target as well as the potential difference between the cathode (electron source) and the anode (target). With X-ray machines, the energy level referred to is the peak energy. The pulses of energy build up to that peak and decay to zero from that peak. X-rays, therefore, provide a broad spectrum of energies as opposed to the unique energy levels provided by isotopes.

The Linatron used to radiograph the reactor vessel welds is rated at 7.5 to 8 Mev. The optimum thickness begins at approximately 2 inches thick and can be used for thicknesses up to 20 inches.

The benefit of broad band spectrum of X-rays is that a broad band produces sharp contrast (ability to discern discontinuities against the background). The exposure time is reduced with high energy X-rays, thereby mitigating the image degradation on the film that would result with longer exposure times.

The accuracy of the positioning of the Linatron was assured by using a laser light inherent in the equipment. The code does not accept cracks, lack of penetration or lack of fusion. These types of problems occur mostly at the line of fusion. An X-ray down the line of fusion in this regard will find such indications more assuredly than will an X-ray normal to the weld deposit that might be 15° away from that line. Utilization of multiple angle shots thus assure that planar type defects along the lines of fusion will be detected.

3.3 NDE and Quality Assurance Activities Performed on the Reactor Vessel

3.3.1 Reactor Vessel NDE

The reactor vessel was fabricated in accordance with the requirements identified in Section 3.1. Figure 3.1 shows the weld numbers assigned to vessel welds. Table 3-1 documents the NDE performed on the reactor vessel in a tabular form and will be further explained in subsequent paragraphs. Weld WR-60 at the top of the reactor vessel development shown on Figure 3.1 is the carbon steel weld between the two SA-508, Class 2 ring forgings at the top of the vessel.

Table 3-1 lists all pressure boundary welds in the reactor vessel and is grouped into circle seams, longitudinal welds, and nozzle welds. As can be seen in the table, all welds had radiography and liquid penetrant applied for inspection (except WR-60 which had magnetic particle instead of liquid penetrant).

All welds with a designated design temperature of 900°F (all welds above the thermal liner support ring) received double angle X-ray. The bi-metallic welds (WR-16 and 18) received one normal shot and four angle shots. This examination was in excess of that called for by the requirements described in Section 3.1.3 since the design temperature for this section of the vessel is below 900°F and only one shot is required by the ASME Code. In addition to the final X-rays performed on the WR-18 (SA-508 to Inconel 600) weld, preliminary normal and angle shots were taken for information. The four angle shots oriented the X-ray beam down each 20° sloped face of the double J-prep weld as shown on Figure 1.2. This number of multiple shots oriented on each face gives extremely high confidence that planar flaws at the weld/base metal interface would have been discovered.

All reactor vessel welds were radiographed using a 7.5 or 8.0 Mev Linatron source. Repair welds were radiographed with either the Linatron or an Iridium 192 source. As can be seen from the chart, some welds received preliminary shots which were taken prior to final radiography and in some cases double

angle X-ray was used for information, although not required. The WR-60, SA-508, carbon steel weld received both a preliminary and final ultrasonic inspection in addition to radiography and magnetic particle inspection.

3.3.2 Quality Assurance

During reactor vessel fabrication, a permanent Purchaser Quality Assurance representative was stationed at B&W. This Quality Assurance engineer was qualified to the ASME Code, Level III requirements. First, all RT films were reviewed by the Fabricator Quality Assurance engineer for acceptability. Next, radiographs were overviewed by the Purchaser's representative. Finally, a second overview was performed by another Level III Purchaser Quality Assurance engineer who spot checked the radiographs on a random sampling basis.

Liquid penetrant inspection was witnessed for acceptability by the Fabricator's Quality Assurance personnel. The on-site Purchaser Quality Assurance engineer overviewed a large amount of the total liquid penetrant checks.

Audits were performed and documented to assure conformance with specification requirements.

3.4 Factors Influencing Weld Strength and Bi-Metallic Weld Operating Experience

The reactor vessel is fabricated from 304 austenitic stainless steel with an Inconel 600 transition shell to the SA-508 carbon steel used in the low temperature area at the upper end of the vessel. The operational environment of the reactor vessel transition joint welds is benign and minimizes concern for in-service materials related problems.

Bi-metallic weld strength is discussed in detail in the PSAR Appendix G, Pages G-43 thru G-53 which are attached. The following is a summary of the discussion.

A literature review was performed to evaluate the performance of bi-metallic welds and to classify the causes of observed failures. The factors which contribute to dissimilar metal and weld failure were found to fall into four basic categories: 1) cyclic thermal stress, 2) oxidation at the weld interface, 3) carbon migration away from the interface, and 4) metallurgical deterioration at elevated temperatures. Of these four factors, only one, high temperature operation, was a common factor in all observed failures. Each of the factors was reviewed relative to the CRBRP reactor vessel bi-metallic weld and shown to have effectively no potential to initiate failure.

Bi-metallic weld joints have had a long and satisfactory operating experience. A presentation was made to the Nuclear Regulatory Commission April 6-7, 1982 by P. Patriarca and G. Goodwin concerning "Transition Joint Experience and Technology" (Reference 16). The following is a summarization of the information presented:

PWR's and BWR's use SA-508 and SA-533 ferritic, reactor vessels, clad with 308 stainless steel. The ferritic nozzles are welded to a 304 stainless or Inconel 600 safe end by applying Inconel 82 weld metal which serves as a transition to the 304 nozzles. The CRBRP transition joint is comparable to the PWR and BWR joint since the SA-508 is welded to the Inconel 600 by Inconel 82 weld metal and the Inconel 600 is welded to the 304 stainless by Inconel 82 weld metal.

PWR's that have ten years or more of service have subjected the transition welds to various loads and temperature conditions with satisfactory service, see Table 3-2. BWR's with 9 to 22 years of service have operated without any reported incidents of failure, see Table 3-3. EBR-II has transition welds and has 19 years of operating experience. BN-350 on the Caspian Sea has been on line for 9 years with satisfactory operation, see Table 3-4. Fossil fuel plants have used transition joints for more than 20 years. In 1977, an Oak Ridge report covered a

transition joint from a superheater in a fossil plant that had been operating for 17 years. It ran at 1125°F with 146 thermal cycles at approximately 200°F/hour.

The reactor vessel stainless steel welds are also in a benign environment. The reactor vessel liner and outlet nozzle liner serve to protect the vessel pressure boundary from high temperatures and mitigate transients. The maximum full power steady state temperature of 875°F on the 304 SS is sufficiently low that metal loss due to sodium corrosion will be negligible (<.1 mils over 30 years). Furthermore, at the very low flow rates involved, no allowances are necessary for erosion losses.

The carbon equilibrium value at 875°F is of the order of 1000 ppm, thus neither carburization nor decarburization is likely to be encountered at the highest temperatures. Material exposed to lower temperatures will experience some slight carburization, possibly to a carbon level of the order of 2000 ppm. However, due to the low carbon diffusion coefficients at these temperatures, the carburized layer will not achieve any significant thickness.

In addition to effects due to sodium exposure, consideration was also given to the possibility of neutron-radiation induced embrittlement. The maximum total neutron fluence at the core mid-plane on the reactor vessel is estimated to be 3.1×10^{20} n/cm^2. Generally, the neutron fluence threshold at which radiation effects such as ductility loss begin to be observed in the austenitic stainless steels and their weldments is considered to be 10^{21} n/cm^2. Below this value, radiation effects may be ignored. Since the predicted fluence is only one-third of this threshold value, no radiation damage is foreseen for the reactor vessel.

Consideration has also been given to the possibility of material property changes arising from prolonged operation at 875°F. Thermal aging is known to be detrimental to the austenitic stainless steels and their weldments, particularly when a brittle intermetallic phase is produced and embrittlement results. However, the maximum operating temperature of the reactor vessel

weldments (875°F) is much lower than that required for brittle intermetallic phase formation (approx. 1000°F minimum) and no embrittlement is expected to result from this reaction. The other effect of thermal aging is to induce carbide precipitation, leading to increases in tensile and yield strengths, and associated losses of ductility. Again, although some small amount of carbide formation is likely in the reactor vessel, the low operating temperature will ensure that this reaction will remain very slow and thus will not produce any measurable changes in mechanical properties over the plant lifetime.

TABLE 3-1

REACTOR VESSEL NDE SUMMARY

WELD NO.(1)	DYE PENETRANT(2)		RADIOGRAPHIC SOURCE(3)	RADIOGRAPHY			
	Preliminary ID & OD	Final ID & OD		Preliminary		Final	
				Normal	Angle	Normal	Angle
CIRCUMFERENTIAL							
WR-18	X	X	7.5 or 8.0 Mev	1	1	1	4
WR-16		X	LINATRON			1	4
WR-14		X				1	1
WR-40		X				1	1
WR-12		X				1	
WR-10		X				1	1
WR-8	X	X				1	
WR-6	X	X		1(5)		1	
WR-4		X				1	
WR-2		X				1	
WR-60(4)(6)				1		1	
LONGITUDINAL							
WR-17	X	X				1	
WR-15	X	X				1	1
WR-39	X	X				1	1
WR-13	X	X				1	1
WR-11	X	X				1	
WR-9	X	X				1	1
WR-7	X	X		2		1	
WR-5	X	X				1	1
WR-3		X				1	

TABLE 3-1 (Continued)

WELD NO.(1)	DYE PENETRANT(2) Preliminary ID & OD	DYE PENETRANT(2) Final ID & OD	RADIOGRAPHIC SOURCE(3)	RADIOGRAPHY Preliminary Normal	RADIOGRAPHY Preliminary Angle	RADIOGRAPHY Final Normal	RADIOGRAPHY Final Angle
NOZZLES			7.5 or 8.0 Mev				
WR-28		X	LINATRON			1	
WR-27		X				1	
WR-26		X				1	
WR-25		X				1	
WR-24		X				1	
WR-23		X				1	

NOTE (1) - See Figure 3.1

NOTE (2) - P_T was performed on all weld preps and backgrooved or backgouged areas

NOTE (3) - Some repair welds were radiographed with an IR-192 source

NOTE (4) - M_T was performed on preliminary O.D. & I.D. and final O.D.

NOTE (5) - R_T was performed on first half

NOTE (6) - U_T was performed both preliminary and final

TABLE 3-2

TRANSITION JOINTS IN PWR PLANTS IN OPERATION
IN EXCESS OF TEN YEARS

Unit	MW(e)	Nuclear Supplier	Commercial Operation (Month/Year)
Shippingport	150	Westinghouse	12/1957
Yankee-Rowe	175	Westinghouse	7/1961
San Onofre-1	430	Westinghouse	1/1968
Connecticut Yankee	582	Westinghouse	1/1968
Ginna	490	Westinghouse	7/1970
Point Beach-1	497	Westinghouse	12/1970
Robinson-2	665	Westinghouse	3/1971
Point Beach-2	497	Westinghouse	10/1972
Surry-1	788	Westinghouse	12/1972
Maine Yankee	825	Combustion Engineering	12/1972
Turkey Point-3	728	Westinghouse	12/1972

TABLE 3-3

TRANSITION JOINTS IN BWR PLANTS IN OPERATION
IN EXCESS OF TEN YEARS

Unit	MW(e)	Nuclear Supplier	Commercial Operation (Month/Year)
Dresden-1	207	General Electric	8/1960
Big Rock Point	71	General Electric	12/1965
Genoa-2	48	Allis-Chalmers	11/1969
Oyster Creek	620	General Electric	12/1969
Nine Mile Point-1	610	General Electric	12/1969
Dresden-2	794	General Electric	7/1970
Millstone-1	652	General Electric	12/1970
Monticello	536	General Electric	6/1971
Dresden-3	794	General Electric	10/1971
Pilgrim-1	655	General Electric	7/1972
Quad Cities-1	789	General Electric	8/1972
Vermont Yankee	514	General Electric	12/1972

TABLE 3-4

TRANSITION JOINTS IN LMFBR DEMONSTRATION PLANT SUPERHEATERS
OPERATING IN EXCESS OF TEN YEARS

Parameters	EFAPP (US)	EBR-II (US)	BN-350 (USSR)	SNR-300 (FRG)
Years Operated	1963–1973	1963–Present	1973–Present	1983 Criticality
MW(t)	200	62	1000	762
MW(e)	60	19	150*	312
Type of Steam Generator Unit	Once-through single wall involute	Recirculating duplex tube	Shell & Bayonet Single Wall	Once-through 2 loops straight tube single wall 1 loop helical
Number of Units/Plant	3	1	12	9
Superheat Steam Temperature, °C(°F)	416(780)	438(320)	435(815)	495(920)
Steam Pressure, MPa (psig)	6.21(900)	8.62(1250)	5.67(735)	1.59(2300)
Sodium Inlet Temperature, °C(°F)	438(820)	465(870)	450(842)	526(980)
Tube Material	2 1/4 Cr-1 Mo	2 1/4 Cr-1 Mo	2 1/4 Cr-1 Mo	2 1/4 Cr-1 Mo-Nb

*BN-350 — Balance for Desalting.

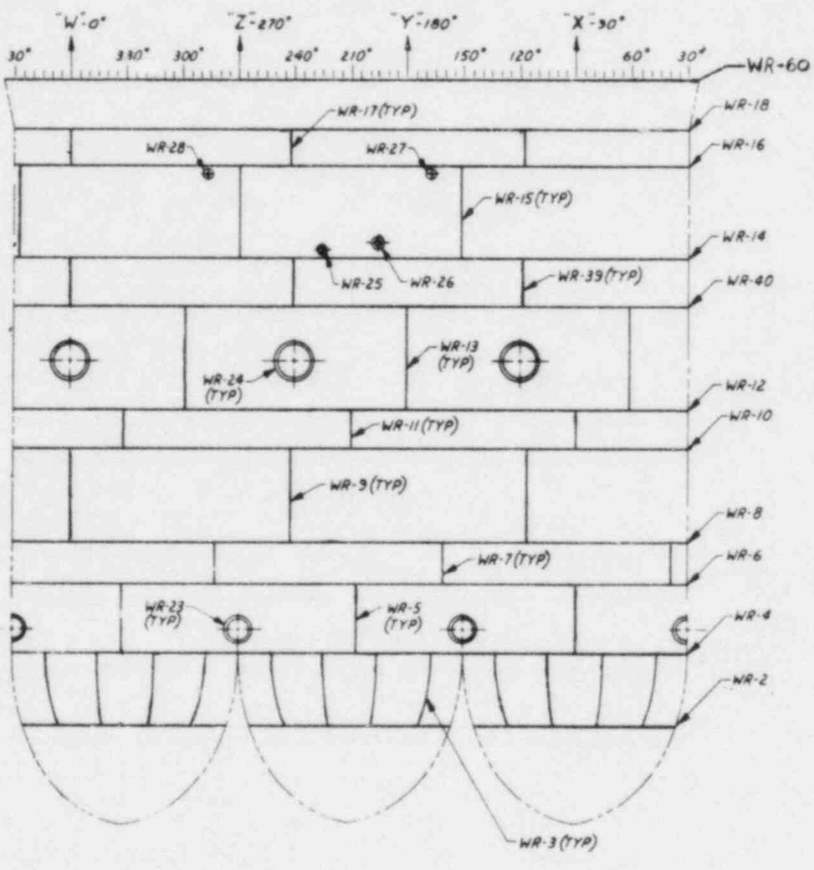

Figure 3.1 Reactor Vessel Weld Development

4.0 ASSESSMENT OF ULTRASONIC INSPECTION CAPABILITY OF THE CRBRP REACTOR VESSEL

This section will present a short description of ultrasonic principles, methodology, and the effects of material properties on ultrasonic inspection. A survey of the major nuclear pressure vessel fabricators and national laboratories was performed to assess current ultrasonic inspection technology for the examination of stainless steel and CRBRP type bi-metallic weldments. Discussions with the light water reactor industry indicate that the inspection methods developed to assess stainless weld integrity in the commercial nuclear utility arena are at the forefront of ultrasonic technology. Such development has occurred since the ultrasonic development program which was initiated during fabrication of the FFTF reactor vessel.

4.1 Description of Ultrasonic Methodology

4.1.1 Review of Ultrasonic Examination Principles

Ultrasonic testing is a nondestructive examination method for detecting flaws or discontinuities within materials capable of conducting sound. The technique which uses high frequency vibratory energy in the 0.5 to 25 megahertz range, is particularly suited for finding, locating, and characterizing relatively small discontinuities in heavy section materials and fabricated components.

In ultrasonic testing, an electrical pulse is applied to a piezo-electric crystal or transducer, causing it to vibrate. When this vibration is applied or coupled through an incompressable liquid to a sound conducting material, a pressure wave is induced in the material and is propagated until interrupted and/or reflected by a discontinuity or boundary. The reflected energy is detected by the same or another transducer. These detected signals are amplified and visually presented, usually on a cathode-ray tube, for analysis to determine the size, location, and other characteristics of the discontinuity or boundary.

The longitudinal and shear wave modes of propagation or vibration are the modes used in the ultrasonic examination of heavy section steels and weldments. The longitudinal mode exists when the motion of the particles in the medium is parallel to the direction of propagation. The advantages of the longitudinal mode are:

A. Easily generated and detected.

B. High velocity of travel with shorter wave lengths resulting in good penetrability (low attenuation), directivity and sensitivity.

In the shear wave mode, the particle movement in the medium is at right angles to the direction of wave propagation. Shear waves have a velocity that is approximately one-half that of the longitudinal mode. Because of this lower velocity, the wave length of shear waves is much shorter than that of the longitudinal waves. The size of the flaw that can be detected using shear waves is therefore on the order of one-half that detectable using the longitudinal mode. In addition to the increased sensitivity available, the shear wave techniques provide detection of near surface flaws and flaws oriented at varying angles from the test surface.

The instruments and test systems used in ultrasonic testing must be calibrated. This is usually done by using reference or calibration blocks to establish the capabilities of the equipment and test system. The use of reference blocks is the recommended way to provide (1) simulated defects for comparison purposes, (2) attenuation characteristics, (3) flaw evaluation and sizing estimates, (4) flaw location and orientation determinations, etc. Materials for reference blocks should be of the same alloy type, heat treat condition, product form, length, and surface conditions, etc., so that attenuation characteristics, test frequencies, sensitivity levels and acceptance criteria can be accurately established. Reference blocks for weldments, particularly austenitic and bi-metallic weldments should duplicate the joint to be examined with regard to size (thickness), joint configuration, weld and base metal compositions, product forms, pre-heat, post heat and interpass temperatures and deposition position.

The scanning procedure, i.e., the wave mode (longitudinal or shear test direction, straight or angled test surface, reporting level etc.) to be used in examining the test item depends on the type, size, orientation and location of the discontinuity considered to be detrimental. The straight beam, longitudinal mode techniques are used primarily to detect laminations, inclusions, porosity, segregation, shrinkage, bursts, flaking, etc. in plate, forgings and castings. The angled shear and longitudinal wave techniques are best suited for the detection of radial cracks, lack-of-fusion, slag, porosity and intergranular stress corrosion and fatigue cracking in welded structures and components (vessels). The angled techniques are also useful in the detection and characterization of laps, cracks and other near surface discontinuities oriented at varying angles to the test surface. For best results in using either the shear or longitudinal modes, angles or straight beam techniques, it is recommended that the surfaces for ultrasonic testing be uniform, preferably 250 rms or better and free from waviness.

4.1.2 Effects of Materials on Ultrasonic Inspection

The difficulties in connection with ultrasonic inspection of austenitic welds are well documented in the literature (Reference 1). The problems are associated with the metallurgical characteristics of the deposited weld metal which dominates the UT wave propagation and response. The elastic anisotropy of the different grains leads to signal scattering coupled with mode conversion problems (Reference 2).

The dendritic structure in the weld zone is characterized by large preferred orientation, columnar grains which result from slow cooling and directional solidification during welding. The dendritic structure serves as a diffracting and diffusing medium to normal ultrasonic examination techniques using shear or transverse wave modes. The UT wave velocities and propagation modes within the dentritic structure exhibit anisotropy, which is described by well known relations derived for face-centered cubic crystal structures, i.e., the austenitic phase. These variations in wave velocities with crystallographic direction produce large acoustic impedance differences at grain boundries and near the edges of the dendrites due to composition gradients. These gradients form during cooling when alloy content changes as

lower melting point microconstituents occupy volume between dendrites. This microstructure dominates the wave propagation response and a substantial portion of the UT beam is reflected by local impedance changes at grain boundries. The resultant signal responses are often described as grain noise and are difficult to interpret (Reference 3).

The extent to which the microstructure dominates UT wave propagation response is evident at the austenitic base and weld material interface. The interface is comprised of the random crystallographic orientation of the weld metal which results in a partially reflective boundary to the UT beam.

One of the problems encountered with meaningful ultrasonic examination of austenitic materials has been the assumption of similar acoustic properties for austenitic materials. Even material of similar composition such as Type 304 or Type 316 can have acoustic properties drastically altered by heat treatment or method of processing. Additionally, sensitization of austenitic materials is a common cause for alteration of acoustic properties. Examination of wrought Type 316 stainless steel with shear waves using 2.25 MHz transducers may pick up a reflector while one of similar size in cast 316 would not be detected. Sundstrum (TRC) and Greer (SwRI) noted additional UT problems with stainless steel material in which Inconel buttering is utilized (Reference 4).

Ultrasonic signals reflected off of grain boundries, and base/weld metal interface preclude complete and reliable examination using conventional techniques. The current thrust of ultrasonic research programs is continued testing of strongly attenuating structures and computerized signal analysis which is necessary in order to optimize signal interpretation (Reference 5).

4.2 State of the Art Ultrasonic Technology

A survey of the major nuclear vessel fabricators and national laboratories was performed to assess the state-of-the-art technology for ultrasonic examination of austenitic and bi-metallic weldments. Generally, there is optimism that

ultrasonic examination of austenitic materials can be performed. However, extensive procedural development and operator training is necessary to assure meaningful interpretation.

Ultrasonic examination performed by some of the fabricators utilized the angle beam shear wave technique. This conventional angle beam technique is believed to be limited to a maximum thickness range of 1-1/2 inches. However, to develop even that capability, one fabricator spent many months developing the technique, procedure and extensive training of UT personnel. Another fabricator has refused to UT stainless welds and plans no further investigation into UT development capability. Investigation is continuing in the use of refracted longitudinal wave examination and so far has been successful in resolving the T/4, T/2 and 3/4 T calibration holes in a 2 inches thick weld with a special crystal. The consensus of vessel fabricators contacted, is that UT of stainless steel weldments of 1-1/2 inches thickness is the upper limit of current technology (Reference 6).

A number of national laboratories have on-going research programs to develop and optimize ultrasonic techniques for examination of austenitic materials and stainless steel welds. Stainless welds require separate development based on process and joint geometry. The sharp demarcation between the weld and base metal is difficult to resolve for signal interpretation and analysis. Some success has been met in detecting intergranular stress corrosion cracking (IG SSC) with the shear wave mode (Reference 7).

At this time, there are no foreign LMFBR type reactors routinely performing in-service ultrasonic inspection of the reactor vessel. The French plan to do periodic UT of the reactor vessel in Super Phenix, as well as, visual inspection on the exterior surface. An in-service inspection system has been developed for use, however, its first operation is not expected until near the end of 1983. This system is untested and results of inspection have not been obtained. Sodium leak detection systems are also installed in Super Phenix. The German Reactor, SNR-300, has no provisions or plans for volumetric or surface examinations unless the leak detection system indicates a problem.

inspection to assure all defects were detected. The Japanese reactor, MONJU, will periodically be examined visually and has continuous monitoring using sodium leak detection. No ultrasonic inspections are planned for the reactor vessel. CRBRP will continue to monitor foreign reactor design and experience for any developments in the area of in-service inspection.

Reliable signal analysis for defect determination is operator dependent and differences in defect determination can result. Automatic signal analysis is not yet commercially available and is still in the development process. Both the fabricators and national laboratories surveyed appear to be in agreement that stainless steel weld material presents unique difficulties to obtaining meaningful, repeatable and reliable weld integrity information. Many research programs are attempting to optimize stainless steel signal analysis, computer acquisition and analysis techniques which will permit repeatable defect location identification and techniques which will classify signals in order to size stainless steel weld defects.

4.3 Technical Feasibility of Performing Ultrasonic Examination on the Reactor Vessel

An extensive assessment of the feasibility of ultrasonic examining stainless steel welds was performed in 1973 as part of the FFTF reactor vessel contract, (references 8 thru 14). Ultrasonic examination of the FFTF reactor vessel welds utilized angle beam shear mode, straight beam longitudinal mode, single and double transducers, a delta technique, variations in frequency, pulse-echo, and angle beam longitudinal mode tests at 4, 9, and 11 degrees in a pitch-catch technique. All efforts to develop a reliable ultrasonic procedure for stainless steel weld examination were unsuccessful. It was determined that the variation in the acoustic properties of the FFTF reactor vessel simulated test block weld was so large that meaningful ultrasonic examination was not possible. At that time it was determined that further UT examination of the FFTF reactor vessel welds would be deferred based on the inconclusive results obtained to that date using "state-of-the-art" technology.

The various problems encountered at the time of FFTF reactor vessel fabrication and the ultrasonic development program were mainly due to the variability of the stainless steel weld associated with differing weld techniques and weld positions. This problem can be minimized to a degree by using calibration blocks made with the same weld techniques, duplication of the point to be examined with regard to size (thickness), joint configuration, weld and base metal compositions, product forms, preheat, post-heat and interpass temperatures and deposition position.

A meeting was held at ARD, March 1, 1983, with representatives of Westinghouse Nuclear Technology Division (NTD) in order to discuss technical feasibility of performing meaningful ultrasonic examination of the CRBRP reactor vessel. NTD noted that it is required to volumetrically inspect the stainless primary coolant loop piping of pressurized water reactors. They inspect both centrifugally cast and wrought stainless steel pipe with thicknesses up to 3.0 inches. The following advances in ultrasonic technology were discussed:

A. There have been advances in transducer design which eliminate some of the problems. The transmitter and receiver are now housed within the same element which reduces spurious noise.

B. The refracted longitudinal wave is used, which reduces the effect of the metallurgical structure. If the defect lies within 0.5 inch of the surface on which the crystal is placed, the defect can be located and interpreted with a very high level of confidence using a 70° dual element longitudinal wave search unit. The confidence level is lower (but useful) when the inspection is made from the opposite surface.

C. If an in-service inspection is to be performed and compared with a pre-service examination baseline, the crystal must be positioned to within 1/10 inch of the position it occupied during the pre-service examination. The 1/10 inch positioning accuracy is consistent with current PWR ISI tooling. Development of inspection tooling would be required to assure 1/10 inch position accuracy on the CRBRP reactor vessel.

D. Transducers, couplants and wedges with capabilities up to 400°F exist. This represents the limit of their capability. Development would be required to verify long term stability. Questions relative to specific application of high temperature UT remain to be resolved. Potential problems include development of couplants compatable with materials in the reactor vessel/guard vessel annulus, application and containment of the couplant, removal of the couplant, and transportation of the UT device for accurate positioning.

E. NTD reports over 200 PWR stainless steel welds have been ultrasonically examined up to 2-1/4 inches to 2-1/2 inches thick and 30 inches O.D. No repairs were found to be required as a result of these pre-service and in-service examinations

In summary, the problems which were encountered in attempting to apply ultrasonic technology to the FFTF reactor vessel have to some extent been resolved. There have been the following advances in search unit design that optimize signal interpretation.

A. Signal resolution is improved through the use of improved transducer materials so that high frequency (5.0 MHz) resolution is now available in 2.25 MHz and 1.0 MHz transducers.

B. Spurious noise is reduced through advances in transducer design in which the transmitter and receiver are now housed in the same unit.

C. Control over signal direction is obtained with new wedge materials which permit smaller sound beam angles.

D. Smaller search units with improved instrumentation have improved accessability. Dual element angle beam units are commercially available which enable separate functions of transmitting and receiving.

E. The detrimental effect of the metallurgical structure on interpretation has been reduced with the 70° dual element, refracted longitudinal wave search unit.

These advances coupled with a greater knowledge of the metallurgical characteristics of the austenitic weld metal/base metal interface problems and a restricted inspection volume (1/3 T on the inner diameter of PWR primary coolant piping) have resulted in a higher confidence level for detection of smaller discontinuities. While many research programs are developing methods for optimization of stainless steel signal analysis, further development of computer acquisition and analysis techniques is necessary to permit repeatable defect location and identification and classification signals in order to size stainless steel weld defects.

Stringent fabrication process controls and nondestructive examinations were imposed on the CRBRP reactor vessel and result in a level of confidence equivalent to that obtained for ultrasonic inspections. As reported by Westinghouse NTD, no repairs were found to be required after UT examination of over 200 PWR stainless steel welds in the primary coolant loop piping. Additional nondestructive examination at this time would result in extra costs and project delay without design, technical advantage or an increased weld integrity confidence level.

5.0 PHYSICAL CONSTRAINTS ON PERFORMING PRE-SERVICE ULTRASONIC EXAMINATIONS

5.1 Fabrication Status and Accessibility

5.1.1 Fabrication Status

The Reactor Vessel has been fabricated and is presently in storage at the B&W Facility in Mt. Vernon, Indiana. This facility has been closed and is staffed only to maintain storage of the vessel. No fabrication capability currently exists. Upon completion of fabrication, the vessel was final cleaned and packaged. The packaging for the vessel consists of a multi-layered heavy thickness bag which is a one-piece fabrication. The vessel is purged both inside and between the bag and vessel by a recirculating dry air system. The air is dryed by a dehumidifier which also has a backup dehumidifier.

The vessel is stored on the shipping frame in the shipping configuration. The nozzles have temporary storage covers installed. The vessel lift beam is installed on the vessel flange and tied to the shipping skid. The vessel is also tied to the shipping skid by holddown straps.

5.1.2 Accessibility for Further Inspection

Because the reactor vessel is in a bagged storage configuration, to perform any additional inspection, the vessel must have the storage bag removed to gain access. In order to remove this bag the reactor vessel must be separated from the shipping skid and upended. Upending requires crane capability and shop height, such as exists in a nuclear vessel fabrication shop. It is currently planned to remove the vessel from storage and send it to a fabricators facility so that the lower internals can be installed prior to shipment to the construction site. Additional UT inspections required could be accomplished at that time since access will be available. Any ultrasonic inspection of vessel shell welds behind the vessel liner or radiological shield must be performed from the outside or with the use of specially designed equipment since access to the welds would have to be through the 2 1/2 inch annulus between the vessel and thermal liner.

Another time period for additional inspection would be at site prior to installation when the vessel has been upended and the storage bag removed. However, this time period will be on the critical path for installation and would be highly undesirable.

5.2 Consequences of False Indications

If the reactor vessel is inspected by ultrasonic methods, based on previous discussion of Section 4.0, it is possible that some false signals or indications could be obtained. Pre-service inspection acceptance criteria have not been identified; therefore, it is assumed, that any indications beyond fabrication acceptance limits for ultrasonic inspection would be excavated and repaired. The carbon steel-to-carbon steel joint of the vessel flange to vessel support has been ultrasonically inspected. The remainder of the vessel has not been ultrasonically examined.

The vessel was completed by B&W and Code Stamped upon completion of the pressure testing. If any further work is performed on the vessel prior to installation and system pressure testing, the Code Stamp is invalidated. Additionally, the pre-installation vendor has not been chosen and, due to the status of the B&W, Mt. Vernon Facility, it is doubtful that the reactor vessel fabricator would perform any necessary repair welding and retesting.

The site system pressure testing cannot be substituted for vessel testing since it will only be tested for the 15 psig system pressure. The vessel outlet plenum was tested for the 15 psig system pressure plus the equivalent hydrostatic sodium head in the outlet plenum. The reactor vessel inlet plenum was tested for the 200 psig design pressure. If it becomes necessary for the lower portion of the vessel to be retested at the pre-installation vendor, the plugs for the core support structure to separate the upper and lower vessel would need to be re-fabricated since they were dispositioned as scrap. The test cover for the vessel outlet plenum pressure test has also been dispositioned as scrap and would need to be re-fabricated. The pre-installation vendor would need to retest and Code stamp the vessel under the vendor's Code authorization.

An alternative to repressure testing as a result of weld repairs would be to attempt to obtain a Code Case on this subject, but if not approved, pressure testing of the vessel would be required. A Code Case would have to be processed and resolved prior to pre-installation.

Another consideration with respect to any vessel repair is the weld of the Inconel 600 steel to SA-508 carbon steel forging (WR-18) which was stress relieved after welding in accordance with ASME Code requirements. If any additional weld repair is performed on this weld joint, due to ultrasonic indications, the joint would require a subsequent stress relief or use of the half bead weld technique. Due to the size of the completed vessel a local stress relief would have to be performed rather than a furnace stress relief.

6.0 CONCLUSIONS

A review of pre-service requirements for PWR and LMFBR vessels identifies that differences exist in inspections being performed or planned. However, the philosophy involved in defining requirements for the two types of systems is the same; to perform sufficient fabrication, pre-service and in-service inspection such that degradation of vessel structural capability will be identified and corrective action taken prior to the occurrence of a safety related failure. Differences in inspection requirements are the result of technical considerations associated with the materials of construction and operating conditions to which each system is exposed. PWR vessels are constructed of heavy walled, low alloy carbon steel which is subject to radiation degradation over its lifetime. The CRBRP vessel incurs essentially no measurable radiation damage. A primary concern and reason for volumetric in-service inspection in PWR vessels is brittle fracture. High operating pressures in PWRs result in large amounts of stored energy which influence the nature of crack propagation and amplify the concern for failure due to brittle fracture. Similar abrupt type failure mechanisms do not exist in the stainless steel CRBRP vessel. The CRBRP reactor vessel is surrounded by a guard vessel which, in the unlikely event of a leak, is sized to contain and maintain sodium at minimum safe level or above. The annulus formed by the RV and guard vessel is monitored by leak detectors. These operational differences are reflected in the ASME Code, Section XI inspection requirements. Volumetric in-service examination is required for PWR's; visual examination and continuous monitoring leak detection is specified for LMFBR's. Inspection requirements of the ASME Code will be met for the CRBRP vessel austenitic and carbon steel welds. An exception to in-service requirements for the CRBRP vessel transition welds have been taken based on the low loadings and benign operating conditions imposed. Special attention has been given to rationale supporting this exception, which is based on historical data as well as specific assessment of the weldment.

A. The CRBRP vessel welds have been extensively examined and meet all fabrication requirements of the ASME Code. Multiple angle, broad spectrum X-rays have been taken for all welds having a 900°F design temperature and for the welds on both sides of the Inconel 600 transition joint. Multiple angle examination provides high assurance that all flaws including planar cracks are detected. The bi-metallic weld is made using a high nickel filler metal. This filler metal in conjunction with the low operating temperature of the welds virtually eliminate the historical causes of bi-metallic weld failure.

B. An assessment was made to evaluate the current state of the art for application of ultrasonic examination to austenitic weldments. It was found that considerable progress had been made in overcoming problems encountered in the attempt to UT the FFTF vessel welds. Improvements are mainly in the area of crystal and couplant design, and in the use of highly prototypic weldments to calibrate equipment and train operators in the interpretation of results. However, UT examination of austenitic welds remains subjective, especially in the interpretation of results. It has been concluded that although useful information may be obtained from such examinations, application of it to the CRBRP vessel is unlikely to significantly increase assurance that flaws of unacceptable size do not exist in the weldments. This conclusion is supported by PWR experience which is based on examining approximately 200 weldments without identifying any repairable defects in the 2 1/2 inch thick stainless primary coolant loop piping.

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