4.5 TESTS AND INSPECTIONS

4.5.1 GENERAL

Shop inspection and tests of all major components were performed at the vendor's plant prior to shipment. An inspection at the site was performed to assure that no damage has occurred in transit. Testing of the Primary Coolant Systems was performed at the site upon completion of the Plant construction. These tests included hydrostatic tests of primary and secondary loops. A complete visual inspection of all welds and joints was performed prior to the installation of the insulation. All field welds were radiographically and dye penetrant inspected in accordance with the requirements of the ASME B&PV Code, Section III, Class A, 1965, W65a and special erection specifications prepared by Combustion Engineering.

A hot flow test was made of the primary loop up to zero power operating pressure and temperature without the core installed. The system was checked for vibration and cleanliness. Auxiliary systems were checked for performance.

4.5.2 NIL DUCTILITY TRANSITION TEMPERATURE DETERMINATION

The reactor vessel is designed and fabricated in such a manner that significant operational limitations will not be imposed on the Primary Coolant System resulting from shifts in reactor vessel Nil Ductility Transition (NDT) temperature. The vessel material monitoring program is designed within the guidelines of ASTM E 185-66, "Recommended Practice for Surveillance Tests on Structural Materials in Nuclear Reactors." The preirradiated NDT temperature of the baseplate material was established using drop weight tests in accordance with ASTM E 208-63T and correlations were made with Charpy impact specimen tests conducted in accordance with ASTM E 23-60. This correlation, along with the Charpy impact specimens irradiated in the surveillance program, is used to monitor vessel material NDT temperatures. For the drop weight tests (performed for unirradiated materials), the test temperature is selected to bracket the NDT temperatures of the material.

The NDT temperature is determined by testing initial specimens from the plate material used in the intermediate and lower shell courses of the reactor vessel. The plate with the highest NDT temperature was selected from these initial tests and then a series of specimens from this plate were used to establish the NDT temperature. The NDT temperature is considered to be fixed when specimens 10°F above the temperature judged to be the NDT temperature exhibit a no-break performance.

For the preirradiated Charpy tests, a minimum of three specimens of each material were tested at any one temperature. Tests were performed at a sufficient number of different temperatures to establish the energy-temperature curve.

The test material used in establishing the unirradiated NDT temperature of the base metal were obtained from (1/4) T (where T is plate thickness) and/or (3/4) T locations of sections of the plate used in the intermediate and lower shell courses. The thermal history of the plate from which the specimens are taken is representative of that of the shell plating. The impact properties at these locations are considered to be representative of the material through the plate. Since the NDT temperature of the material of the plate surface is lower than at (1/4) T, it is conservative to use the properties of (1/4) T to establish the initial minimum operating temperature and as the base for the predicted minimum operating temperature after irradiation.

4.5.3 SURVEILLANCE PROGRAM

The surveillance program monitors the radiation-induced changes in the mechanical and impact properties of the pressure vessel materials. Changes in the impact properties of the material were determined by the comparison of pre- and post-irradiation Charpy impact test specimens. Changes in mechanical properties are determined by the comparison of pre- and post-irradiation data from tensile test specimens.

Three metallurgically different materials are included in the surveillance program. These are base metal, weld metal and heat-affected zone material. Base metal specimens are fabricated from sections of that plate used to form the intermediate shell course of the reactor vessel at the (1/4) T (thickness) and (3/4) T locations. Weld metal and heat-affected zone specimens are obtained by welding material cut from the (1/4) T and (3/4) T locations of the identical plate material used in the core region of the vessel. The welding and inspection procedures employed for fabrication of the pressure vessel were used in the preparation of weld material and heat-affected zone material test specimens. A complete record of the chemical analysis, fabrication history and mechanical properties of these surveillance test materials is maintained.

The test specimens are placed within corrosion-resistant capsules to prevent deterioration of the test specimens during irradiation. The design of the surveillance capsule incorporates features which minimize the temperature differentials between the test specimens and the reactor environment. The capsule size and shape is chosen to minimize neutron flux, thermal and hydraulic perturbations within the surveillance capsules. The capsule design also makes provisions for inclusion of radiation dosimeters, temperature monitors and correlation specimens.

The location of the surveillance capsule assemblies is shown on Figure 4-11. A typical surveillance capsule assembly is shown on Figure 4-12. A typical Charpy impact compartment assembly is shown on Figure 4-13. A typical tensile monitor compartment assembly is shown on Figure 4-14. The exposure locations and a summary of the specimens at each location is presented in Table 4-17.

Fission threshold detectors (U-238) are included in each surveillance capsule to measure the fast neutron flux. Threshold detectors of Ni, Ti, Fe, S and Co-free Cu are included to monitor the fast neutron spectrum.

Selection of threshold detectors was based on the recommendations of ASTM E 261-65T, "Method for Measuring Neutron Flux by Radioactive Techniques." Activation of the specimen material is also analyzed to determine the amount of exposure they received.

Temperature monitors are included to provide an indication of the highest temperature to which the surveillance specimens are exposed, but not the time-temperature history, or the variance between the time-temperature history, of different specimens. These factors, however, affect the accuracy of the estimated vessel RT_{NDT} temperature to only a small extent.

Correlation monitor specimens included in the surveillance capsules (Charpy v-notch specimens machined from ASTM standard reference material) are irradiated along with the surveillance test specimens. The standard reference material was obtained through Subcommittee II of ASTM Committee E-10 on Radioisotopes and Radiation Effects. Use of standard reference material test specimens permits correlation of the postirradiation data obtained in the course of this surveillance program with data obtained from other surveillance programs or irradiation experiments. In addition, changes in impact properties of the correlation monitors provide a cross-check on the neutron dosimetry.

The surveillance capsules are placed in the reactor at three locations. One series of capsules are placed on the outside of the core support barrel to obtain an accelerated exposure. These specimens receive the design lifetime neutron exposure in a relatively short time and provide data for predicting the RT_{NDT} temperature shift for the pressure vessel material over the design life of the vessel.

A second series of specimens are located on the inside of the pressure vessel wall. These specimens receive, at any given time, a slightly higher neutron dose than the pressure vessel. The RT_{NDT} temperature shifts resulting from the irradiation of these specimens closely approximate the RT_{NDT} temperature shift of the vessel materials and serve as a check on the data obtained from the accelerated exposure specimens.

A third series of specimens are located in a low flux region above the core. These specimens are exposed to all reactor temperature cycles but receive a very low neutron dose. Changes in the mechanical and impact properties of the vessel materials due to thermal exposure only can, therefore, be monitored on the basis of changes in properties of these specimens.

The schedule for removal of the surveillance samples is shown in Table 4-20. All surveillance capsules were inserted into their designated holders during the final reactor assembly operation. The capsules remain in the reactor until the desired fluence level has been attained by the specimens.

Test specimens removed from the surveillance capsules are tested in accordance with ASTM Standard Test Methods for Tension and Impact Testing. The data obtained from testing the irradiated specimens are compared with the unirradiated data and an assessment of the neutron embrittlement of the pressure vessel material made. This assessment of the RT_{NDT} temperature shift is based on the temperature shift in the average Charpy curves; the average curves being considered representative of the material.

The periodic analysis of the surveillance samples permits monitoring of the neutron radiation effects upon the vessel materials. If, with due allowance for uncertainties in RT_{NDT} temperature determination, the measured RT_{NDT} temperature shift turns out to be greater than predicted, then appropriate limitations would be imposed on permissible operating pressure-temperature combinations and transients to ensure that the existing reactor vessel stresses are low enough to preclude brittle fracture.

The integrated fast neutron flux to the reactor vessel has been calculated using the methods described in Subsection 3.3.2.6. Assuming an average future cycle capacity factor of 95%, future cycle flux levels comparable with Cycle 21, and an end of License Renewal period date of March 24, 2031, the maximum fast fluence the vessel wall will receive is ~3.429 x 10^{19} [n/cm^2] (References 3, 56, and 57.)

Pursuant to Amendment 34 to the Provisional Operating License DPR-20, capsule A-240 was removed first after 2.26 equivalent full-power years of reactor operations. The test results are contained in reference 15.

Two surveillance capsules were removed from the Palisades reactor vessel during the 1983 outage after 5.20 EFPY (see References 17 and 31). The thermal capsule, T-330, was subject to negligible neutron exposure, while W-290, the wall capsule, was subjected to an exposure closely approximating that of the vessel inner diameter. The mechanical testing of the wall capsule Charpy specimens indicated a very large RT_{NDT} shift of 290°F for the weld metal. Chemistry sampling of the test specimens indicated very high nickel content and a very high variation of nickel content across the specimen thickness. As with the case of the accelerated capsule earlier, the 536°F thermal monitor did not melt. All of these data are contained in Reference 18.

The W-290 data precipitated a review of Combustion Engineering reactor vessel fabrication records as well as Consumers Power Company core physics records. In addition, the 536°F thermal monitor was tested to its melting temperature. As a result of these reviews and tests, it was concluded that:

1. The reactor surveillance weld material was not the MIL-B4 modified wire, heat 27204, as reported to the staff on May 23, 1978, but a RACO wire used with a nickel addition wire.

2. Changes in the core power distribution over the last three fuel cycles had resulted in greater neutron fluence at the vessel wall and the fluences computed by Westinghouse were accurate.

3. The thermal monitors, expected to melt at 536°F, melt at a significantly higher temperature.

Surveillance capsule W-110 was removed from the reactor vessel after 9.95 EFPY. The test results are described in Reference 19. As in prior capsules, the shift in weld metal RT_{NDT} was quite large and the thermal monitors did not melt.

The Palisades reactor vessel surveillance program was designed in accordance with the requirements of ASTM E185-66. If designed today, the surveillance program would include different material than was selected in 1968. Because the material in the surveillance program is not the material that would have been selected if it had been designed in accordance with a more recent edition of the standard, the condition of the reactor vessel is estimated from unirradiated material properties adjusted by generic correlations based on material type, the copper and nickel content of the material, and the best estimate fluence. The limiting reactor vessel beltline material pertaining to the fracture toughness requirements of 10CFR50 Appendix G is plate D-3804-1 located in the lower shell. This plate is projected to drop below the Upper Shelf Energy (USE) lower limit of 50 ft-lb in December 2016 (Reference 55).

During the fall of 1994, CPCo performed material properties tests and chemistry analyses of newly acquired samples of weld material that had been fabricated using the same procedures and weld wire heat number as the limiting weld in the reactor vessel. These material samples were acquired from the shells of the steam generators that had been removed from service at Palisades. These tests and analyses indicated that the degree of embrittlement of the Palisades reactor vessel could be higher than previously calculated. An updated fluence evaluation was performed. Analyses performed in accordance with the PTS rule and accepted by the NRC indicated that the reactor vessel would satisfy the requirements of the PTS rule until approximately 2003 (Reference 45).

At the beginning of Cycle 12, two supplemental surveillance capsules, designated SA-60-1 and SA-240-1, were installed in the capsule holders located on the core support barrel. See Table 4-17 for a listing of the capsule material and test specimens. These capsules contain weld specimens fabricated using similar materials and procedures as those used to fabricate the welds in the limiting portion of the reactor vessel. The capsules also contain standard reference material fabricated from the same plate as the standard reference material included in most reactor vessel surveillance programs. Temperature and flux monitors were also included. Most of the impact specimens in these capsules were modified to increase the number of specimens that could be installed.

Supplemental surveillance capsule SA-60-1 was removed at EOC 13. Testing of the subject materials was performed to determine the irradiated Charpy 30 ft-lb transition temperatures (T_{30}) and upper shelf energies for comparison with testing previously performed on similar materials in the unirradiated condition. Testing is described in DeVan, “Test Results of Capsule SA-60-1 Consumers Energy Palisades Nuclear Plant Reactor Vessel Material Surveillance Program,” BAW-2341, Revision 2, May 2001.

Supplemental surveillance capsule SA-240-1 was removed at EOC 14. Test results are described in DeVan, “Test Results of Capsule SA-240-1 Consumers Energy Palisades Nuclear Plant Reactor Vessel Material Surveillance Program,” BAW-2398, May 2001.

The limiting PTS screening criterion date calculated in Reference 45 utilized inputs that have been updated as documented in WCAP-15353, Revision 0, (Reference 3) and WCAP-15353 Supplement 1-NP (Reference 56). Reference 57 contains the updated PTS evaluation for the reactor vessel. The evaluation determined the date when the most limiting reactor vessel material is projected to reach the PTS screening criterion limit is April 2017. The NRC concluded that the PTS screening criteria will not be reached until April 2017 (Reference 58).

Capsule W-100 was removed at EOC 16. Test results are described in BWXT Report, “Analysis of Capsule W-100 from the Nuclear Management Company Palisades Reactor Vessel Material Surveillance Program,” February 2004 (Reference 51).

4.5.4 NONDESTRUCTIVE TESTS

Prior to and during fabrication of the reactor vessel, nondestructive tests based upon the ASME B&PV Code, Section III, 1965, W65a, were performed on all welds, forgings and plates as follows:

All full penetration pressure containing welds were 100% radiographed to the standards of the ASME B&PV Code, Section III, Subparagraph N-624.8, 1965, W65a. Other pressure containing welds such as used for the attachment of mechanism housings, vents and instrument housings to the reactor vessel head were inspected by liquid penetrant tests of the root pass, each 1/2 inch of weld deposit and the final surface.

All forgings were inspected by ultrasonic testing, using longitudinal beam techniques. In addition, ring forgings were tested using shear wave techniques. Rejection under longitudinal beam inspection, with calibration so that the first back reflection is at least 75% of screen height, was based on interpretation of indications causing complete loss of back reflection. Rejection under shear wave inspection was based on indications, exceeding in amplitude the indication from a calibration notch whose depth is 3% of the forging thickness, not exceeding 3/8 inch with a length of 1 inch.

All forgings were also subjected to magnetic particle examination. Rejection was based on relevant indications of:

1. Any cracks and linear indications

2. Rounded indications with dimensions greater than 3/16 inch

Plates were ultrasonically tested using longitudinal and shear wave ultrasonic testing techniques. Rejection under longitudinal beam testing performed in accordance with ASME B&PV Code, Section II, SA-435, 1965, with calibration so that the first back reflection is at least 50% of screen height, was based on defects causing complete loss of back reflection. Any defect which showed a total loss of back reflection which could not be contained within a circle whose diameter is the greater of 3 inches or one-half the plate thickness was unacceptable. Two or more defects smaller than described above which caused a complete loss of back reflection were unacceptable unless separated by a minimum distance equal to the greatest diameter of the larger defect unless the defects were contained within the area described above. For shear wave testing, the maximum permissible flaw was one which did not exceed that from a calibrated notch having a depth of 3% of the plate thickness and 1 inch long.

Nondestructive testing of the vessel was performed during several stages of fabrication with strict quality control in critical areas such as constant calibration of test instruments, metallurgical inspection of all weld rod and wire, and strict adherence to the nondestructive testing requirements of the ASME B&PV Code, Section III, Class A, 1965, W65a.

The detection of flaws in irregular geometries was facilitated because most nondestructive testing of the materials was completed while the material is in its simplest form. Nondestructive inspection during fabrication was scheduled so that full penetration welds are capable of being radiographed to the extent required by ASME B&PV Code, Section III, Class A, 1965, W65a.

Each of the vessel studs received two ultrasonic tests and one magnetic particle inspection during the manufacturing process.

The first ultrasonic test was a radial longitudinal beam inspection and the standard for rejection was 100% loss of back reflection or an indication which reduced the adjusted back reflection by greater than 20%. The second ultrasonic test was a radial inspection using the angle beam technique with the rejection standard the same as for forgings. The use of these techniques insured that only such materials were accepted that have flaws no greater than 1/2 inch and no observable cracks or sharply defined linear defects.

Magnetic particle inspection was performed on the finished studs. Axially aligned defects whose depths are greater than thread depth and nonaxial defects were unacceptable.

The vessel closure contains 54 studs, 7 inches in diameter with 8 threads per inch. The stud material is ASTM A 540-65, Grade B24, with a minimum yield strength of 130,000 psi. The tensile stress in each stud when elongated for operational conditions is approximately 40 ksi. Calculations show that 32 uniformly distributed studs can fail before the closure will separate at design pressure. However, 16 uniformly distributed broken studs or 4 adjacent broken studs will cause O-ring leakage. Failure of at least 16 adjacent studs is necessary before the closure will fail by "zippering" open.

The vessel studs are stressed as they are elongated by the stud tensioners during the initial installation of the vessel head and at each refueling. The amount of elongation versus hydraulic pressure on the tensioner will be compared with previous readings to detect any significant changes in the elongation properties of the studs. Studs which yield questionable data during the head installation, or receive damage to the threads, will be replaced before returning the vessel to pressure operations.

Table 4-18 summarizes the inspection program by component.

4.5.5 ADDITIONAL TESTS

During design and fabrication of the reactor vessel, a number of operations over and above the requirements of the ASME B&PV Code, Section III, Class A, 1965, W65a, were performed by the vendor. Table 4-19 summarizes the additional tests by component.

During the design of the reactor vessel, detailed calculations were performed to assure that the final product would have adequate design margins. The design adequacy was established by stress concentration factors which have been obtained through the use of photoelastic models for areas which are not amenable to calculation. A detailed fatigue analysis of the vessel for all design conditions has been performed. In addition, Combustion Engineering has performed test programs for the determination, solution and verification of analytical solutions to thermal stress problems. Also, fracture mechanics and brittle fracture evaluations have been performed.

All material used in the reactor vessel was carefully selected and precautions were taken by the vessel fabricator to ensure that all material specifications were adhered to. To assure compliance, the quality control staff of Combustion Engineering reviewed the mill test reports and the fabricator's testing procedures.

All welding methods, materials, techniques and inspections comply with ASME B&PV Code, Sections III and IX, 1965, W65a. Before fabrication was begun, detailed qualified welding procedures including methods of joint preparation, together with certified procedure qualification test reports, were prepared. Also, prior to fabrication, certified performance qualification tests were obtained for each welder and welding operator. Quality control was exercised for all welding rod and wire by subjection to a complete and thorough testing program in order to insure maximum quality of welded joints.

During the manufacture of the reactor vessel, quality control by the vendor, in addition to and in areas not covered by the ASME B&PV Code, Section III, 1965, W65a, included: Preparation of detailed purchase specifications which included cooling rates for test samples; requiring vacuum degassing for all ferritic plates and forgings; specification of fabrication instructions for plates and forgings to provide control of material prior to receipt and during fabrication; use of written instructions and manufacturing procedures which enabled continual review based on past and current manufacturing experiences; performance of chemical analysis of welding electrodes, welding wire and materials for automatic welding, thereby providing continuous control over welding materials; the determination of NDT temperature through use of drop weight testing methods and test programs on fabrication of plates up to 15 inches thick to provide information about material properties as thickness increases. Shear wave and longitudinal wave ultrasonic testing was performed on 100% of all plate material.

Cladding for the reactor vessel was a continuous integral surface of corrosion-resistant material, 1/4-inch nominal thickness. The detailed procedure used; ie, type of weld rod, welding position, speed of welding, nondestructive testing requirements, etc, was in compliance with the ASME B&PV Code, Sections III and IX, 1965, W65a.

Combustion Engineering has checked the cladding on completed reactor vessels and such tests have not shown the need for 100% ultrasonic testing for weld deposited cladding after fabrication. The clad surface is ultrasonically inspected transverse to the direction of welding for lack of bond at intervals of 12 inches or 1.4 times the base metal thickness, whichever is less.

Upon completion of all postweld heat treatments, the reactor vessel was hydrostatically tested, after which all weld surfaces, including those of welds used to repair material, were magnetic particle inspected in accordance with ASME B&PV Code, Section III, Paragraph N-618, 1965, W65a.

Quality control by the licensee was also carried out during the manufacture of the vessel by a resident inspector. This work included independent review of all radiographs, magnetic particle tests, ultrasonic tests and dye penetrant tests conducted during the manufacture of the vessel. A review of material certifications, and vendor manufacturing and testing procedures was also conducted. This review included all manufacturers' records such as heat treat logs, personnel qualification files and deviation files.

4.5.6 INSERVICE INSPECTION

Provision was made in the design to permit inservice inspection as may be required. The location of the more highly stressed portions of the reactor vessel was identified. These areas are equipped with removable insulation and portions may be inspected at various intervals, utilizing appropriate nondestructive testing techniques. In addition, the inside of the reactor vessel and the internals may be subjected to routine visual inspection during refueling outages. An inspection of accessible areas of the reactor vessel and internals, with a television camera or other suitable means, may be accomplished at any time when the reactor core is completely unloaded. The design permits all vessel internals except the flow skirt to be removed so that a complete internal vessel visual inspection would be possible. During refueling outages, the reactor vessel head and the closure sealing surfaces may be visually inspected. The internal parts of the vessel which are visible, including the cladding and components, may also be visually checked, as well as the accessible external surface of the vessel, nozzles and the vessel studs.

A combination of ultrasonic, dye penetrant, magnetic particle and visual inspections will be used to conduct the inspections. The planned inspection program takes into account the mechanisms which may lead to failure in the Primary Coolant System. Emphasis has been placed on the expected high stress areas as determined by a design evaluation and experience.

The major premises of this inspection program are:

1. Selected areas of expected maximum stress will be inspected at intervals in accordance with ASME B&PV Code Section XI, except as adjusted by NRC-approved code cases and relief requests. These inspections will serve to indicate potential problems before significant flaws develop there or at other areas.

2. If flaw initiation or growth is detected in one of the selected areas, all comparable areas in the primary system will be inspected.

3. Regardless of the results of inspection of the selected areas, all major discontinuity areas in the primary system will be inspected within a ten-year period, except as adjusted by NRC-approved code cases and relief requests.

4. A surveillance program will determine the shift in reference temperature (RT_{NDT}) of the vessel in the core region due to irradiation. The vessel will not be fully pressurized below the RT_{NDT}.

Refer to Section 6.9 for a detailed description of the inservice inspection program. A summary of the inspection program is as follows:

1. Areas of expected maximum stress selected for periodic inspections are:

    a. The flange-to-shell weld of the vessel

    b. The flange-to-torus weld of the vessel head

    c. The primary coolant outlet nozzle-to-shell welds and nozzle-shell radii on the vessel ID

    d. The dissimilar welds between the primary coolant piping and pumps

    e. Longitudinal and circumferential welds in the primary coolant piping

    f. Branch piping connections to the primary coolant piping

    g. The tube sheet-to-head weld of the steam generators

h. The nozzle-to-head welds in the lower head of the steam generators

i. The internal support stand welds in the lower head of the steam generators

j. The support stand to steam generator head

2. Areas to be inspected within a ten-year period, except as adjusted by NRC-approved code cases and relief requests, are:

   a. The nozzle-to-shell welds of primary coolant nozzles not inspected on a more frequent basis

   b. The longitudinal and circumferential welds in the core region of the reactor pressure vessel

   c. The outlet nozzle-to-shell weld of the pressurizer

   d. The longitudinal weld in the pressurizer

Ultrasonic inspection of components provides indications from discontinuities, impedance mismatches (such as a junction between Inconel weld metal and carbon steel) and from changes in component geometry. Baseline data to assist in interpretation of future inspection results will be acquired from a preservice inspection and pertinent shop data.

If indications of defect initiation or growth are noted, the program will be reviewed and sufficient inspections performed to determine that defects are not being initiated or propagated in other areas of the pressure vessel or components.

The bases for the above inspection points and the frequency of inspection are the result of a review of design drawings, the test results available from the PVRC vessel test program conducted at Southwest Research Institute, the present knowledge available on the mechanics of failure of such systems, ASME Section XI Code requirements, and NRC-approved code cases and relief requests. They are also based on the fact that the component fabricator for this Plant has a history of successful vessel fabrication in accordance with the practices of the ASME B&PV Code and more restrictive self-imposed specifications. Code manufacturing procedures and inspection techniques precluded the initial presence of large flaws in the vessel. Therefore, it is believed that the most likely location of a failure would be at a point of expected maximum stress concentration and not at some random location.

Thus, high stress locations are selected for monitoring of initiation of flaws. Furthermore, with baseline ultrasonic readings obtained on the pressure vessel and other inspection points in the reactor primary system, added assurance is attained that no significant flaws exist in the pressure boundary components of the Primary Coolant System.

Additional inservice inspection requirements have been established in Technical Specifications to address augmented steam generator tube inspection and inspection of primary coolant pump flywheels.

During the 2004 Refueling Outage, inspections of the reactor head required by NRC Order EA-03-009 (Reference 52) resulted in the need to repair two of the CRD nozzles. The NRC Order required that, if repairs are necessary, visual and volumetric inspections be completed during each subsequent refueling outage.

Palisades was granted relief in a NRC Safety Evaluation dated February 11, 2009 (Reference 59) to extend the third inservice inspection interval for reactor vessel weld examinations until December 12, 2015.

4.5.7 NDTT OF OTHER PRIMARY SYSTEM COMPONENTS

The impact properties of all carbon steel and alloy steel materials which form a part of the pressure boundary of the Primary Coolant System were determined in accordance with the requirements of the ASME B&PV Code, Section III, Paragraph N-330, 1965, W65a. The materials were required to pass the acceptance test noted in Paragraph N-330 at 40°F, although it was an objective that the materials meet this requirement at 10°F. The operating stress limits for these materials in the Primary Coolant System other than the reactor vessel will be the same as those for the reactor vessel. Shortly after Plant start-up, the integrated neutron flux will result in the reactor vessel being the controlling component.

4.5.8 NONDESTRUCTIVE TESTS OF OTHER PRIMARY SYSTEM COMPONENTS

Prior to and during fabrication of the original components of the Primary Coolant System, nondestructive testing based upon the requirements of the ASME B&PV Code, Section III, Class A, 1965, W65a, was used to determine the acceptance criteria for various size flaws. The requirements for the Class A vessels are the same as the reactor vessel. Vessels designated as Class C were fabricated to the standards of the ASME B&PV Code, Section III, Article 21, 1965, W65a. Requirements for replacement parts and components are as specified in Section 4.2.1.