Zion Station

Updated Final Safety Analysis Report

Volume 3

Commonwealth Edison Company

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4. REACTOR

4.1 SUMMARY DESCRIPTION

The principal function of any nuclear power reactor is to generate heat at a demanded rate. One way of accomplishing this function is to design a reactor core consisting mainly of the Zircaloy clad, slightly enriched UO_2 fuel arranged in canless assemblies, various internal structures, reactivity control components and monitoring instrumentation, and then to accommodate this core in a reactor vessel filled with the pressurized light water which acts both as moderator and coolant. The reactor under consideration is designed to operate at the licensed power rating of 3250 MWt with sufficient margins to allow for transient operation and instrument error without causing damage to the core and without exceeding the pressure of the safety valve settings in the coolant system.

This chapter describes various aspects of the design of the reactor constituents. It presents supporting information and evaluations of that design demonstrating that it will perform intended functions throughout its lifetime under all normal operational modes, including both steady state and transients, without releasing unacceptable amounts of fission products to the coolant. In addition, this chapter incorporates a study of the reactor's potential to act as a source of abnormal conditions and provides supporting information for the analyses discussed in Chapter 15.

The reactor core is a multiple region cycled core. It consists of 193 fuel assemblies with 204 individual fuel rods per assembly except that limited substitution of rods by filler rods may be made if justified by a cycle specific analysis. The fuel rods are cold worked, partially annealed Zircaloy tubes containing slightly enriched uranium dioxide fuel. To reduce the axial leakage, the top and bottom six inches of the VANTAGE 5 fuel rod may contain natural or low enriched uranium pellets. In addition, Integral Fuel Burnable Absorber (IFBA) rods may also be utilized. The IFBA rod contains coated fuel pellets identical to the enriched uranium dioxide pellets except for the addition of a thin boride coating less than 0.001 inch in thickness on the pellets' cylindrical surface. Coated pellets occupy the central region of the fuel column. The number and pattern of IFBA rods within an assembly may vary depending on the specific application such as predictable power distribution and moderator temperature control.

All fuel rods are pressurized with helium during fabrication to reduce stresses and strains and to increase fatigue life.

The fuel assembly is a canless type, with the basic assembly consisting of the rod cluster control guide thimbles mechanically attached to the grids and the top and bottom nozzles. The fuel rods are supported at several points along their length by the spring-clip grids. There are four types of fuel assemblies used by Zion: the initial and early reloads were Low Parasitic (LOPAR) and later cycles were transitioned to Optimized Fuel Assemblies (OFA), VANTAGE 5, and VANTAGE 5 with Intermediate Flow Mixer (IFM) designs. The differences between these designs are discussed in more detail in References 1 and 9 of Section 4.2.

Rod cluster control assemblies (RCCAs), secondary sources, and burnable absorber rods, if required, are inserted into the guide thimbles of the fuel assemblies. The absorber sections of the control rods are fabricated of a silver-indium-cadmium alloy sealed in stainless steel tubes. The absorber material in the discrete burnable absorber rods is in the form of an aluminum oxide - boron carbide annulus sealed in Zircaloy called a Wet Annular Burnable Absorber (WABA).

The control rod drive mechanisms for the RCCAs are of the magnetic latch type. The latches are controlled by three magnetic coils. They are designed so that upon a loss of power to the coils, the rod cluster control assembly is released and falls by gravity into the core to shut down the reactor.

The reactor is capable of meeting the performance objectives throughout core life under both steady state and transient conditions without violating the integrity of the fuel elements. Thus, the release of unacceptable amounts of fission products to the coolant is prevented.

The limiting conditions for operation established in the Technical Specifications specify the functional capacity of performance levels permitted to assure safe operation of the facility.

Sections 4.2 through 4.5 specify design parameters which are pertinent to safety limits for the nuclear, control, thermal and hydraulic, and mechanical aspects of the design.

4.2 FUEL SYSTEM DESIGN

4.2.1 Design Basis

4.2.1.1 Performance Objectives

The licensed reactor thermal power rating is 3250 MWt. Hot channel factors, as calculated from measured power distributions, are considerably less than those used for design purposes.

The turbine-generator and plant heat removal systems have been designed for a thermal rating of 3391 MWt. Containment and Engineered Safeguards are designed and evaluated for operation at the power rating of 3391 MWt. Most postulated accidents including the loss-of-coolant accident (LOCA) are evaluated at 102% of 3250 MWt (3315 MWt).

The reactor core fuel loading and duty are designed for extended burnup as described in the extended burnup topical report (see Reference 10). The fuel rod cladding is designed to maintain its integrity for the anticipated core life. The effects of gas release, fuel dimensional changes, and corrosion-induced or irradiation-induced changes in the mechanical properties of cladding are considered in the design of the fuel assemblies.

Rod Cluster Control Assemblies (RCCAs) are employed to provide sufficient reactivity control to terminate any credible power transient prior to reaching the design minimum departure from nucleate boiling (DNB) ratio (see Section 4.4). This is accomplished by ensuring sufficient control cluster worth to shut the reactor down by at least 1.3% in the hot condition with the most reactive control cluster stuck in the fully withdrawn position.

Redundant equipment is provided to add soluble poison to the reactor coolant in the form of boric acid to maintain shutdown margin when the reactor is cooled to ambient temperatures.

The RCCAs provide the capability of meeting the minimum required shutdown margin which corresponds to the shutdown assumed in the steam break analysis discussed in Chapter 15.

4.2.1.2 Fuel Assemblies

The fuel assemblies are designed to perform satisfactorily throughout their lifetime. The loads, stresses, and strains resulting from the combined effects of flow-induced vibrations, LOCA blowdown coincident with earthquake, reactor pressure, fission gas pressure, fuel growth, thermal strain, and differential expansion during both steady state and transient reactor operating conditions have been considered in the design of the fuel rods and fuel assemblies. The assemblies are also structurally designed to withstand handling and shipping loads prior to irradiation, and to maintain sufficient integrity at the completion of design burnup to permit safe removal from the core subsequent to handling during cooldown, shipment, and fuel reprocessing.

The fuel rods are supported at seven locations along their length within the fuel assemblies by grid assemblies which are designed to maintain control of the lateral spacing between the rods throughout the design life of the assemblies. The magnitude of the support loads provided by the grids are established to minimize possible fretting without overstressing the cladding at the points of contact between the grids and fuel rods and without imposing restraints of sufficient magnitude to result in buckling or distortion of the rods.

The fuel rod cladding is designed to withstand operating pressure loads without rupture and to maintain encapsulation of the fuel throughout the design life.

4.2.1.3 Rod Cluster Control Assemblies

The criteria used for the design of the cladding on the individual absorber rods in each RCCA are similar to those used for the fuel rod cladding. The cladding is designed to be free standing under all operating conditions and will maintain encapsulation of the absorber material throughout the absorber rod design life. Allowance for wear during operation is included for the RCCA cladding thickness.

Adequate clearance is provided between the absorber rods and the guide thimbles which position the rods within the fuel assemblies so that coolant flow along the length of the absorber rods is sufficient to remove the heat generated without overheating of the absorber cladding. The clearance is also sufficient to compensate for any misalignment between the absorber rods and guide thimbles and to prevent mechanical interference between the rods and guide thimbles under any operating conditions.

4.2.2 Description and Design Drawings

This subsection title has been created in order to implement the UFSAR format delineated by Regulatory Guide 1.70, Revision 3. However, the section is not used due to the level of detail required at the time of license application and subsequent revisions.

4.2.3 Design Evaluation

Experimental measurements from critical experiments or operating reactors, or both, are used to validate the methods employed in the design. At the time of initial design, nuclear parameters were calculated for every phase of operation and were compared with design limits to show that an adequate margin of safety existed. Subsequent core designs have been evaluated to demonstrate that they have not adversely affected the safety of the plant. Those accidents analyzed (Chapter 15), which may be affected by fuel reload, are reviewed and the nuclear design parameters are checked to determine if they remain within current design limits. If the parameters are outside of the current limits, the accidents affected by the nuclear parameters are re-evaluated or re-analyzed for the more limiting nuclear parameter. Reload design evaluations are in accordance with Reference 11.

4.2.3.1 Cladding

The integrity of the fuel rod cladding, so as to retain fission product or fuel material, is directly related to cladding stress and strain under normal operating and overpower conditions. The cladding stress is limited to the yield strength of the cladding material and the steady-state tensile strain is limited to 1.0%.

The cladding stresses at constant local fuel rod power are low. Compressive stresses are created by the pressure differential between the coolant pressure and the rod internal pressure. Tensile stresses could be created once the cladding has come in contact with the pellet. These stresses would be induced by the fuel pellet swelling during irradiation. Fuel swelling can result in small cladding strains (<1%) for expected discharge burnups, but the associated cladding stresses are low because of cladding creep (thermal and irradiation induced creep). Furthermore, the 1% strain criterion is extremely conservative for fuel swelling driven cladding strain because the strain rate associated with solid fission product swelling is very slow. In-pile experiments (Reference 12) have shown that Zircaloy tubing exhibits "superplasticity" at slow strain rates during neutron irradiation. Uniform cladding strains of >10% have been achieved under these conditions with no sign of plastic instability.

Pellet thermal expansion caused by power increases is considered the one mechanism by which significant stresses and strains can be imposed on the cladding. Power increases in the reactor can result from fuel shuffling, reactor power escalation following extended reduced power operation, and full length control rod movement. In the mechanical design model, lead rods are depleted using best estimate rod power histories as determined from core physics calculations. During the depletion, the amount of diametral gap closure is evaluated based upon the pellet densification and swelling model, cladding creep model, and fission gas release model (References 12 and 13). At various times during the depletion, the power is increased locally on the rod to burnup-dependent attainable power density, and axial cladding stresses resulting from the power increase are combined into a volume average effective cladding stress. The von Mises criterion is used to evaluate whether the cladding yield strength has been exceeded. The yield strength correlation is for irradiated cladding since fuel cladding contact occurs at high burnup. The yield strength is also a function of the cladding temperature. Slow transient power increases can result in large cladding strains without exceeding the cladding yield strength because of cladding creep and stress relaxation. Therefore, in addition to the yield stress criterion, a criterion on allowable cladding tensile strain was set at 1% based upon high strain rate burst and tensile test data on irradiated tubing.

The internal gas pressure contributes to the cladding stresses. The maximum rod internal gas pressure is limited to a value below which could cause the diametral gap to increase due to outward cladding creep during steady-state operation and below that which could cause extensive DNB propagation to occur. The maximum rod internal pressue occurs at end-of-life and is dependent upon the initial pressure, void volume, and fuel rod power history.

Section 4.2.3.2.1 discusses the effects of fuel densification on fuel cladding.

4.2.3.2 <u>Fuel</u>

The fuel pellets are right circular cylinders consisting of slightly enriched uranium-dioxide powder which has been compacted by cold pressing and then sintered to the required density. The ends of each pellet have a small chamfer at the cylindrical surface and are dished slightly to allow the greater axial expansion at the center of the pellets to be taken up within the pellets themselves and not in the overall fuel length.

The VANTAGE 5 and the OFA fuel rods have the same clad wall thickness and outer diameters. Also, the bottom end plug has an internal grip feature to facilitate rod loading on both designs. The VANTAGE 5 fuel rod length is larger by 0.320 inch to provide for a longer plenum and bottom end plug. The bottom end plug is longer to provide a longer lead-in for the removable top nozzle reconstitution feature. The VANTAGE 5 also has axial blankets and Integral Fuel Burnable Absorber (IFBA) features.

The axial blankets are a nominal 6 inches of unenriched fuel pellets at each end of the fuel rod pellet stack. Axial blankets reduce neutron leakage and improve fuel utilization. The axial blankets utilize chamfered pellets which are physically different (length) than the enriched pellets to help prevent accidental mixing during manufacturing.

The IFBA coated fuel pellets are identical to the enriched uranium dioxide pellets except for the addition of a thin zirconium diboride (ZrB_2) coating on the pellet cylindrical surface. Coated pellets occupy the central portion of the fuel column (up to 134 inches). The number and pattern of IFBA rods within an assembly may vary depending on the specific application. The ends of the IFBA enriched coated pellets, like the enriched uncoated pellets, are also dished to allow for greater axial expansion at the pellet centerline and void volume for fission gas release. An evaluation and test program for the IFBA design features is given in Addendum 1 (15x15 application) of Reference 14.

Void volume and clearances are provided within the rods to accommodate fission gases released from the fuel, differential thermal expansion between the cladding and the fuel, and fuel density changes during irradiation, thus avoiding overstressing of the cladding or seal welds. Shifting of the fuel within the cladding during handling or shipping prior to core loading is prevented by a stainless steel helical spring which bears on top of the fuel.

As shown in Table 4.2-1, a different fuel enrichment was used for each of the three regions in the Cycle 1 core loading.

In the event of cladding defects, the high resistance of uranium dioxide fuel pellets to attack by hot water protects against fuel deterioration or decrease in fuel integrity. Thermal stress in the pellets, while causing some fracture of the bulk material during temperature cycling, does not result in pulverization or gross void formation in the fuel matrix. As shown by operating experience and extensive experimental work in the industry, the thermal design parameters conservatively account for any changes in the thermal performance of the fuel element due to pellet fracture.

The consequences of a breach of cladding are greatly reduced by the ability of uranium dioxide to retain fission products including those which are gaseous or highly volatile. This retentiveness decreases with increasing temperature and fuel burnup, but remains a significant factor even at full power operating temperature in the maximum burnup element.

A survey of high burnup uranium dioxide (see Reference 2) fuel element behavior indicates that, for an initial uranium dioxide void volume, which is a function of the fuel density, it is possible to conservatively define the fuel swelling as a function of burnup. The fuel swelling model considers the effect of burnup, temperature distribution, and internal voids. It is an empirical model which has been checked with data from Bettis, Yankee, CVTR, Saxton, and others. The pellet densities for the three regions are listed in Table 4.2-1.

4.2.3.2.1 Fuel Densification

The evaluation of fuel densification effects and the treatment of fuel swelling and fission gas release are described in References 12 and 13.

4.2.3.2.2 Reload Fuel

The procurement of reload core regions from a vendor other than the original NSSS vendor would normally be done by competitive bidding based on a fuel specification issued by Commonwealth Edison Company. Typically, the specification would call for one or more reload regions and fuel management services.

Qualification of such fuel would occur in several stages. The first stage would generally be the screening and acceptance of the contractors allowed to bid on the fuel specification. The basic criteria employed at this stage are that the vendor has demonstrated the capabilities needed to meet the performance requirements specified through past performance of other reload fuel or that the vendor demonstrate by detailed testing and analytical methods that he has such capability. A second criterion is that the vendor has or can show that he will have an approved quality assurance program compatible with 10CFR50 Appendix B.

A successful bidder (the bidder awarded a contract) will be required to furnish fuel designed and fabricated in accordance with the approved quality assurance program. The fuel will be required to be compatible with the nuclear steam supply system and the in-place fuel from the standpoints of mechanical, thermal-hydraulic, and nuclear design. The fuel furnished will be required to be compatible with the existing safety evaluations, or will be required to be such that reanalysis of those safety evaluations would not show undue risk to the public health and safety as defined in 10CFR20 and 10CFR100.

4.2.3.3 Fuel Assemblies

There are four types of fuel assemblies used at Zion: the initial and early reloads were Low Parasitic (LOPAR) and later cycles were transitioned to Optimized Fuel Assemblies (OFA), VANTAGE 5, and VANTAGE 5 with Intermediate Flow Mixer (IFM) designs. The differences between these designs are discussed in more detail in References 1 and 9. The overall configuration of the fuel assemblies are shown in Figures 4.2-1, 4.2-2, 4.2-8, and 4.2-10. The assemblies are square in cross-section, nominally 8.426 inches on a side. The LOPAR, OFA, and VANTAGE 5 designs have an overall length of 159.710 inches, 159.765 inches, and 159.975 inches, respectively. Typical mechanical design parameters for the LOPAR fuel assembly design is given in Table 4.2-1. A comparison of the fuel assembly design parameters for the OFA, VANTAGE 5, and VANTAGE 5 with IFM designs are given in Table 4.2-2.

The fuel rods in a fuel assembly are arranged in a square array with 15 rod locations per side and a nominal centerline-to-centerline pitch of 0.563 inch between rods. Of the total possible 225 rod locations per assembly, 20 are occupied by guide thimbles for the RCCA rods and one for incore instrumentation. The remaining 204 locations contain fuel rods. In addition to fuel rods, a fuel assembly is composed of a top nozzle, a bottom nozzle, 7 grid assemblies, 20 rod guide thimbles, and one instrumentation thimble. The VANTAGE 5 with IFM fuel assembly design has three IFM grids.

The guide thimbles in conjunction with the grid assemblies and the top and bottom nozzles comprise the basic structural fuel assembly skeleton. The top and bottom ends of the guide thimbles are secured to the top and bottom nozzles respectively. The grid assemblies, in turn, are mechanically attached to the guide thimbles at each location along the height of the fuel assembly at which lateral support for the fuel rods is required. Within this skeletal framework the fuel rods are contained and supported and the rod-to-rod centerline spacing is maintained along the assembly.

4.2.3.3.1 Bottom Nozzle

The bottom nozzle is a square box-like structure which controls the coolant flow distribution to the fuel assembly and functions as the bottom structural element of the fuel assembly. The nozzle, which is square in cross-section, is fabricated from Type 304 stainless steel parts consisting of a perforated adaptor plate, four angle legs, and four pads or feet. A skirt around the bottom nozzle perforated adaptor plate is included for the VANTAGE 5 design to increase structural capability for abnormal load conditions. The angle legs and skirt, if applicable, are fastened to the plate forming a plenum space for coolant inlet to the fuel assembly. The perforated adaptor plate also prevents accidental downward ejection of the fuel rods from the fuel assembly. The bottom nozzle is fastened to the fuel assembly guide tubes by stainless steel screws which penetrate through the nozzle and mate with a threaded plug in each guide tube. The screw is prevented from loosening by a stainless steel lock pin welded to the bottom nozzle for the LOPAR and OFA designs. The reconstitutable bottom nozzle for the VANTAGE 5 designs incorporates a thimble screw with a circular locking cup around the screw head. The locking cup is crimped into mating detents (lobes) on the bottom nozzle. A comparison of the thimble screw locking features is shown in Figure 4.2-11.

The VANTAGE 5 designs incorporate the Debris Filter Bottom Nozzle (DFBN) concept to reduce the possibility of fuel rod damage due to debris-induced fretting. The relatively large flow holes in the conventional nozzle used in the LOPAR and OFA designs are replaced with a new pattern of smaller flow holes. The flow holes are sized to minimize passage of debris particles large enough to cause damage while providing sufficient flow area, comparable pressure drop, and continued structural integrity of the nozzle. A comparison of the standard and debris filter bottom nozzles are shown in Figure 4.2-12.

The perforated adaptor plate served as the bottom end support for the fuel rods in Region 1 fuel. All subsequent Regions are built with at least a $^1/_2$ inch gap between the fuel rods and bottom plate. The bottom support surface for the fuel assembly is formed under the plenum space by the four pads which are welded to the corner angles.

Coolant flow to the fuel assembly is directed from the plenum in the bottom nozzle upward to the interior of the fuel assembly and to the channel between assemblies. The perforations in the bottom adaptor plate are positioned and are sized so that the fuel rods cannot pass through them.

The RCCA guide thimbles, which carry axial loads imposed on the assembly, are fastened to the bottom nozzle perforated adaptor plate. These loads, as well as the weight of the assembly, are distributed through the nozzle to the lower core support plate. Indexing and positioning of the fuel assembly in the core is controlled through two holes in diagonally opposite pads which mate with locating pins in the lower core plate. Lateral loads imposed on the fuel assembly are also transferred to the core support structures through the locating pins.

4.2.3.3.2 Top Nozzle

The top nozzle is a box-like structure which functions as the fuel assembly upper structural element and forms a plenum space where the heated fuel assembly discharge coolant is mixed and directed toward the flow holes in the upper core plate. The nozzle is comprised of an adaptor plate enclosure, top plate, two clamps, four leaf springs, and assorted hardware. All parts, with the exception of the springs and their hold down screws, are constructed of Type 304 stainless steel. The springs are made from age hardenable Inconel 718 and the spring screws from Inconel 600.

The adaptor plate is square in cross-section, and is perforated by machined slots to provide for coolant flow through the plate. At assembly, the top holes in the plate are fastened to thimble sleeves through individual bored holes in the plate and welded to the plate around the circumference of each hole for the LOPAR and OFA designs. A reconstitutable top nozzle is incorporated into the VANTAGE 5 designs. In the VANTAGE 5 reconstitutable top nozzle designs, a stainless steel nozzle insert is mechanically connected to the top nozzle adaptor plate by means of a preformed circumferential bulge near the top of the insert. The insert engages a mating groove in the wall of the adaptor plate thimble tube through hole. The insert has four equally spaced axial slots which allow the insert to deflect inwardly at the elevation of the bulge, thus permitting the installation and removal of the nozzle. The insert bulge is positively held in the adaptor plate mating groove by placing a lock tube with a uniform inside diameter identical to that of the thimble tube into the insert. The lock tube is secured in place by a top flare which creates a tight fit and six non-yielding projections on the outside diameter which interface with the concave side of the insert to preclude escape during core component transfer. A schematic of the reconstitutable top nozzle joint is shown in Figure 4.2-13. Thus, the adaptor plate acts as the fuel assembly top end plate and provides a means of evenly distributing among the guide thimbles any axial loads imposed on the fuel assemblies.

The nozzle enclosure is actually a square thin-walled tubular shell which forms the plenum section of the top nozzle. The bottom end of the enclosure is pinned and welded to the periphery of the adaptor plate, and the top end is welded to the periphery of the top plate.

The top plate is square in cross-section with a square central hole. The hole allows clearance for the RCCA absorber rods to pass through the nozzle into the guide thimbles in the fuel assembly and for coolant exiting from the fuel assembly to the upper internals area. Two pads containing axial through-holes, which are located on diametrically opposite corners of the top plate, provide a means of positioning and aligning the top of the fuel assembly. As with the bottom nozzle, alignment pins in the upper core plate mate with the holes in the top nozzle plate.

Hold-down forces of sufficient magnitude to oppose the hydraulic lifting forces on the fuel assembly are obtained by means of the leaf springs which are mounted on the top plate. The springs are fastened in pairs to the top plate at the two corners where alignment holes are not used and radiate out from the corners parallel to the sides of the plate. Fastening of each pair of springs is accomplished with a clamp which fits over the ends of the springs and two screws (one per spring) which pass through the clamp and spring and thread into the top plate. At assembly, the spring mounting screws are torqued sufficiently to preload against the maximum spring load and then lockwelded to the clamp which is counter-bored to receive the screw head.

The spring load is obtained through deflection of the spring by the upper core plate. The spring form is such that it projects above the fuel assembly and is depressed by the core plate when the internals are loaded into the reactor. The free end of the spring is bent downward and captured in a key slot in the top plate to guard against loose parts in the reactor in the event (however remote) of spring fracture. In addition, the fit between the spring and key slot and between the spring and its mating slot in the clamp are sized to prevent rotation of either end of the spring into the control rod path in the event of spring fracture.

In addition to its plenum and structural functions, the nozzle provides a protective housing for components which mate with the fuel assembly. In handling a fuel assembly with a control rod inserted, the control rod spider is contained within the nozzle. During operation in the reactor, the nozzle protects the absorber rods from coolant cross flows in the unsupported span between the fuel assembly adaptor plate and the end of the guide tube in the upper internals package. Plugging devices which fill the ends of the fuel assembly thimble tubes at unrodded core locations and the spiders which support the source rods and burnable absorber rods are all contained within the fuel top nozzle.

4.2.3.3.3 Guide Thimbles

The control rod guide thimbles in the fuel assembly provide guided channels for the absorber rods during insertion and withdrawal of the control rods. They are fabricated from a single piece of Zircaloy-4 tubing, which is drawn to two different diameters. The larger inside diameter at the top (0.512 inch for LOPAR and 0.499 inch for OFA and VANTAGE 5 designs) provides a relatively large annular area for rapid insertion during a reactor trip and accommodates a small amount of upward cooling flow during normal operations. The bottom portion of the guide thimble is of reduced diameter (0.455 inch) to produce a dashpot action when the absorber rods near the end of travel in the guide thimbles during a reactor trip. The transition zone at the dashpot section is conical in shape so that there are no rapid changes in diameter in the tube. The guide thimble diameter and dashpot length for the OFA and VANTAGE 5 designs are shown in Figure 4.2-14.

Flow holes are provided just above the transition of the two diameters to permit the entrance of cooling water during normal operation, and to accommodate the outflow of water from the dashpot during reactor trip.

The dashpot is closed at the bottom by means of a welded end plug. The end plug is fastened to the bottom nozzle during fuel assembly fabrication.

The top ends of the guide thimbles are mechanically attached to stainless steel sleeves which are fitted through individual bored holes in the plate and welded to the plate around the circumference of each hole for the LOPAR and OFA designs. Stainless steel top nozzle inserts are attached to the guide thimbles by the three rows of four bulges as shown in Figure 4.2-13 for the VANTAGE 5 designs. The VANTAGE 5 nozzle insert-to-adaptor plate bulge joints replace the uppermost grid sleeve-to-adaptor plate welded for the LOPAR and OFA designs.

4.2.3.3.4 Grids

There are two primary materials used to construct grids for LOPAR, OFA, and VANTAGE 5 fuel designs. LOPAR uses Inconel 718 grids while the OFA and VANTAGE 5 designs use Inconel 718 for the two end grids and Zircaloy for the five middle grids and, if aplicable, the three IFM grids. Inconel 718 is chosen for the grid material because of its corrosion resistance and high strength properties. After the combined brazing and solution annealing temperature cycle, the grid material is age hardened to obtain the material strength necessary to develop the required grid spring forces. Zircaloy is used because it has a small capture cross-section for thermal neutrons and because it is resistant to corrosion by water at operating temperatures. A more detailed description can be found in Reference 1.

The Zircaloy interlocking strap joints are laser welded at the top and bottom intersects while the grid assemblies consist of individual slotted straps which are assembled and interlocked in an "egg-crate" type arrangement. Inconel grid joints are furnace brazed along the length of the strap intersects to permanently join the straps at their points of intersection. Details such as spring fingers, support dimples, mixing vanes, and tabs are punched and formed in the individual straps prior to assembly.

The VANTAGE 5 with IFM fuel assembly design has a low pressure drop (LPD) Zircaloy structural grid design and IFM grids located in the three uppermost spans between the Zircaloy structural grids. The LPD grid has diagonal springs and a reduced grid height. The LPD grid cells use the standard four dimples and two spring support locations per cell. The function of the IFM grids is mid=span flow mixing in the hottest fuel assembly spans. Each IFM grid cell contains four dimples as shown in Figure 4.2-15 which are designed to prevent mid-span channel closure in the spans containing IFMs and to prevent fuel rod contact with the mixing vanes.

Grid assemblies with and without mixing vanes are used in the fuel assembly. One type having mixing vanes which project from the edges of the straps into the coolant stream is used in the high heat region of the fuel assemblies for mixing of the coolant. Grids of the second type, located at the bottom and top ends of the assembly, are of the nonmixing type.

The spacing between grids along the axial length of the fuel assembly is shown on Figures 4.2-2 and 4.2-10. The variation in span lengths is the result of optimization of the thermal-hydraulic and structural parameters. The grids are fastened securely to each guide thimble.

The outside straps on the grids contain mixing vanes which, in addition to their mixing function, aid in guiding the grids and fuel assemblies past projecting surfaces during handling or loading and unloading the core. Additional small tabs on the outside straps and the irregular contour of the straps are also for this purpose. The VANTAGE 5 designs use an anti-snag outer strap design on each of the grid assemblies.

4.2.3.3.5 Fuel Rods

The fuel rods consist of uranium dioxide ceramic pellets contained in a slightly cold worked and partially annealed Zircaloy-4 tubing which is plugged and seal welded at the ends to encapsulate the fuel. Sufficient void volume and clearances are provided within the rod to accommodate fission gases released from the fuel, differential thermal expansion between the cladding and the fuel, and fuel swelling due to accumulated fission products without overstressing of the cladding or seal welds.

The VANTAGE 5 designs utilize axial blanket and IFBA features. The axial blankets are a nominal six inches of unenriched fuel pellets at each end of the fuel rod pellet stack. Axial blankets reduce neutron leakage and improve fuel utilization. The axial blankets utilize chamfered unenriched pellets physically different than enriched pellets to prevent accidental mixing during manufacturing. The chief physical difference is a longer unenriched pellet length than the enriched pellet (see Table 4.2-2). The IFBA coated pellets are identical to the enriched uranium dioxide pellets except for the addition of a thin boride coating less than 0.001 inch in thickness on the pellet cylindrical surface. Coated pellets occupy the central portion of the fuel column. The number and pattern of IFBA rods within an assembly may vary depending on the specific application. The ends of the enriched coated pellets and enriched uncoated pellets are dished to allow for greater axial expansion at the pellet centerline and increase void volume for fission gas release.

Shifting of the fuel within the cladding is prevented during handling or shipping prior to core loading by a stainless steel helical compression spring which bears on the top of the fuel.

At assembly, the pellets are stacked in the cladding to the required fuel height. The compression spring is then inserted into the top end of the fuel and the end plugs pressed into the ends of the tube and welded. All fuel rods are internally pressurized with helium during the welding process. A hold-down force of approximately four to six times the weight of the fuel is obtained by compression of the spring between the top end plug and the top of the fuel pellet stack.

Each fuel assembly will be identified by means of a serial number engraved on the upper nozzle. The fuel pellets will be fabricated by a batch process so that only one enrichment region is processed at any given time. The serial numbers of the assemblies and corresponding enrichment will be documented by the manufacturer and verified prior to shipment.

Each assembly will be assigned a specific core loading position prior to insertion. A record will then be made of the core loading position, serial number and enrichment. Prior to initial core loading, two independent checks will be made to ensure that this assignment is correct. Subsequent refueling cores will be verified with a television camera.

During initial core loading and subsequent refueling operations, detailed written handling and checkoff procedures were utilized throughout the sequence. Handling and checkoff procedures will continue to be used for future refueling operations. The Cycle 1 core was loaded in accordance with the core loading diagram similar to Figure 4.5-7 which shows the typical location for each of the three enrichment types of fuel assemblies used in this loading and in subsequent reloads.

4.2.3.3.6 Neutron Source Assemblies

Four neutron source assemblies were utilized in the Cycle 1 core. These consisted of two assemblies with four secondary source rods each and two assemblies with one primary source rod each. The source rods in each secondary assembly were fastened to a spider at the top end. The primary source rods were attached to a burnable absorber assembly. Currently, only two secondary source assemblies are in each core.

In the core, the neutron source assemblies are inserted into the RCCA guide thimbles in fuel assemblies at unrodded locations. The location and orientation of the assemblies in the core is shown in Figure 4.2-3.

The primary and secondary source rods both utilize the same type of cladding material as the absorber rods (cold-worked type 304 stainless steel tubing, with 0.019 inch thick walls) into which the sources are inserted. The secondary source rods contain antimony-beryllium (Sb-Be) pellets stacked to a height of 121.75 inches. The primary source rods are of two types. One source rod contains capsules of plutonium 238-beryllium (Pu 238-Be) source material 24-inches-long at a neutron strength of approximately 2×10^8 neutrons/sec. The other primary source rod consists of a californium 252 (Cf-252) capsule two inches long at a neutron strength of approximately 4×10^8 neutrons/sec. Design criteria for the source rods are: the cladding is free standing, the pressure stresses are less than allowable stresses established by using the ASME Boiler and Pressure Vessel Code as a guide, and internal gaps and clearances are provided to allow for differential expansions between the source material and cladding.

4.2.3.3.7 Plugging Devices

In order to limit bypass flow through the RCCA guide thimbles in fuel assemblies which do not contain control rod, peripheral power suppression assemblies, source assemblies, or burnable absorber assemblies, the fuel assemblies at these locations are fitted with plugging devices. The plugging devices consist of a flat retainer plate with short rods suspended from the bottom surface and a spring pack assembly. At installation in the core, the plugging devices fit with the fuel assembly top nozzles and rest on the adaptor plate. The short rods project into the upper ends of the thimble tubes to reduce the bypass flow area. The spring pack is compressed by the upper core plate when the upper internals package is lowered into place. Similar short rods are also used on the source assemblies to fill the ends of all vacant fuel assembly guide thimbles.

All components in the plugging device, except for the springs, are constructed from Type 304 stainless steel. The springs are wound from an age hardenable nickel base alloy to obtain higher strength.

4.2.3.3.8 Burnable Absorber Rods

The burnable absorber rods are statically suspended and positioned in vacant RCCA thimble tubes within the fuel assemblies at nonrodded core locations. The absorber rods in each fuel assembly are grouped and attached together at the top end of the rods by a flat retainer plate which fits with the fuel assembly top nozzle and rests on the top adaptor plate.

The retainer plate (and the absorber rods) are held down and restrained against vertical motion through a spring pack which is attached to the plate and is compressed by the upper core plate when the reactor upper internals package is lowered into the reactor. This ensures that the absorber rods cannot be lifted out of the core by flow forces.

There are two types of burnable absorber rods. The first type is used with LOPAR fuel. It consists of borosilicate glass tubes contained within type 304 stainless steel tubular cladding. The cladding is plugged and seal welded at the ends to encapsulate the glass. The glass is also supported along the length of its inside diameter by a thin wall type 304 stainless steel tubular inner liner (Figure 4.2-4). The second type is used with OFA fuel. This is a Wet Annular Burnable Absorber (WABA) which consists of an annular aluminum oxide - boron carbide (Al_2O_3 - B_4C) absorber clad in two concentric Ziracaloy tubes with water flowing through the center (Figure 4.2-5).

The design criteria for the glass burnable absorber rods are: the cladding is free standing at reactor operating pressures and temperatures, sufficient cold void volume is provided to accommodate the total release of all helium generated in the glass as a result of the B^{10} (n,α) reaction, and pressure stresses are less than allowable stresses established by using the ASME Boiler and Pressure Vessel Code as a guide. The large void volume required for the helium is obtained through the use of glass in tubular form which provides a central void along the length of the rods. A more detailed discussion of the burnable absorber rod design is found in WCAP 9000 (see Reference 6).

Based on available data on properties of borosilicate glass and on nuclear and thermal calculations for the rods, gross swelling or cracking of the glass tubing is not expected during operation. Some minor creep of the glass at the hot spot on the inner surface of the tube is expected to occur but continues only until the glass comes into contact with the inner liner. The inner liner is provided to maintain the central void along the length of the glass and to prevent the glass from slumping or creeping into the void as a result of softening at the hot spot.

The top end of the inner liner is open to receive the helium which diffuses out of the glass.

To ensure the integrity of the glass burnable absorber rods, the tubular cladding and end plugs are procured to similar specifications and standards of quality used for stainless steel fuel rod cladding and end plugs in other Westinghouse plants. In addition, the end plug seal welds are checked for integrity by visual inspection and X-ray. The finished rods will be helium leak checked.

The WABA rod design is used as necessary in the Zion reload cores beginning with Unit 1 Cycle 8 which utilize 15X15 OFA fuel. Compared to the currently used Westinghouse annular borosilicate glass burnable absorber design which has a stagnant gas-filled central tube and outer stainless steel clad, the WABA design has annular aluminum oxide - boron carbide (Al_2O_3 - B_4C) absorber pellets contained within two concentric Zircaloy tubings with water flowing through the center tube as well as around the outer tube (see Figure 4.2-5). The WABA design provides significantly enhanced nuclear characteristics when compared with the borosilicate absorber rod design. The WABA design satisfies all performance and design requirements for an 18,000 EFPH irradiated life. Fuel cycle benefits result from the reduced parasitic neutron absorption of Zircaloy compared to stainless steel tubes, increased water fraction in the burnable absorber cell, and a reduced boron penalty at the end of each cycle.

4.2.3.3.8.1 Evaluation of Burnable Absorber Rods

The burnable absorber rods are positively positioned in the core inside RCCA guide thimbles and held down in place by attachment to a retainer assembly compressed beneath the upper core plate and hence, cannot be the source of any reactivity transient. Due to the low heat generation rate and to the conservative design of the absorber rods, there is no possibility for release of the absorber as a result of helium pressure increase due to clad heating during accident transients including loss of coolant.

Two glass burnable absorber rods of reduced length, but similar in design to those used in the Indian Point Plant Unit 3 Reactor, were exposed to inpile test conditions in the Saxton Test Reactor. Visual examination of the rods was made in early June 1968. A visual and profilometer examination was made July 30, 1968, after an exposure of 1900 effective full power hours (~25% B^{10} depletion). The rods were found to be in excellent condition and profilometry results showed no dimensional variation from the original new condition.

Subsequently, four demonstration assemblies, each containing two WABA demonstration rods, completed their first cycle of irradiation in the Indian Point Unit 3 reactor during the first quarter of 1982. Incore surveillance of peaking factors did not detect any abnormalities due to the demonstration rods. Visual examination of these rods during an early 1982 refueling shutdown showed satisfactory mechanical integrity. Several demonstration rods were reinserted for a second operating cycle in Indian Point Unit 3. Subsequent non-destructive examination revealed that the rods continued to perform as expected. For more detail concerning WABA rods, see Reference 1.

An experimental verification of the reactivity worth calculations for borosilicate glass tubing is presented in Reference 6.

4.2.3.3.8.2 Peripheral Power Suppression Assemblies

To reduce the neutron flux at the reactor pressure vessel, power suppression assemblies are placed at selected peripheral core locations. The suppression assemblies are the same as the plugging devices except the short rods (plugs) are replaced with longer hafnium neutron absorber rods which utilize the same Zircaloy cladding material as the WABA rods. The design criteria for the hafnium rods containing gaps and clearances allows for differential expansion between the absorber material and the cladding.

4.2.3.3.9 Removable Fuel Rod Assemblies

As part of a continuing Westinghouse fuel performance evaluation program, two surveillance fuel assemblies, each containing 52 removable fuel rods are included in Region III of the initial Zion Unit 1 core loading. The objective of this program was to facilitate interim and end-of-life fuel evaluation as a function of exposure. The rods were removed, non-destructively examined, and reinserted at the end of intermediate fuel cycles. At end-of-life, the rods can be removed easily and subjected to a destructive examination.

These assemblies were used as two of the four assemblies in an EPRI/Westinghouse extended burnup program. The extended burnup program irradiated these assemblies for five cycles to demonstrate satisfactory fuel performance. Higher assembly burnups would improve uranium fuel utilization if reprocessing of nuclear fuel elements is not used.

Commonwealth Edison Company has reviewed the EPRI/Westinghouse program to assure that, as a minimum, the program will not:

1. Adversely affect safety operation of the Zion plants;

2. Increase coolant activity levels; or

3. Require a reduction in plant capacity, availability, or flexibility.

This detailed review has included examining the effects of extended burnups on fission gas release, clad strain including strain concentrations, clad stress, halogen stress-corrosion cracking, leaking fuel element performance, rod bow penalties on $F^N_{\Delta H}$, accelerated "waterside corrosion", massive local internal hydriding, uranium oxidation in leaking fuel, and clad flattening.

The overall dimensions, rod pitch, number of rods, and materials are the same as for other Region 3 assemblies. These fuel rods will be fabricated in parallel with the regular Region III rods using selected Region 3 clad and pellets assembled and released to the same manufacturing tolerance limits. Mechanically, the special assemblies differ only slightly from other Region 3 assemblies. These differences are:

1. The end plugs on the removable rods are designed to facilitate removal and reinsertion.
2. The upper nozzle adapter plate on the two assemblies is modified to allow access to the removable rods.
3. The base plate on thimble plug assemblies is modified to provide the axial restraint of the fuel rods normally provided by the upper end plate. The distances between the top of the rods and the restraining plates are identical in both types of assembly.

Figure 4.2-6 compares the mechanical design of a removable fuel rod to that of a standard rod; Figure 4.2-7 shows the removable rod fuel assembly, the modified upper end plate and thimble plug assembly, to compare to a standard assembly shown in Figures 4.2-8 and 4.2-2. The location of the removable rods within the fuel assembly is shown in Figure 4.2-9. The surveillance assembly locations within the core for the first cycle are in positions C-13 and N-3 as given in Figure 4.5-7.

The mechanical design of the surveillance assemblies is essentially identical to the design of the Surry Unit 1 Special Assemblies and the San Onofre EEl Demonstration Program Removable Rod Assemblies.

In the past, experience with removable rods has been attained at Saxton, Yankee and Zorita; and additional experience will be acquired at the San Onofre Cycle 2 and Surry Unit 1. Over 300 fuel rods were removed and reinserted into assemblies during the Saxton reconstitution without evidence of failure. Leak detection tests were performed on the assemblies after all rods were reinserted, and no leakage was detected. An equally large number of Saxton rods have been successfully removed, examined and reinserted into over 12 3x3 subassemblies at Saxton. In addition, 28 full length Yankee rods were removed, examined and reinserted into Yankee Core V special assemblies. Similar handling of 22 removable rods was successfully completed during the first Zorita refueling. All such fuel handlings have been done routinely and without difficulty.

The same fuel rod design limits indicated in Section 4.5.2, fuel temperature and internal pressure, are maintained for these removable rods and there is no reduction in margin to DNB. Their inclusion in the initial Zion Unit 1 core loading introduces no additional safety considerations and in no way changes the safeguard analyses and related engineering information presented in previously submitted material in support of the license application.

4.2.3.3.10 Effects of Vibration and Thermal Cycling on Fuel Assemblies

Analyses of the effect of cyclic deflection of the fuel rods, grid springs, RCCA control rods, and burnable absorber rods due to hydraulically induced vibrations and thermal cycling show that the design of the components is good for an infinite number of cycles.

In the case of the fuel rod grid spring support, the amplitude of a hydraulically induced motion of the fuel rod is extremely small (~0.001), and the stress associated with the motion is significantly small (<100 psi). Likewise, the reactions at the grid spring due to the motion is much less than the preload spring force, and contact is maintained between the fuel clad and the grid spring and dimples. Fatigue of the clad and fretting between the clad and the grid support is not anticipated. The effect of thermal cycling on the grid-clad support is merely a slight relative movement between the grid contact surfaces and the clad which is gradual in nature during heatup and cooldown. Since the number of cycles of the occurrence is small over the life of a fuel assembly (~3 years), negligible wear of the mating parts is expected.

The dynamic deflection of the full-length control rods and the burnable absorber rods is limited by their fit with the inside diameter of either the upper portion of the guide thimble or the dashpot (0.0765-inch diametral clearance at guide thimble; 0.0145-inch diametral clearance at the dashpot). With this limitation, the occurrence of truly cyclic motion is questionable. However, an assumed cyclic deflection through the available clearance gap results in an insignificantly low stress in either the clad tubing or in the flexture joint at the spider or retainer plate. The above consideration assumes the rods are supported as cantilevers from the spider, or the retainer plate in the case of the burnable absorber rods.

A calculation, assuming the rods are supported by the surface of the dashpots and at the upper end by the spider or retainer, results in a similar conclusion.

4.2.3.3.10.1 Fuel Rod Bowing

Rod bow is a phenomenon which results in a flow area reduction in a given subchannel. Different thermal expansion forces within the fuel assembly lattice arrangement are a primary cause of increased stress in the rods. This is due to the fact that the rods are held in place vertically through the use of a frictional force applied by the grid springs. These induced stresses are released through the mechanism of cladding creep which will tend to magnify any as-built bow in a given rod. Due to the lattice arrangement, any bowed rod which is not on the periphery of the core must reduce the flow area of an adjacent channel, thus affecting DNB. The effect of rod bowing on DNB is discussed in Section 4.4.2.3.3.1.

4.2.3.3.11 Fuel Assembly and RCCA Mechanical Evaluation

To confirm the mechanical adequacy of the fuel assembly and full-length RCCA, functional test programs have been conducted on a full-scale Indian Point No. 2 prototype 12-foot canless fuel assembly and control rod. The prototype assembly was tested under simulated conditions of reactor temperature, pressure, and flow for approximately 1000 hours. The prototype mechanism accumulated 2,260,892 steps and 600 trips. At the end of the test the control rod drive mechanism was still operating satisfactorily. A correlation was developed to predict the amplitude of flow excited vibration of individual fuel rods and fuel assemblies. Inspection of the fuel assembly and drive line components did not reveal significant fretting. The wear of the absorber rods, fuel assembly guide thimbles, and upper guide tubes was minimal. The control rod free fall time against 125% of nominal flow was less than 1.5 seconds to the dashpot (10 feet of travel). Additional tests had previously been made on a full-scale San Onofre mock up version of the fuel assembly and control rod (see Reference 7). A discussion of Reactor Coolant System pipe rupture is contained in Section 15.6.5.

4.2.3.3.11.1 Indian Point No. 2 Mockup Tests

A $^1/_7$ scale model of the Indian Point No. 2 internals was designed and built for hydraulic and mechanical testing. The tests provided information on stresses and displacements at selected locations on the structure due to static loads, flow induced loads, and electromagnetic shaker loads. Flow distribution and pressure drop information were obtained. Results of the static tests indicated that mean strains in the upper core support plate and upper support columns are below design limits. Strains and displacements measured in the model during flow tests verified that no damaging vibration levels were present. Additional information gained from the tests were the natural frequency and damping of the thermal shield and other components in air and water. Model response can be related to the full scale Zion plant for most of the expected exciting phenomena, but across the board scaling is not possible. Specifically, exciting phenomena which are strongly dependent on Reynolds number cannot be scaled. In areas where Reynolds number may be important, either (1) the measured vibration amplitudes were many times lower than a level that would be damaging, or (2) full-scale vibration data has been obtained.

4.2.3.3.11.2 Loading and Handling Tests

Tests simulating the loading of the prototype fuel assembly into a core location have also been successfully conducted to determine that proper provisions had been made for guidance of the fuel assembly during refueling operation.

4.2.3.3.11.3 Axial and Lateral Bending Tests

Axial and lateral bending tests have been performed in order to simulate mechanical loading of the assembly during refueling operation. Although the maximum column load expected to be experienced in service is approximately 1000 lb, the fuel assembly was successfully loaded to 2200 lb axially with no damage resulting. This information is also used in the design of fuel handling equipment to establish the limits for inadvertent axial loads during refueling.

4.2.3.3.11.4 Withdrawal of a Single Control Rod

No single electrical or mechanical failure in the Rod Control System could cause the accidental withdrawal of a single RCCA from the inserted bank at full power operation. The operator could deliberately withdraw a single RCCA in the control bank. This feature is necessary in order to retrieve a rod, should one be accidentally dropped. In the extremely unlikely event of simultaneous electrical failures which could result in single RCCA withdrawal in the four-loop plant, Rod Deviation and Rod Control Urgent Failure would both be displayed on the plant annunciator, and the rod position indicators (RPI) would indicate the relative positions of the rods in the bank. The Urgent Failure Alarm also inhibits automatic rod motion in the group in which it occurs. Withdrawal of a single RCCA by operator action, whether deliberate or by a combination of errors, would result in activation of the same alarm and the same visual indications.

Each bank of control rods in the system is divided into two groups of 4 mechanisms each (except group 2 of bank D which consists of 5 mechanisms). The rods comprising a group operate in parallel through multiplexing thyristors. The two groups in a bank move sequentially such that the first group is always within one step of the second group in the bank. A definite schedule of actuation (or deactuation) of the stationary gripper, movable gripper, and lift coils of a mechanism is required to withdraw the RCCA attached to the mechanism. Since the four stationary gripper, movable gripper, and lift coils associated with the four RCCAs of a rod group are driven in parallel, any single failure which would cause rod withdrawal would affect a minimum of one group, or four RCCAs. Mechanical failures are in the direction of insertion, or immobility.

In the extremely unlikely event of multiple failures which result in continuous withdrawal of a single RCCA, it is not possible, in all cases, to provide assurance of automatic reactor trip such that core safety limits are not violated. Withdrawal of a single RCCA results in both positive reactivity insertion tending to increase core power, and an increase in local power density in the core area "covered" by the RCCA.

If analysis of possible failure mechanisms is disregarded, two cases must be considered as follows:

1. If the reactor is in the automatic control mode, withdrawal of a single RCCA will result in insertion of the other RCCAs in the controlling bank to compensate for the positive reactivity insertion. Core power and coolant temperature thus remain close to their initial values. However, for the continuous withdrawal which might cause the D rods to go below the bottom of the insertion limit, core limits might be exceeded. The power density in the area of the withdrawn rod increases, resulting in hot channel factors greater than design values. While considerable margin in $F_{\Delta H}$, in terms of DNB, exists at nominal conditions, the possible consequences of this accident include a DNB ratio of less than the DNBR limit (see Section 4.4) for a small number of fuel rods in the area of the withdrawn RCCA.

2. If the reactor is in the manual control mode, continuous withdrawal of a single RCCA results in both an increase in core power and coolant temperature, and an increase in the local hot channel factor in the area of the failed RCCA. In terms of overall system response, this case is similar to those presented in Section 15.4.2 of the Zion UFSAR; however, the increased local power peaking in the area of the withdrawn RCCA results in lower minimum DNB ratios than for the withdrawn bank cases. Depending on initial power level and location of the failed RCCA, automatic reactor trip may not occur sufficiently fast to prevent the minimum core DNB ratio from falling below 1.30. Thus, for this case, the possible consequences include having fuel rods in the area of the RCCA with a DNB ratio less than the DNBR limit (see Section 4.4).

It should be noted that for accidents such as discussed in 1 above, no automatic reactor trip will result; however, core total power and coolant temperature will remain close to their values. For cases such as 2 above, a trip will ultimately ensue, although not sufficiently fast in all cases to prevent a minimum DNB ratio in the core of less than the DNBR limit. For both cases, where the minimum core DNB ratio is less than the limit*, the cladding temperature is below 2100°F.

For both cases, the indicators and alarms mentioned previously would function to alert the operator to the malfunction. For case 1, discussed above, the insertion limit alarms (Lo and Lo-Lo alarms) would serve in this regard.

Table 15.4-4 provides the resulting peaking factor ($F^N_{\Delta H}$) and DNBR with Bank D fully inserted and the control rod (H-8) completely withdrawn. For the central control rod missing with Bank D at the insertion limits, the calculated F^N_Q is less than the original design value of 2.70. The rod misalignment condition is detectable from the actual rod position information displayed on the control board. The rod misalignment can also be detected by a ΔT of greater than 10°F on the thermocouple monitoring the misaligned RCC and a ΔT of greater than 4°F on the adjacent thermocouple to the misaligned rod when the normal rod insertion case and misaligned case are compared. Confirmation of misaligned rod condition may be made by comparing traces of movable detectors made in the vicinity of symmetrical control rods to the suspected misaligned control rod.

Section 4.6.1.1 contains additional information related to control rod withdrawal.

4.2.4 Testing and Inspection Plan

4.2.4.1 Fuel Quality Control

In addition to the quality assurance discussion for the core components described above, the nuclear fuel fabrication quality control measures are outlined below.

Quality Control philosophy is generally based on inspections being performed to a 95% confidence that at least 95% of the product meets specification, unless otherwise noted, using either a hypergeometric function with zero defectives for small lots or the latest revision of Mil-105D for large lots. The following inspections are included.

4.2.4.2 Inspections

4.2.4.2.1 Component Parts

All parts received are inspected to a 95 x 95 confidence level. The characteristics inspected depend upon the component parts. Inspections include dimensional, visual, and check audits of test reports, material certification and nondestructive testing such as X-ray and ultrasonic. Westinghouse materials process and component specifications specify in detail the inspection to be performed.

All material used in the manufacture of this core is accepted and released by Quality Control.

4.2.4.2.2 Pellets

Inspection is performed to a 95 x 95 confidence level for diameter and density. Visual inspections are performed for cracks, chips, porosity, and other visual characteristics according to established visual standards. Chemical analyses and hydrogen samples are taken on a blend lot basis throughout pellet production.

4.2.4.2.3 Rod Inspection

To assure that manufactured fuel rods meet a high standard of excellence from the standpoint of functional requirements, many inspections and tests are performed both on the raw material and the finished product. These tests and inspections include chemical analysis, elevated temperature, tensile testing of fuel tubes, dimensional inspection, X-ray or ultrasonic testing, of both end plug welds, helium leak tests, and gamma assay of fuel rods.

Rod inspection consists of 100% nondestructive inspection, and is based on the experience, specifications, procedures, and standards established on previously manufactured and operated cores.

The following tests and inspections ensure that 100% of the rod welds have been checked by several different techniques.

4.2.4.2.3.1 Leak Testing

Each rod is tested to a known leakage ($\leq 1x10^{-6}$ ATM [atmospheric pressure] cc/sec) using mass spectrometry with helium being the detectable gas. The leak detection systems are qualified using calibrated leak standards purchased from qualified suppliers.

4.2.4.2.3.2 Nondestructive Testing

All fuel rod welds are tested in at least one of the following means:

1. X-ray: Fuel rod weld enclosures are X-rayed at 0°, 60°, and 120° using weld correction blocks. X-rays are taken in accordance with Westinghouse Material Technical (MTS) 11200, latest version, using 2-2T penetrometer as the basis of acceptance; or

2. Ultrasonic (UT) Examination: Fuel rod welds receive ultrasonic examination utilizing qualified UT equipment, periodically calibrated with test rods.

4.2.4.2.3.3 Dimensional and Visual Inspection

All rods are dimensionally and visually inspected prior to final release and upgrading. The requirements include such items as length, camber, girth weld outer diameter (OD) using ring go-gages, and visual inspection of welds and OD surface utilizing Engineering approved visual standards.

4.2.4.2.3.4 Gamma Assay

All rods are axially scanned to verify zone and stack lengths as well as plenum length. Also, the fuel stack is scanned for gaps, off-specification pellets, and zone/stack enrichment averages.

4.2.4.2.4 Rod Upgrading

Upon final inspection, the rods are upgraded by a computer system, to ensure all inspections have been performed and are acceptable prior to fuel assembly loading.

4.2.4.2.5 Assembly Inspection and Clean Check

Inspection consists of 100% inspection for cleanliness and visual inspection of accessible surfaces for rod loading and fuel assembly handling damage.

4.2.4.2.6 Visual Examination for First Core

The visual examination program was conducted for the first core fuel in cooperation with Commonwealth Edison Company and Westinghouse. No unacceptable conditions were observed (see Reference 8).

4.2.4.2.7 Other Inspections

The following inspections will be performed as part of routine inspection operations:

1. Measurements, other than those specified above, which are critical to thermal and hydraulic analyses are obtained to enable evaluation of manufacturing variations to a 99.5% confidence level.

2. Tool and gage inspection and control includes standardization to primary and secondary working standards. Tool inspection is performed at prescribed intervals on all serialized tools. Complete records are kept of calibration and the condition of tools.

3. Check audit inspection of all inspection activities and records are performed to assure that prescribed methods are followed and that all records are correct and properly maintained.

4. Surveillance of outside contractors, including approval of standards and methods, are performed where necessary. However, all final acceptance is based upon inspection performed at the Westinghouse plant.

4.2.4.3 Assembly Enrichment

To prevent the possibility of mixing enrichments during fuel manufacture and assembly, meticulous process control is exercised. The UO_2 powder is blended in approximately 1,500 kg bulk container blends.

Powder withdrawal from bulk container storage can be made by one authorized group only who direct the powder to the correct pellet production line. All pellet production lines are physically separated from each other and pellets of only a single enrichment and density are produced in a given production area.

Finished pellets are placed on trays and then into tray storage carts pending loading into fuel tubes.

Loading of the pellets into the fuel tubes is again accomplished in isolated production lines and only one enrichment is loaded on a line at a time. Fuel rod components (clad, end plug, spring) and pellet tray traceability and accountability are maintained via computer transactions keyed to a unique bar code that is laser etched on each fuel tube. The completed fuel rod is released to a channel. The channel is then assigned to a loading magazine which "mirrors" the fuel assembly loading pattern. Each step is traced via the fuel rod bar code and the final fuel assembly loading pattern must agree with the loading magazine to be released. The top nozzle identification then becomes the fuel assembly identification for future rod traceability.

4.2.5 References, Section 4.2

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2. Daniel, R. C., et al, "Effects of High Burnup on Zircaloy-Clad Bulk UO2, Plate Fuel Element Samples," WAPD-263, (September 1965).

3. (Deleted)

4. (Deleted)

5. (Deleted)

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7. Large Closed Cycle Water Reactor Research and Development Program Quarterly Progress Reports for the Period January 1963 through June 1965 (WCAP-3738, 3739 2743, 3750, 3269-2, 3269-2 3269-5, 3269-6, 3269-12 and 3269-13).

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14. Davidson, S.L., Kramer, W.R., (Eds.), "Reference Core Report VANTAGE 5 Fuel Assembly," WCAP-10444-P-A, September 1985 and Addendum 1, March 1986.

TABLE 4.2-1 (1 of 2)

TYPICAL CORE MECHANICAL DESIGN PARAMETERS FOR LOPAR FUEL [1]

Active Portion of the Core	
Equivalent Diameter, in.	132.7
Active Fuel Height, in.	144
Length-to-Diameter Ratio	1.09
Total Cross-Section Area, Ft^2	96.06
Fuel Assemblies	
Number	193
Rod Array	15 x 15
Rods per Assembly	204[2]
Rod Pitch, in.	0.563
Overall Dimensions	8.426 x 8.426
Fuel Weight, (as UO_2),pounds	216,600
Total Weight, pounds	276,000
Number of Grids per Assembly	7
Number of Guide Thimbles	20
Diameter of Guide Thimbles (upper part), in.	0.545 O.D. x 0.515 I.D.
Diameter of Guide Thimbles (lower part), in.	0.484 O.D. x 0.454 I.D.
Fuel Rods	
Number	39,372
Outside Gap, in.	0.422
Diametral Gap, in.	
Regions 1, 2 and 3	0.0075
Clad Thickness, in.	
Regions 1, 2 and 3	0.0243
Clad Material	Zircaloy
Overall Length	151.85
Length of End Cap, overall, in.	0.688
Length of End Cap, inserted in rod, in.	0.250
Fuel Pellets (Unit 1 core values are for cycle 1)	
Material	UO_2 sintered
Density (% of Theoretical)	
Region 1	93.68
Region 2	94.5 (93 for Unit 2)
Region 3	94.5
Feed Enrichments w/o	
Region 1	2.25
Region 2	2.80
Region 3	3.30

TABLE 4.2-1 (2 of 2)

TYPICAL CORE MECHANICAL DESIGN PARAMETERS FOR LOPAR FUEL [1]

Fuel Pellets, cont.	
Diameter, in.	
Regions 1, 2 and 3	0.3659
Length, in.	0.600
Rod Cluster Control Assemblies	
Neutron Absorber	5% Cd, 15% In, 80% Ag
Cladding Material	Type 304 SS - Cold Worked
Clad Thickness, in.	0.019
Number of Clusters	
Full Length	53
Part Length	0 (Removed per Mod 78-20)
Number of Control Rods per Cluster	20
Length of Rod Control, in.	156.436 (overall) 149.136 (insertion length)
Length of Absorber Section, in.	142.00 (full length)
Core Structure	
Core Barrel, in.	
I.D.	148.0
O.D.	152.5
Thermal Shield, in.	
I.D.	158.5
O.D.	164.0
Burnable Poison Rods (First core values included as an example; specific reload values are available in design reports.)	
Number	1434
Material	Borosilicate Glass
Outside Diameter, in.	0.4395
Inner Tube, O.D. in.	0.2365
Clad Material	S.S.
Inner Tube Material	S.S.
Boron Loading (natural) gm/cm of glass rod	0.0603

---

(1) All dimensions are for cold conditions.

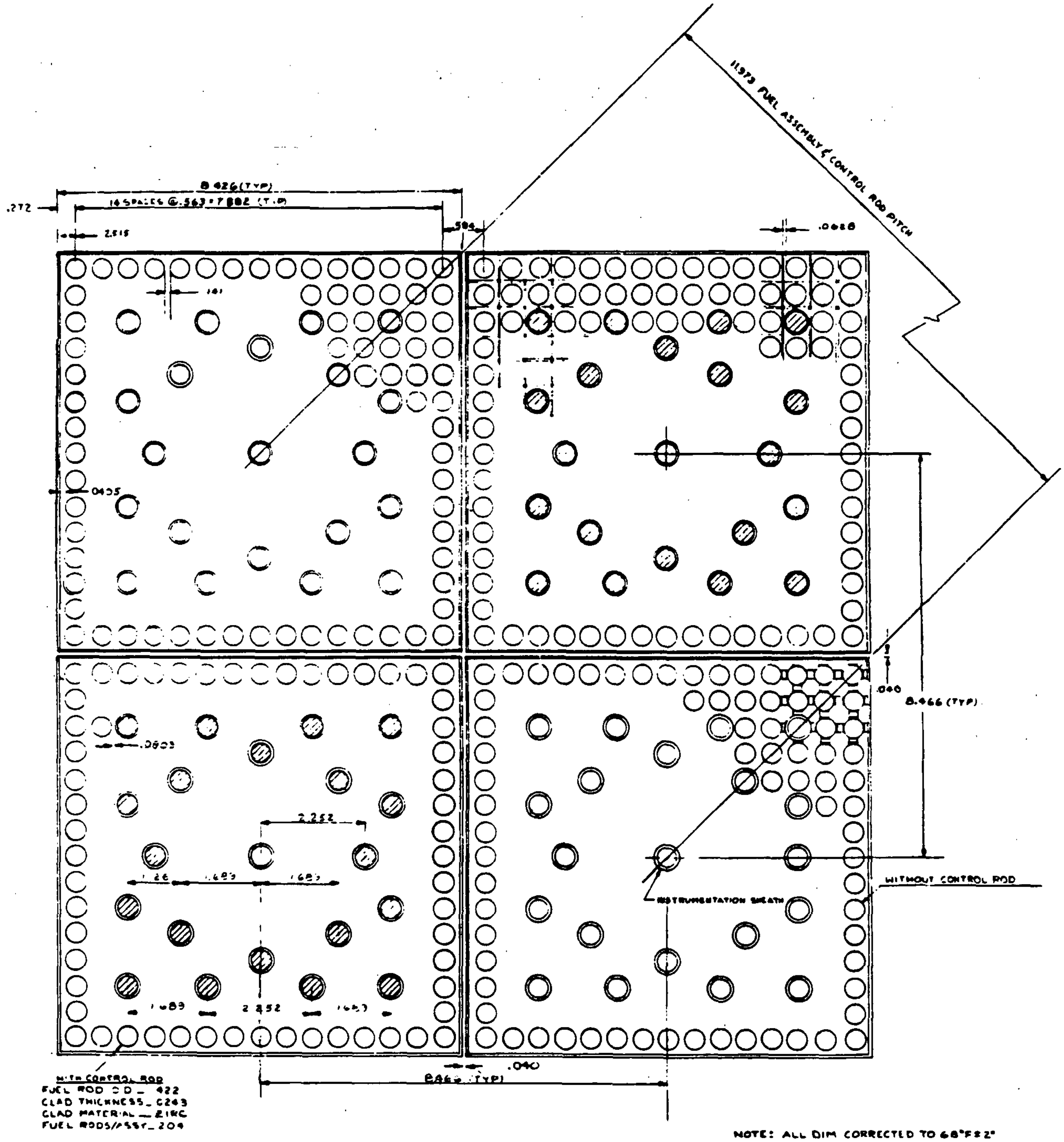
(2) Twenty-one rods are omitted: Twenty provide passage for control rods and one to contain incore instrumentation.

TABLE 4.2-2

COMPARISON OF 15X15 FUEL ASSEMBLY DESIGN PARAMETERS

PARAMETER	OFA	VANTAGE 5	VANTAGE 5 w/IFMs
Cycle #	12	13	14 & 15
Fuel Assy. Length, in[1]	159.765	159.975	159.975
Fuel Rod Length, in[1]	151.85	152.170	152.170
Assy. Envelope, in	8.426	8.426	8.426
Compatible w/Core Internals	Yes	Yes	Yes
Fuel Rod Pitch, in	0.563	0.563	0.563
Fuel Rods/Assy	204	204	204
Guide Tubes/Assy	20	20	20
Instrumentation Tubes/Assy	1	1	1
Compatible w/Movable In-core Detector System	Yes	Yes	Yes
Fuel Tube Material	Zircaloy-4	Zircaloy-4	Zircaloy-4
Fuel Rod Clad OD, in	0.422	0.422	0.422
Fuel Rod Clad Thickness, in	0.0243	0.0243	0.0243
Fuel/Clad Gap, mil	7.5	7.5	7.5
Fuel Pellet Diameter, in	0.3659	0.3659	0.3659
Fuel Pellet Length: Enriched Fuel, in.	0.439	0.439	0.439
Unenriched Fuel, in[1]	---	0.545	0.545
Guide Tube Material	Zircaloy-4	Zircaloy-4	Zircaloy-4
Guide Tube OD (Above Dashpot), in	0.533	0.533	0.533
Guide Tube Wall Thickness, in	0.017	0.017	0.017
Top Nozzle[1]	Standard	Removable	Removable
Top Nozzle Holddown Springs	3-leaf	3-leaf	3-leaf
Compatible With Fuel Handling Equipment	Yes	Yes	Yes
Bottom Nozzle[1]	304 SS Recon[2]	304 SS Recon[2] + Debris Filter	304 SS Recon[2] + Debris Filter

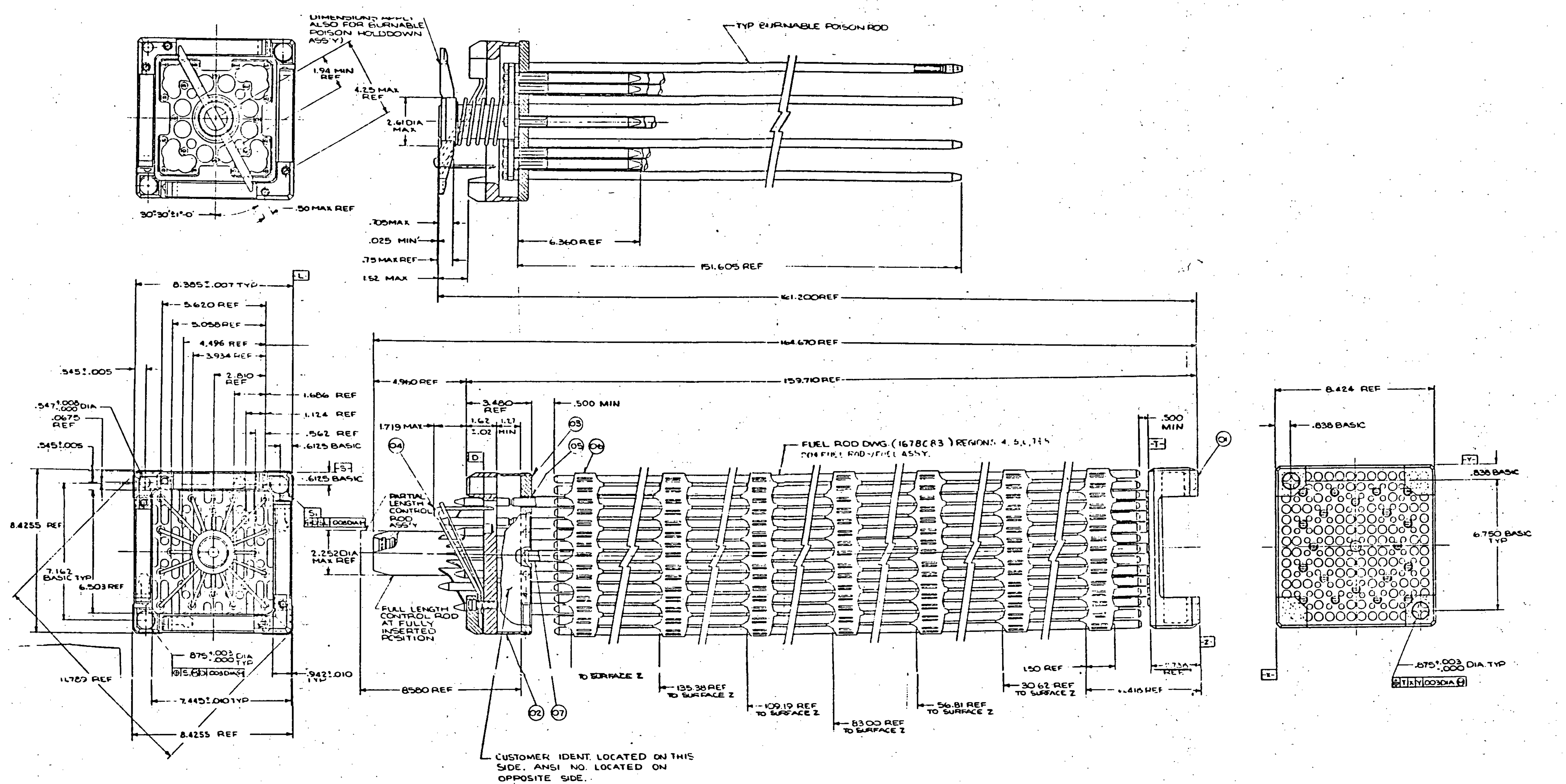
(1) VANTAGE 5 design change compared to the OFA design.
(2) Recon = Reconstitutable



ZION STATION UFSAR

Figure 4.2-1

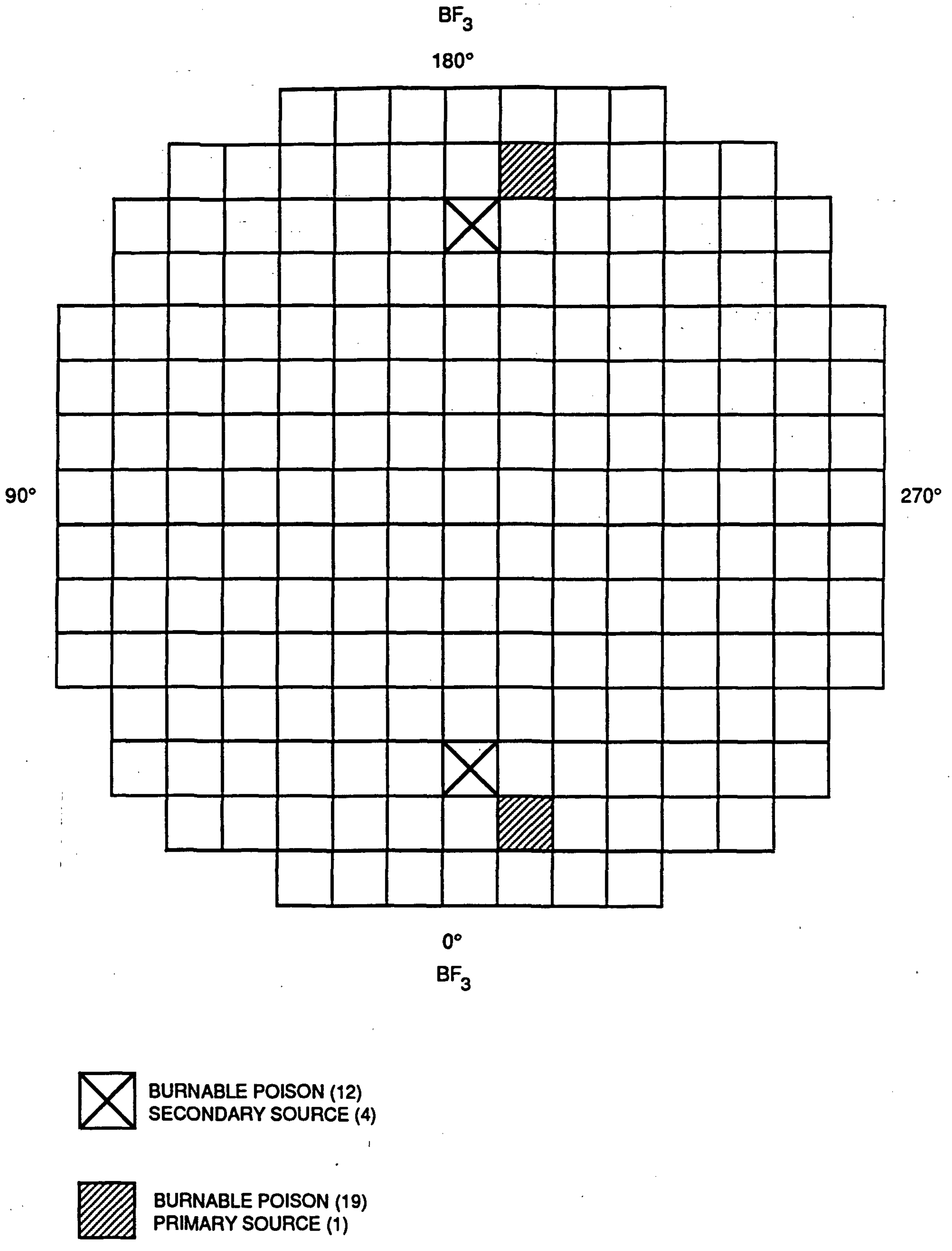
FUEL ASSEMBLY AND CONTROL CLUSTER CROSS SECTION



ZION STATION UFSAR
Westinghouse 1194E59

Figure 4.2-2

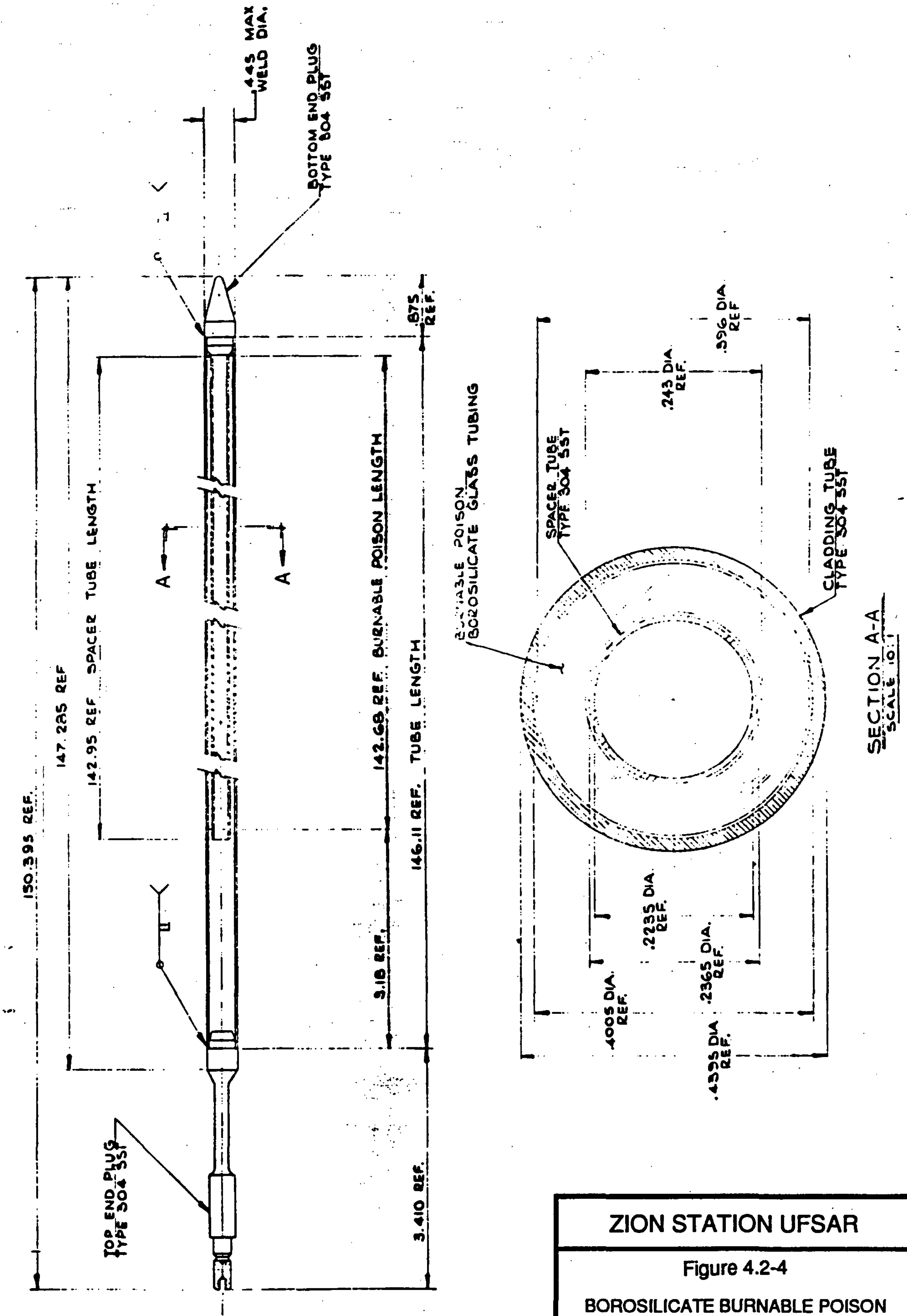
LOPAR
FUEL ASSEMBLY OUTLINE



ZION STATION UFSAR

Figure 4.2-3

FIRST CORE NEUTRON SOURCE LOCATIONS



ZION STATION UFSAR

Figure 4.2-4

BOROSILICATE BURNABLE POISON

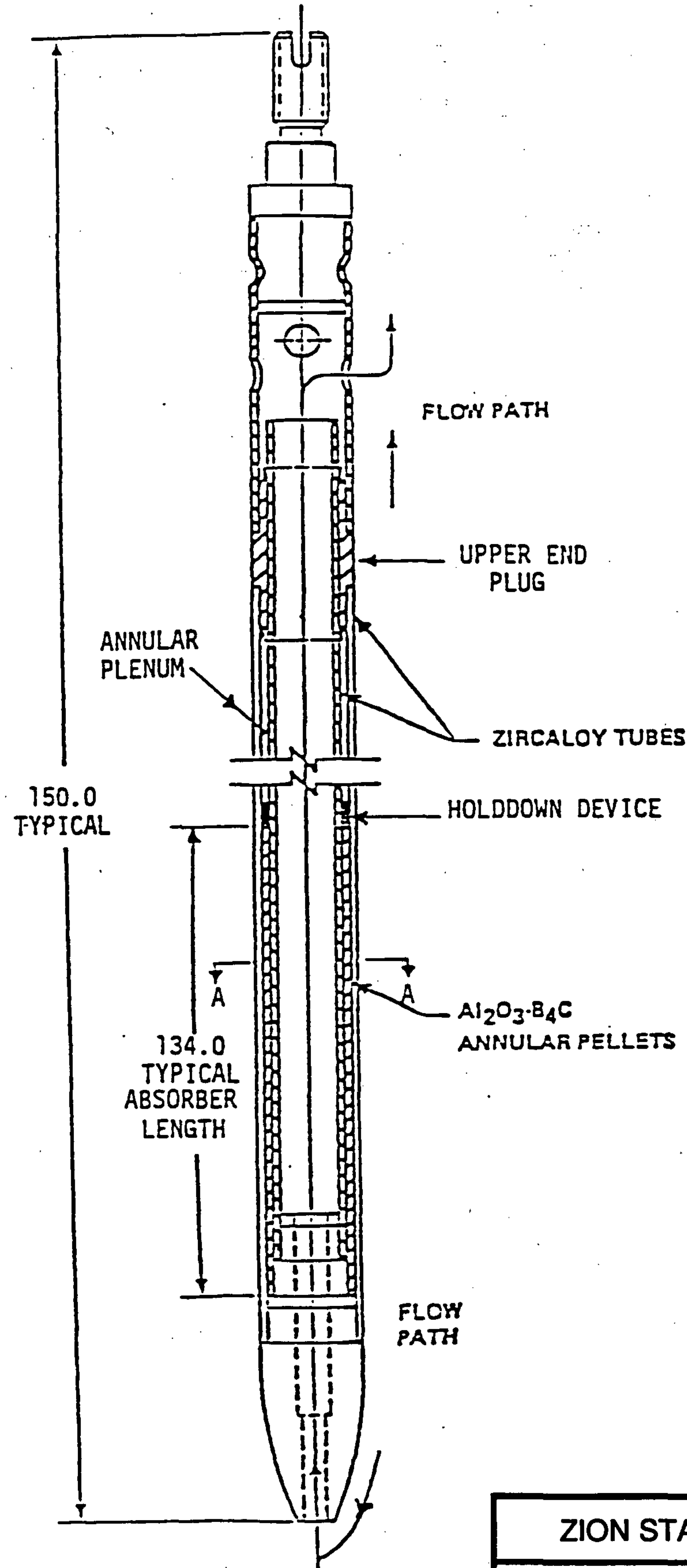


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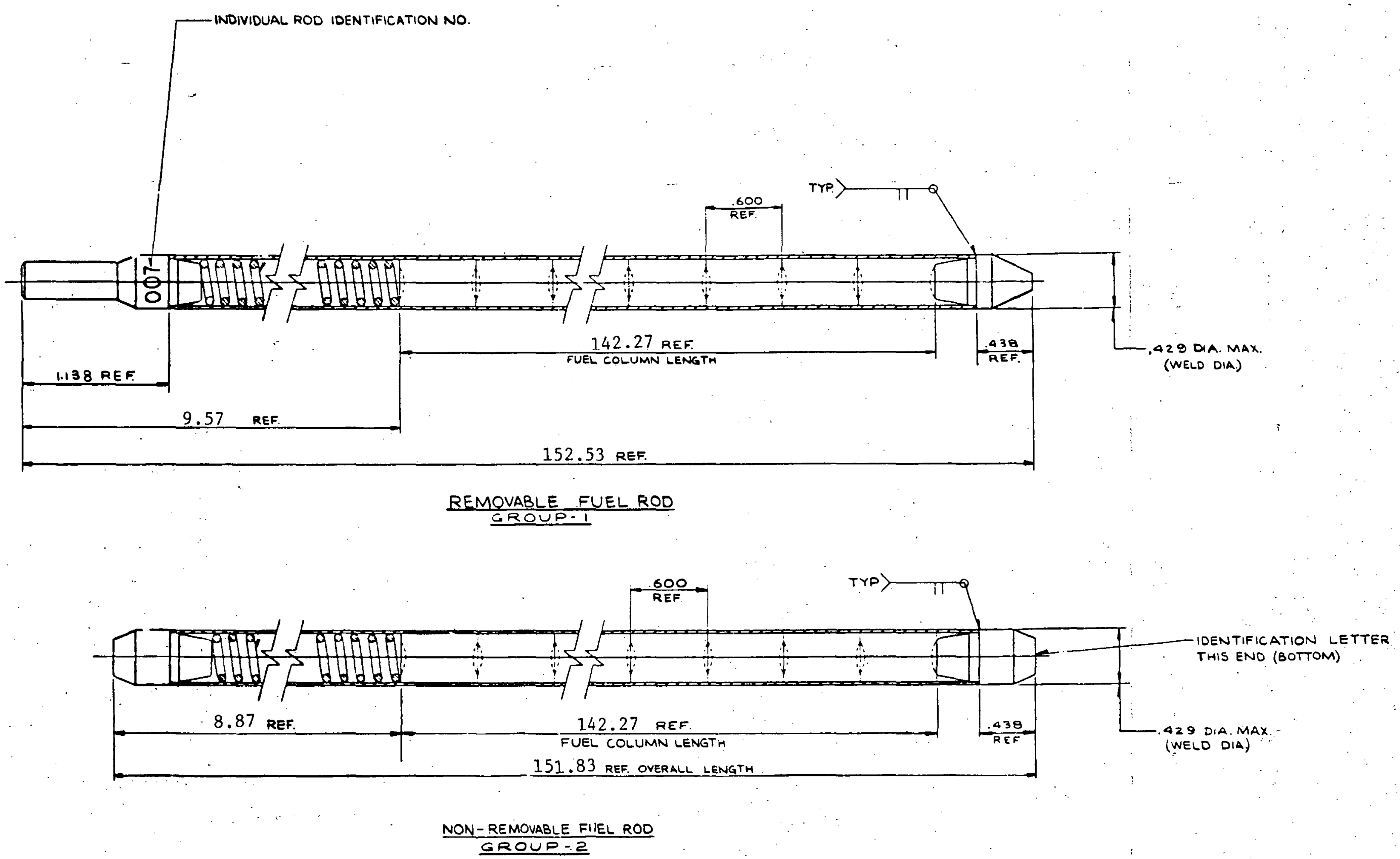
Figure 4.2-5

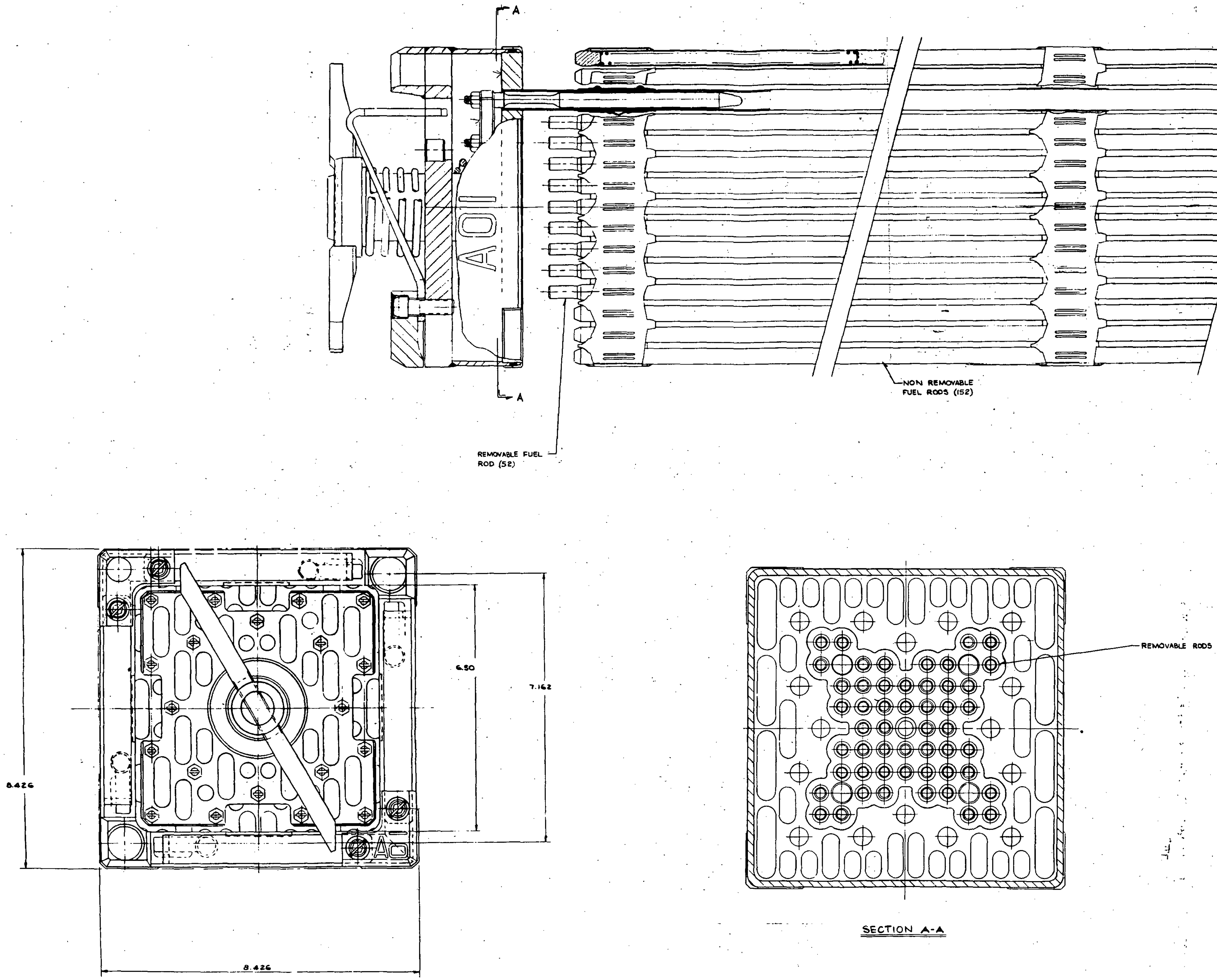
WET ANNULAR BURNABLE
ASSEMBLY (WABA)



ZION STATION UFSAR

Figure 4.2-6

REMOVABLE ROD COMPARED TO STANDARD ROD



ZION STATION UFSAR

Figure 4.2-7

REMOVABLE FUEL ROD
ASSEMBLY OUTLINE

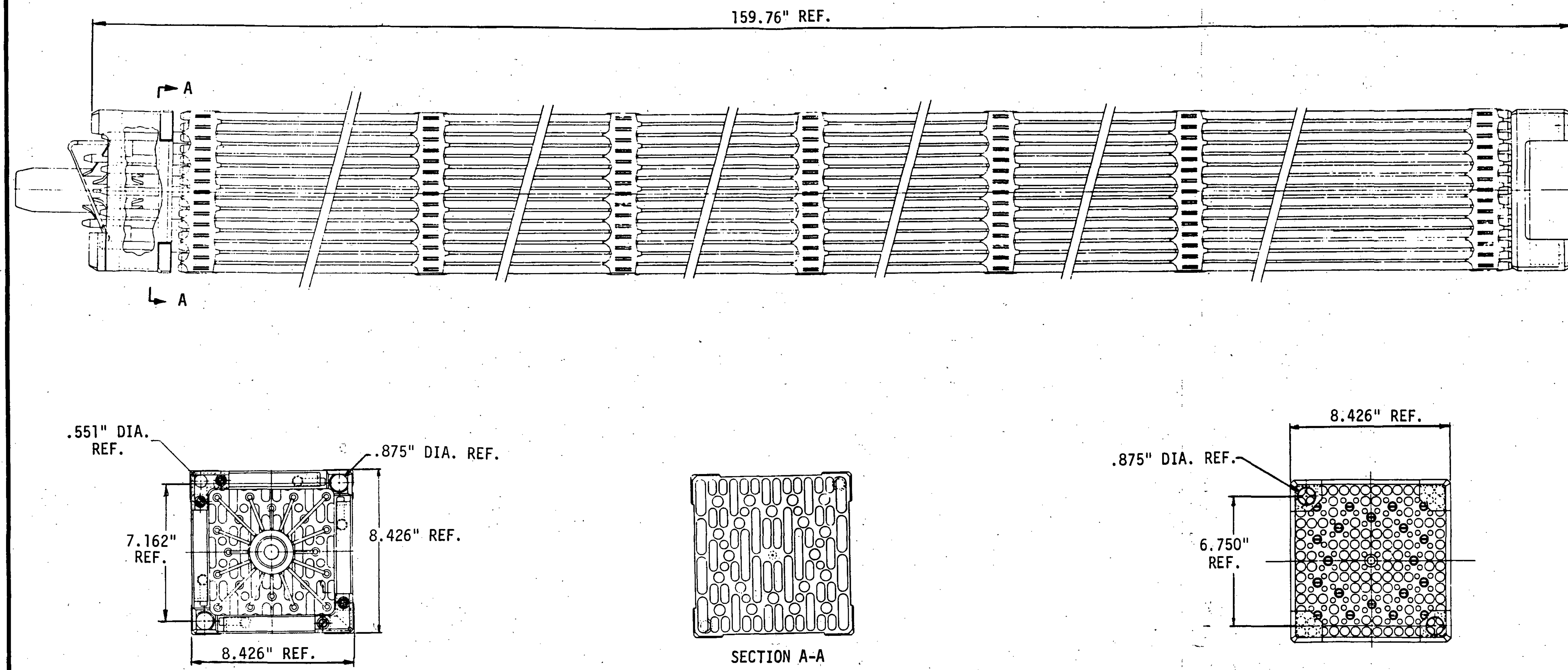


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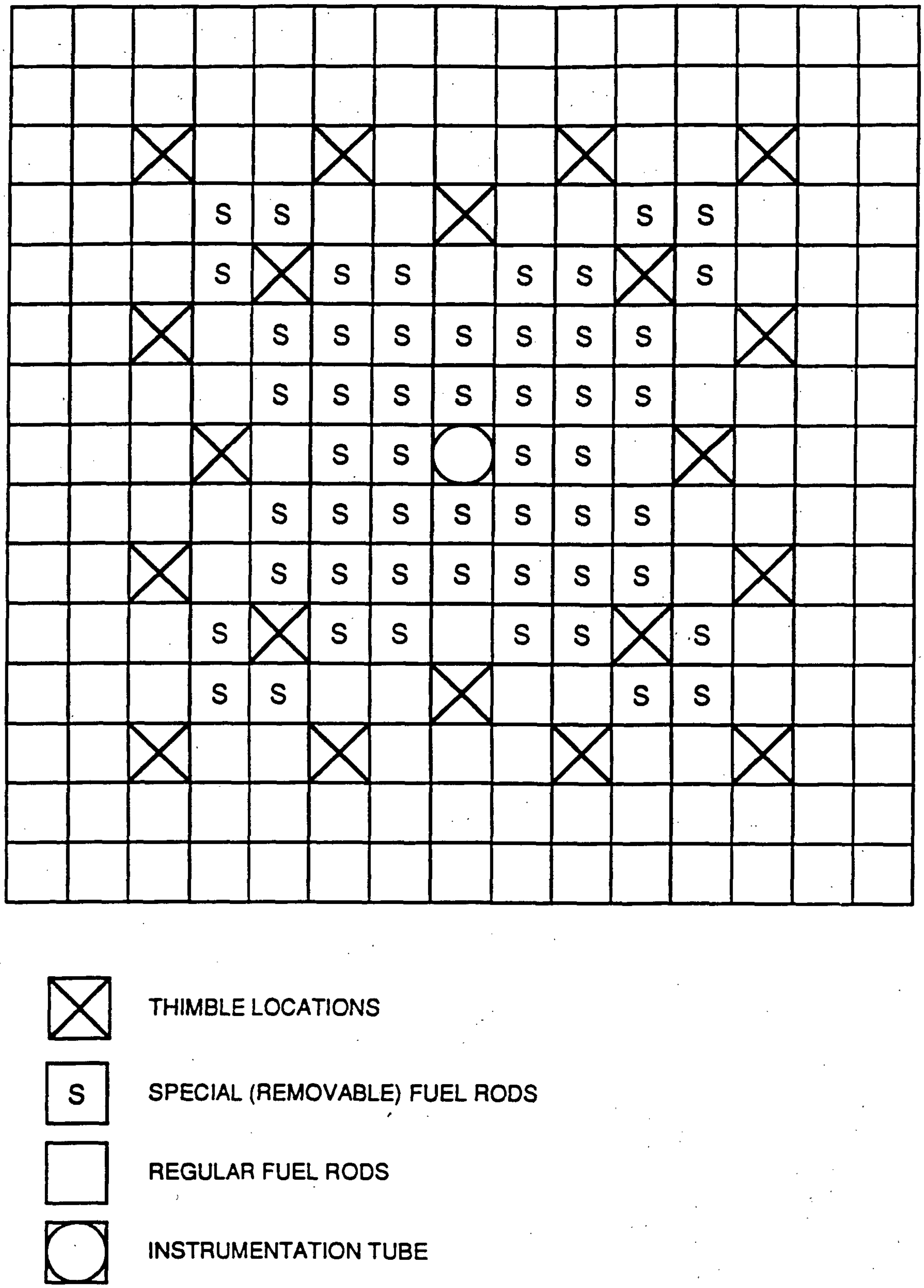
Figure 4.2-8

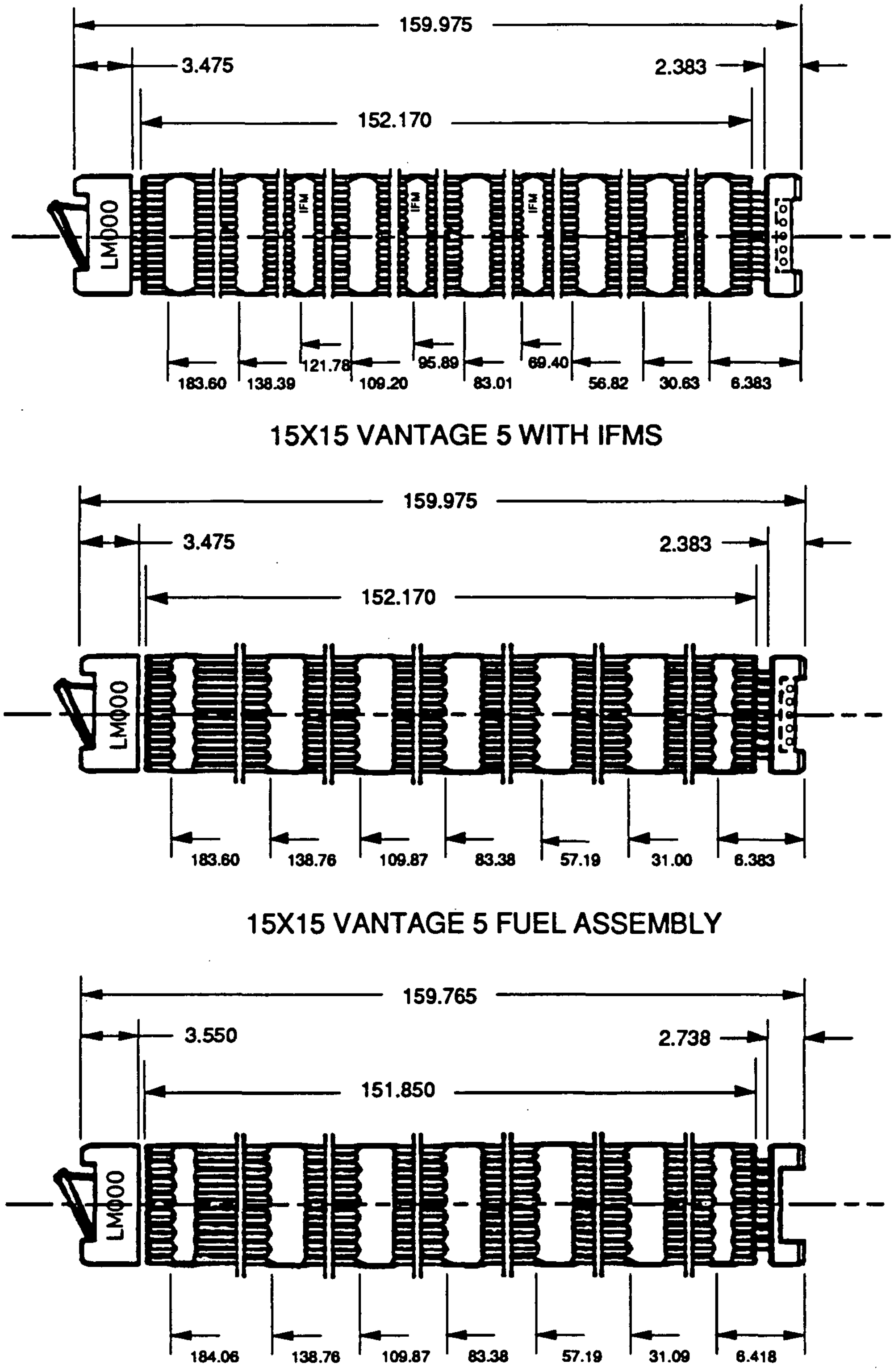
OPTIMIZED FUEL ASSEMBLY (OFA) OUTLINE



ZION STATION UFSAR

Figure 4.2-9

LOCATION OF REMOVABLE RODS WITHIN AN ASSEMBLY

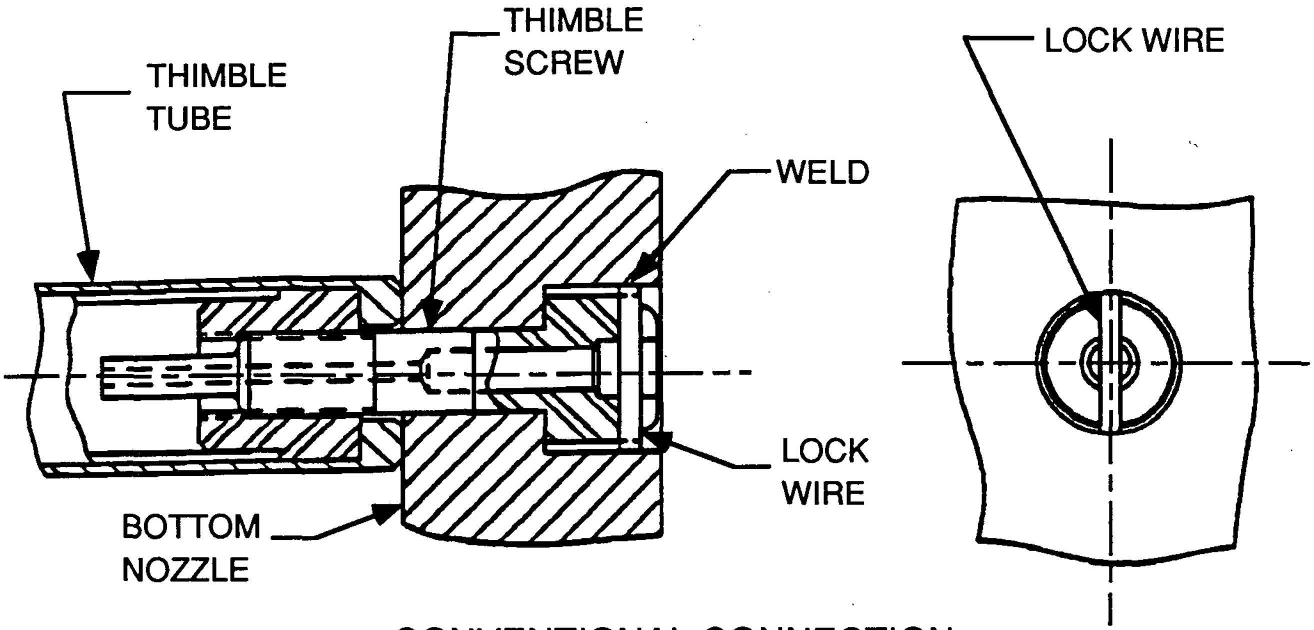


15X15 OFA FUEL ASSEMBLY

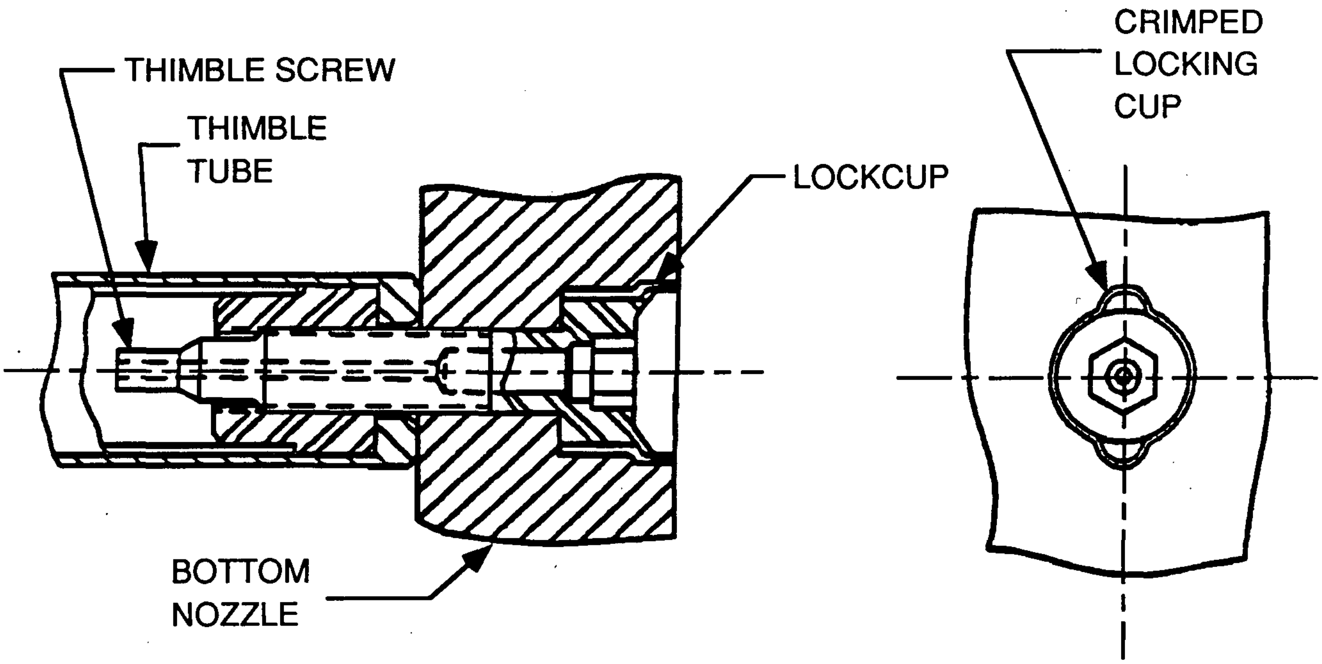
ZION STATION UFSAR

Figure 4.2-10
15X15 OFA / VANTAGE 5/ VANTAGE 5 W / IFMs FUEL ASSEMBLY COMPARISON

JULY 1993



CONVENTIONAL CONNECTION



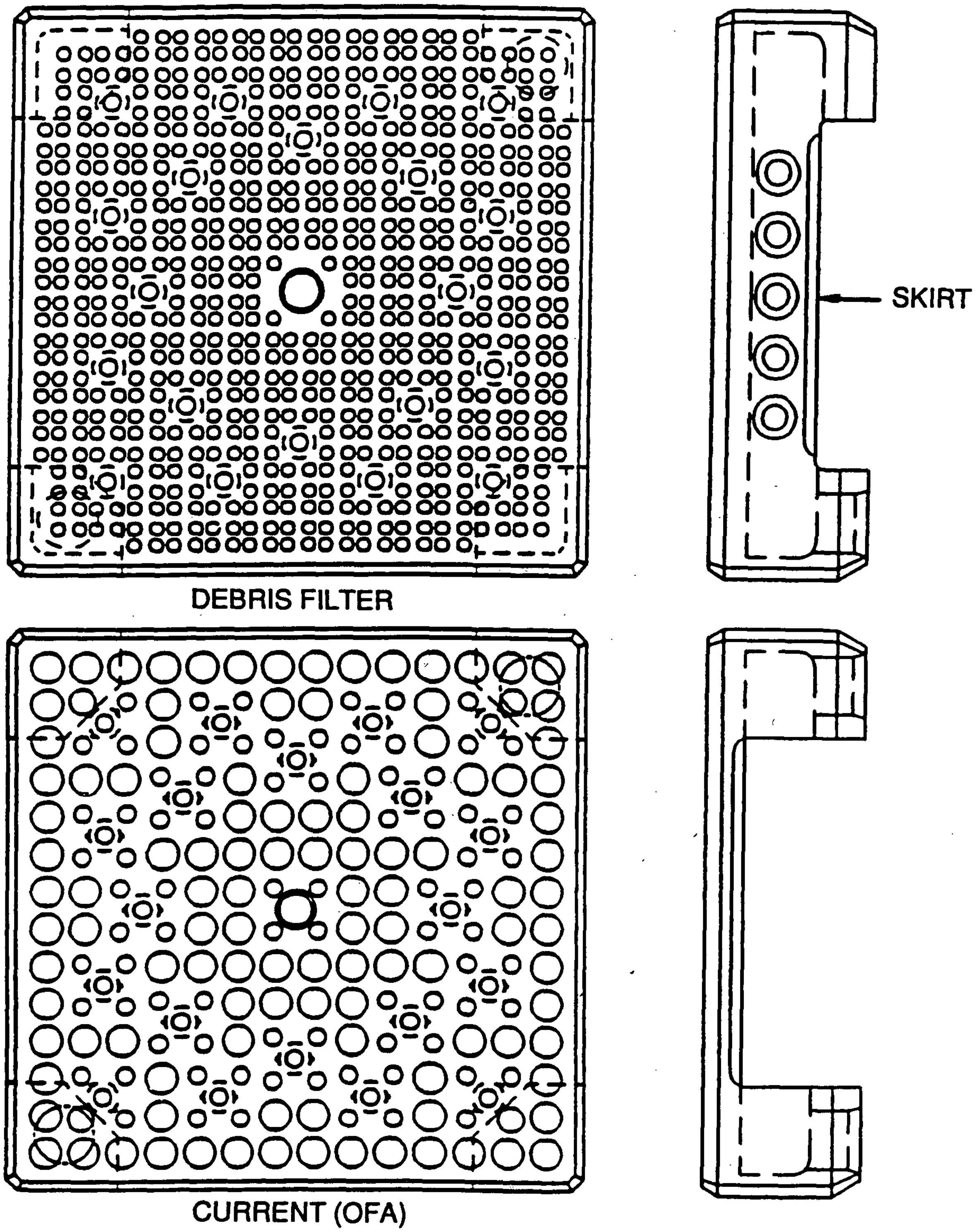
RECONSTITUTABLE CONNECTION

ZION STATION UFSAR

Figure 4.2-11

BOTTOM NOZZLE TO THIMBLE TUBE CONNECTION

JULY 1993

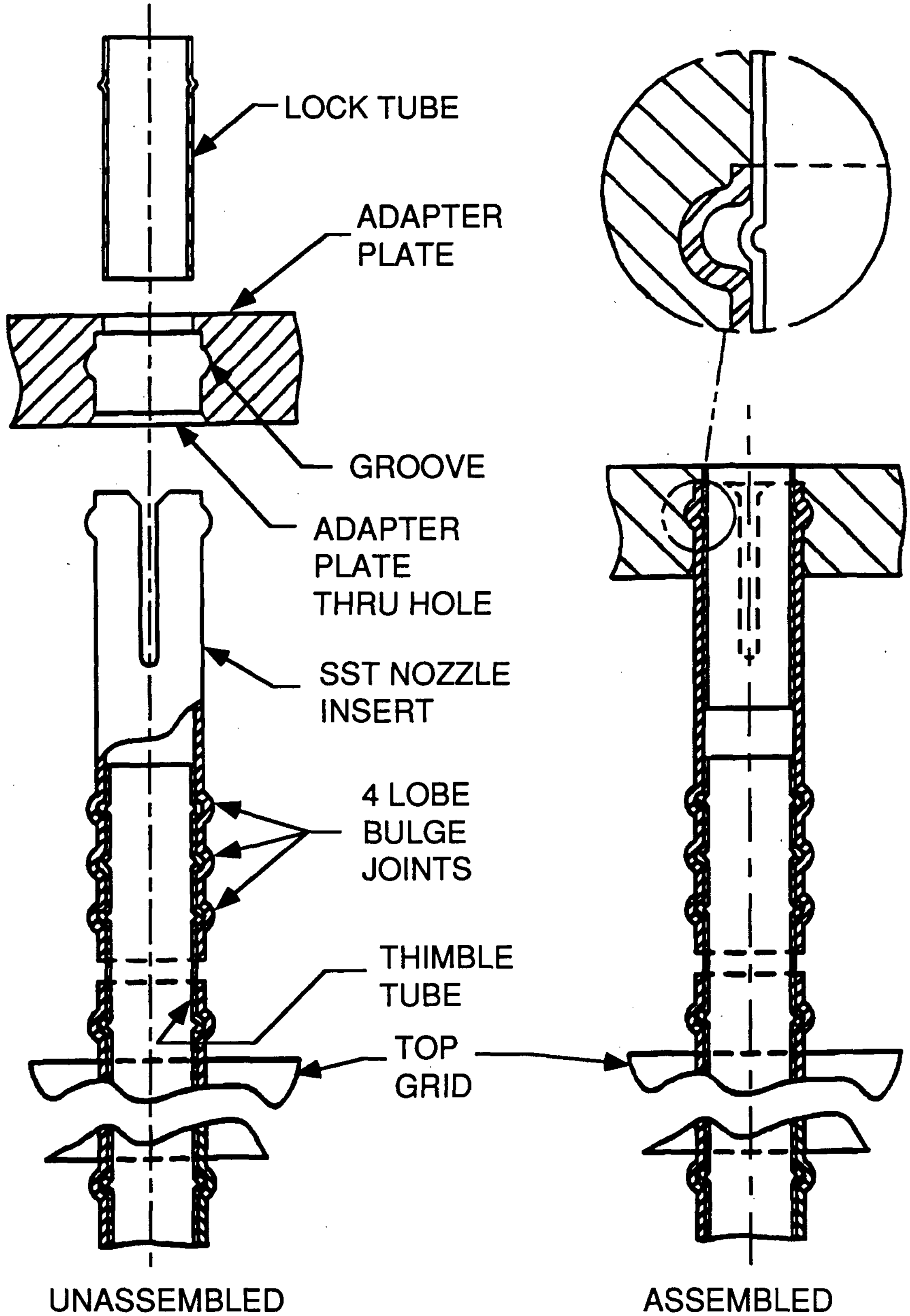


ZION STATION UFSAR

Figure 4.2-12

BOTTOM NOZZLE DESIGN COMPARISON

JULY 1993



ZION STATION UFSAR

Figure 4.2-13

TOP GRID AND RECONSTITUTABLE TOP NOZZLE ATTACHMENT DETAIL

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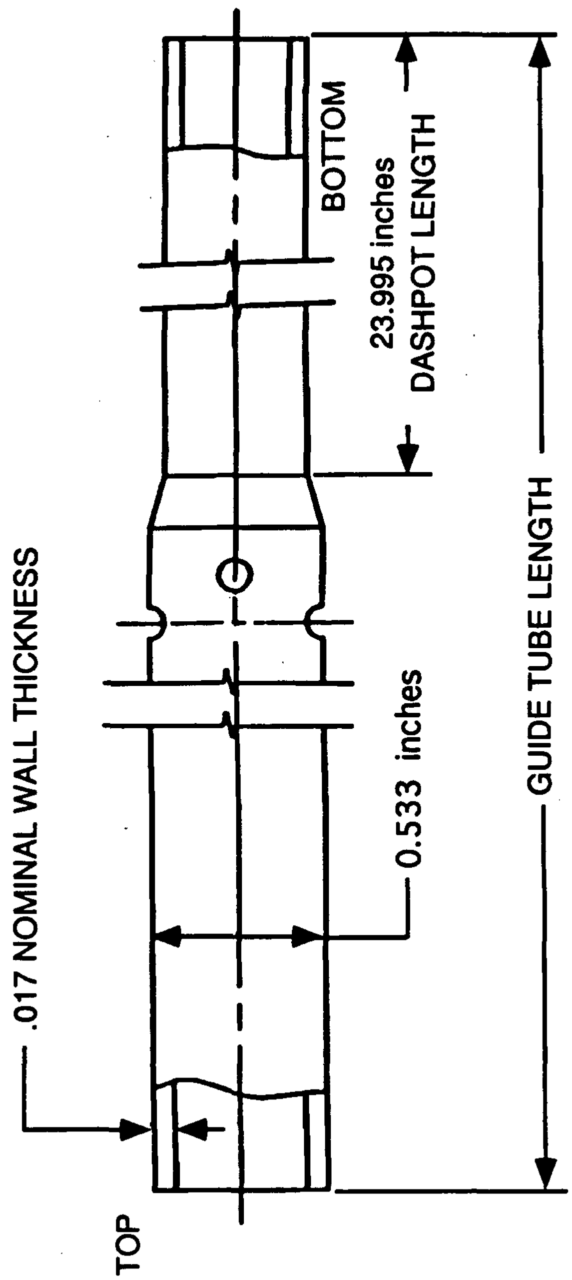
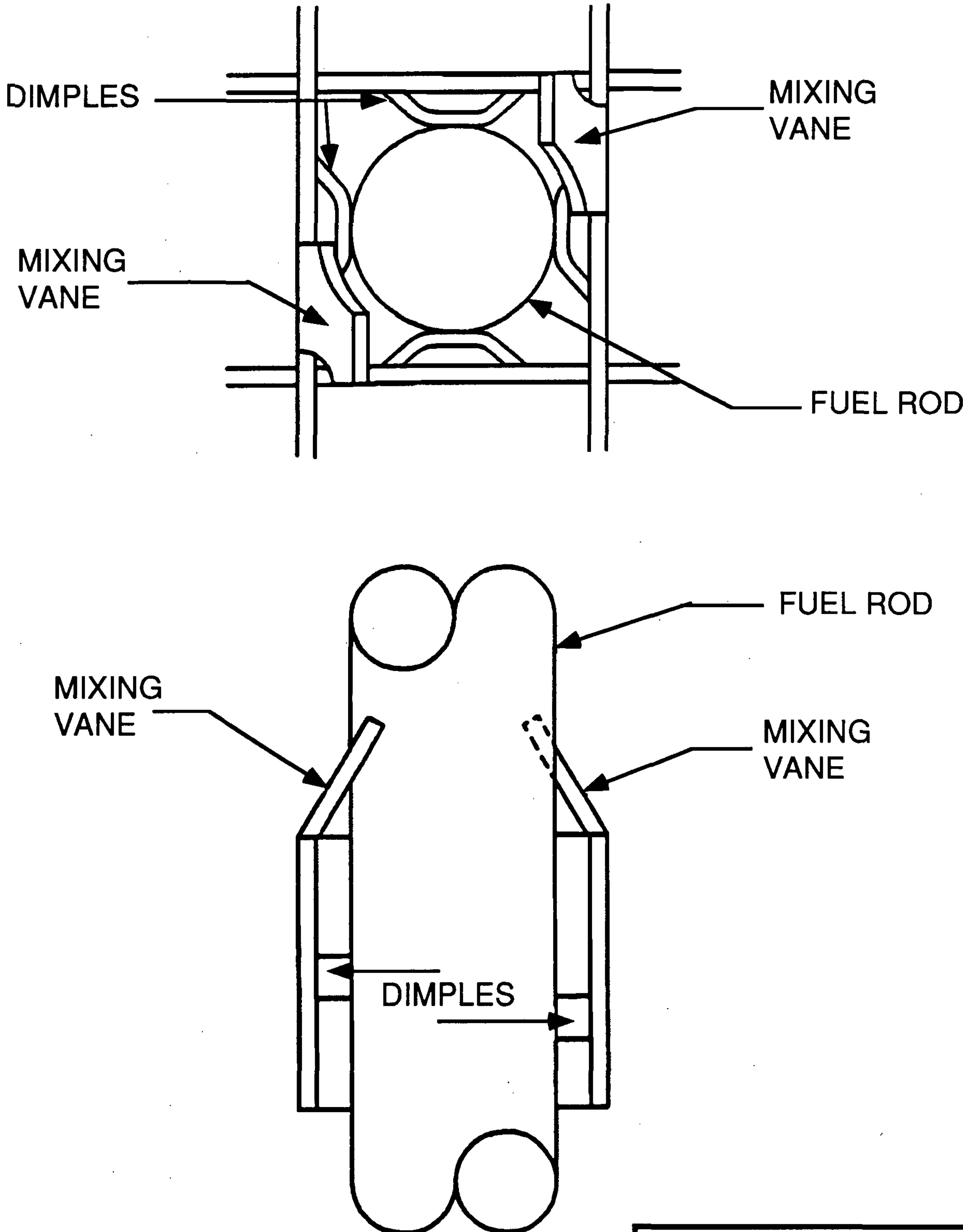


ZION STATION UFSAR

Figure 4.2-14

15X15 VANTAGE 5 / OFA GUIDE THIMBLE TUBE

JULY 1993



ZION STATION UFSAR

Figure 4.2-15

SCHEMATIC OF VANTAGE 5
IFM GRID

JULY 1993

4.3 NUCLEAR DESIGN

This section presents the nuclear characteristics of the core and an evaluation of the characteristics and design parameters which are significant to design objectives. The capability of the reactor to achieve these objectives while performing safely under normal operational modes, including both transient and steady state, is demonstrated.

4.3.1 Design Bases

4.3.1.1 Reactor Core Design

Criterion: The reactor core with its related controls and protection systems shall be designed to function throughout its design lifetime without exceeding acceptable fuel damage limits which have been stipulated and justified. The core and related auxiliary system designs shall provide this integrity under all expected conditions of normal operation with appropriate margins for uncertainties and for specified transient situations which can be anticipated.

The reactor core, with its related control and protection system, is designed to function throughout its design lifetime without exceeding acceptable fuel damage limits. The core design, together with reliable process and decay heat removal systems, provides for this capability under all expected conditions of normal operation with appropriate margins for uncertainties and anticipated transient situations. Refer to Chapter 15 for more details describing the transient situations.

The Reactor Protection System is designed to actuate a reactor trip for any anticipated combination of plant conditions, when necessary, to ensure a minimum departure from nucleate boiling ratio (DNBR) equal to or greater than the DNBR limit (see Section 4.4).

The integrity of fuel cladding is ensured by preventing excessive fuel swelling, excessive clad heating, excessive cladding stress and strain. This is achieved by designing the fuel rods so that the following conservative limits are not exceeded during normal operation or any anticipated transient condition:

1. Minimum DNBR equal to or greater than the DNBR limit
2. Fuel center temperature below the melting point of UO_2
3. Internal gas pressure less than the nominal external pressure (2250 psia), even at the end of life
4. Clad stresses less than the Zircaloy yield strength (as irradiated)
5. Clad strain less than 1%
6. Cumulative strain fatigue cycles less than 80% of design strain fatigue life (as irradiated)

The ability of fuel designed and operated to these criteria to withstand postulated normal service conditions is described in this chapter. Abnormal service conditions are shown by analyses, described in Chapter 15, to satisfy the demands of plant operation well within applicable regulatory limits.

The reactor coolant pumps provided for the plant are supplied with sufficient rotational inertia to maintain an adequate flow coastdown and prevent core damage in the event of a simultaneous loss of power to all pumps.

In the unlikely event of a turbine trip from full power without an immediate reactor trip, the subsequent reactor coolant temperature increase and volume insurge to the pressurizer results in a high pressurizer pressure trip and thereby prevents fuel damage for this transient. A loss of external electrical load of 50% of full power or less is normally controlled by rod cluster insertion coupled with a controlled steam dump to the condenser. This prevents a large temperature and pressure increase in the Reactor Coolant System and thus prevents a reactor trip. In this case, the overpower-overtemperature protection would guard against any combination of pressure, temperature, and power which could result in a DNB ratio less than the DNBR limit during the transient.

In neither the turbine trip nor the loss-of-flow events do the changes in coolant conditions provoke a nuclear power excursion because of the large system thermal inertia and relatively small void fraction. Protection circuits actuated directly by the coolant conditions identified with core limits are therefore effective in preventing core damage.

Reactivity Control Limits

The control system and the operational procedures provide adequate control of the core reactivity and power distribution. The following control limits are met:

1. A minimum hot shutdown margin is available assuming a 10% uncertainty in the control rod calculation.
2. This shutdown margin is maintained with the most reactive Rod Cluster Control Assembly (RCCA) in the fully withdrawn position.
3. The shutdown margin is maintained at ambient temperature by the use of soluble boron.

4.3.1.2 Suppression of Power Oscillations

Criterion: The design of the reactor core with its related controls and protection systems shall ensure that power oscillations, the magnitude of which could cause damage in excess of acceptable fuel damage limits, are not possible or can be readily suppressed.

The potential for possible spatial oscillations of power distribution for this core has been reviewed. It is concluded that low frequency xenon oscillations may occur in the axial dimension, but can be controlled by control rod movement. The VANTAGE 5 reload core, like the Optimized Fuel Assembly (OFA) reload core, is stable to radial x-y xenon oscillations at all times in life. Routine implementation of the low neutron leakage design in the VANTAGE 5 cores serves to make the core more stable with respect to radial power swings. This design effectively "reduces" core diameter through decreased power sharing in the peripheral assemblies. Excore instrumentation is provided to obtain necessary information concerning power distribution. This instrumentation is adequate to enable the operator to monitor and control xenon induced oscillations. (Incore instrumentation is used to periodically calibrate and verify the information provided by the excore instrumentation.) The analysis, detection, and control of these oscillations is discussed in References 2 and 3.

4.3.1.3 Redundancy of Reactivity Control

Criterion: Two independent reactivity control systems, preferably of different principles, shall be provided.

Two independent reactivity control systems are provided, one involving RCCAs and the other involving chemical shimming.

The RCCAs are divided into two categories comprising control banks and shutdown banks. The control banks, used in combination with chemical shim control, provide control of the reactivity changes of the core throughout the life of the core during power operation. These banks of RCCAs are used to compensate for short term reactivity changes at power that might be produced due to variations in reactor power level or in coolant temperature. The chemical shim control is used to compensate for the more slowly occurring changes in reactivity throughout core life such as those due to fuel depletion and fission product buildup.

4.3.1.4 Reactivity Hot Shutdown Capability

Criterion: The reactivity control systems provided shall be capable of making and holding the core subcritical from any hot standby or hot operating condition.

RCCAs provide capability, together with the dissolved boron, of making and holding the reactor subcritical by at least 1.3% $\Delta k/k$ from any mode of operation associated with the steady state, operational and anticipated transients.

4.3.1.5 Reactivity Shutdown Capability

Criterion: One of the reactivity control systems provided shall be capable of making the core subcritical under any anticipated operating condition (including anticipated operational transients) sufficiently fast to prevent exceeding acceptable fuel damage limits. Shutdown margin should assure subcriticality with the most reactive control rod fully withdrawn.

The reactor core, together with the reactor control and protection system is designed such that the minimum allowable DNBR is at least greater than or equal to the DNBR limit and there is no fuel melting during normal operation including anticipated transients.

The shutdown groups are provided to supplement the control groups of RCCAs to make the reactor at least 1.3% subcritical at the hot zero power condition (k_{eff} = 0.987) following trip from any credible operating condition assuming the most reactive RCCA is in the fully withdrawn position.

Sufficient shutdown capability is also provided to maintain the core subcritical assuming the most reactive rod to be in the fully withdrawn position for the most severe anticipated cooldown transient associated with a single active Engineered Safety Feature failure, e.g., a spurious opening, with a failure to close, of the largest of any single steam dump, relief or safety valve. This shutdown is achieved by the combination of control rods and automatic boric acid addition via the emergency core cooling system. The design minimum shutdown margin is 1.30% assuming the maximum worth control rod is in the fully withdrawn position and allowing 10% uncertainty in the control rod calculations.

Manually controlled boric acid addition is used to maintain the shutdown margin for the long term conditions of xenon decay and plant cooldown. Redundant equipment is provided to guarantee the capability of adding boric acid to the reactor coolant system.

4.3.1.6 Reactivity Holddown Capability

Criterion: The reactivity control systems provided shall be capable of making the core subcritical under credible accident conditions with appropriate margins for contingencies and limiting any subsequent return to power such that there will be no undue risk to the health and safety of the public.

Normal reactivity shutdown capability is provided by control rods within 2.4 seconds following a trip signal. Boric acid injection is used to compensate for the long term xenon decay transient and for plant cooldown. As discussed in response to the previous criteria, the shutdown capability prevents return to critical during the cooldown associated with a safety valve stuck fully open.

Any time that the reactor is at power, the quantity of boric acid retained in the boric acid tanks and ready for injection always exceeds that quantity required for the normal cold shutdown. This quantity always exceeds the quantity of boric acid required to bring the reactor to hot shutdown and to compensate for subsequent xenon decay. Boric acid is pumped from the boric acid tanks, by one of two boric acid transfer pumps, to the suction of one of three charging pumps which inject boric acid into the reactor coolant. In less than one hour, sufficient boric acid can be injected by one pump to take the plant to a 1.3% shutdown margin in the hot condition with no rods inserted at beginning-of-life (BOL). In an additional hour, enough boric acid can be injected to compensate for xenon decay, although xenon decay below the equilibrium operating level does not begin until approximately 24 hours after shutdown. Additional boric acid injection is employed if it is desired to bring the reactor to cold shutdown conditions.

On the basis of the above, the injection of boric acid affords backup reactivity shutdown capability, independent of control rod clusters which normally serve this function in the short term situation. Shutdown for long term and reduced temperature conditions can be accomplished with boric acid injection using redundant components, thus achieving the measure of reliability implied by the criterion.

Alternately, boric acid solution at lower concentration can be supplied from the refueling water storage tank. This solution can be transferred directly by the charging pumps or, alternately, by the safety injection pumps. The reduced boric acid concentration lengthens the time required to achieve equivalent shutdown.

4.3.1.7 Reactivity Control Systems Malfunction

Criterion: The reactor protection systems shall be capable of protecting against any single malfunction of the reactivity control system, such as unplanned continuous withdrawal (not ejection or dropout) of a control rod, by limiting reactivity transients to avoid exceeding acceptable fuel damage limits.

The reactor protection systems are capable of protecting against any single credible malfunction of the reactivity control system by limiting reactivity transients to avoid exceeding acceptable fuel damage limits.

Reactor shutdown with rods is completely independent of the normal rod control functions since the trip breakers completely interrupt the power to the rod mechanisms regardless of existing control signals.

4.3.1.8 Maximum Reactivity Worth of Control Rods

Criterion: Limits, which include reasonable margin, shall be placed on the maximum reactivity worth of control rods or elements and on rates at which reactivity can be increased to ensure that the potential effects of a sudden or large change of reactivity cannot (a) rupture the reactor coolant pressure boundary or (b) disrupt the core, its support structures, or other vessel internals sufficiently to lose capability of cooling the core.

Limits, which include considerable margin, are placed on the maximum reactivity worth of control rods or elements and on rates at which reactivity can be increased to ensure that the potential effects of a sudden or large change of reactivity cannot (a) rupture the reactor coolant pressure boundary or (b) disrupt the core, its support structures, or other vessel internals so as to lose the capability to cool the core.

The reactor control system employs control rod clusters. A portion of these are designated shutdown rods and are fully withdrawn during power operation. The remaining rods comprise the control groups which are used to control load and reactor coolant temperature. The rod cluster drive mechanisms are wired into preselected groups, and are therefore prevented from being withdrawn in other than their respective groups. The rod drive mechanism is of the magnetic latch type and the coil actuation is sequenced to provide variable speed rod travel. The maximum reactivity insertion rate is analyzed in the setpoint study assuming two of the highest worth groups are accidentally withdrawn at maximum speed. This yields reactivity insertion rates of the order of 8 x 10^{-4} Δk/sec, which are well within the capability of the overpower-overtemperature protection circuits to prevent core damage. To ensure this, no single credible mechanical or electrical control system malfunction can cause a rod cluster to be withdrawn at a speed greater than 72 steps per minute (45 inches per minute).

4.3.2 Description

4.3.2.1 Nuclear Design Description

A summary of the reactor nuclear design parameters is presented in Table 4.3-1. Table 4.3-4 includes the range of key safety parameters for VANTAGE 5 transition cores and OFA cores. Specific values for Zion reload cycles are calculated and reviewed prior to the cycle and are available in that cycle's Nuclear Design Report.

4.3.2.1.1 Nuclear Safety Limits

At full power (3250 MWt), the nuclear heat flux hot channel factor, F^N_Q = 2.40, is not exceeded.

For any condition of power level, coolant temperature and pressure which is permitted by the control and protection system during normal operation and anticipated transients, the hot channel power distribution is such that the minimum departure from nucleate boiling ratio (DNBR) is greater than the DNBR limit and the linear fuel rating is less than 22.6 kW/ft. For any normal steady-state operating condition, the maximum linear fuel rating does not exceed 16.4 kW/ft.

Potential axial xenon oscillations can be controlled with rods to preclude adverse core conditions. The protection system ensures that the nuclear core limits are not exceeded.

4.3.2.2 Power Distribution

In order to meet the performance objectives without violating safety limits, the peak to average power density must be within the limits set by the nuclear hot channel factors. For the peak power point in the core, the nuclear heat flux hot channel factor, F^N_Q , is 2.40. For the hottest channel, the nuclear enthalpy rise hot channel factor, $F^N_{\Delta H}$, is 1.5.

Extensive power distribution analyses have been performed for the core and support the assertion that the design objectives are achieved. Figures 4.3-1 and 4.3-2 show variation of hot channel factors for various rod positions at beginning of life for a typical VANTAGE 5 design cycle. Incore instrumentation is employed to check the power distributions throughout core lifetime.

The control system for axial power distribution control is based on manual operation of the control rods. Administrative procedures, alarm functions, and automatic rod stops guide and monitor the operator in performing these tasks. The excore Nuclear Instrumentation System (NIS) supplies the necessary information for the operator to control the core power distribution within the limits established for the protection system design. This information consists of a two-pen recorder for each power range channel which displays the upper and lower ion chamber currents and an indicator which gives the difference in these two currents for each power range channel. The ion chamber currents sent to the recorders and indicators are calibrated against incore power distribution obtained from the movable detector system so that the eight individual signals are directly related to the power generated in the adjacent section of the core. This arrangement divides the core into eight sections for monitoring purposes; four in the upper half and four in the lower half. The operator manually positions the control rods to maintain a prescribed relationship between the power generated in the upper and lower sections of the core.

The relationship between core power distribution and excore nuclear instrumentation readings is established during the startup testing program. Incore flux measurements are made over a range of axial offsets at a fairly constant power. These measurements, together with power range detector currents, will be processed to yield the relationships between core average axial power generation, the axial peaking factor, and axial offset as indicated by the excore nuclear instrumentation. This excore instrument calibration may be performed by other methods provided the incore/excore (NIS) axial-offset disagreement becomes no larger than the tolerance allowed in the station NIS calibration procedure.

These relationships are checked and recalculated during operation to assess the effect of core burnup on the sensitivity between incore power distribution and excore readings. A more detailed discussion of the analytical and experimental background data, which forms the basis for this approach, is given in Reference 2.

Operation is maintained using Constant Axial Offset Control (CAOC) (see Reference 4). This requires imposing limits on ΔI, the difference in power generated by the upper and lower half of the core. These limits have been defined as maintaining ΔI within +6 and -7% of the target value. Axial flux difference limits are defined in the Technical Specifications. Annunciators are provided in the Control Room to ensure compliance with the flux difference band operating limits specified in these Technical Specifications. A final acceptance criteria analysis is performed for each reload assuming CAOC to verify power peaking and DNB constraints. Reactivity is controlled using control rods and chemical shim.

The Technical Specifications provide for a periodic functional test of the f(ΔI) function used in the overpower and overtemperature ΔT trips.

Based upon experience obtained from operating plants, quarterly calibration of the upper and lower excore detector chambers for symmetric offset by means of the movable incore detector system is sufficient.

Quadrant Power Tilt Limits

Significant power tilts are not anticipated during normal operation since this phenomenon is caused by some asymmetric perturbation. Operational experience has shown that the normal power tilts are less than 1.01. Some of the mechanisms that can cause power tilts are:

1. dropped RCCA
2. misaligned RCCA
3. inlet temperature mismatch
4. enrichment variations within manufacturer's tolerance
5. x-y xenon transients

The Reactor Control and Protection System includes several systems to detect a dropped or misaligned RCCA. These systems perform the following:

1. Individual rod position is indicated on the control board.
2. Demanded bank position for each bank is indicated on the control board.
3. A rod bottom alarm is actuated when any rod reaches the core bottom out of sequence.
4. A rod deviation alarm is actuated when any rod deviates from the demanded bank position for its bank by a preset value.
5. A rod deviation alarm is actuated when any rod in a bank deviates from the average bank position by a preset value.

Analyses have shown (Figure 4.3-3) that the percent increase in x-y peaking is less than or equal to twice the increase in the indicated quadrant power tilt, i.e., an envelope with 2 : 1 slope. The data shown in Figure 4.3-3 is representative of many calculated x-y power distributions for several different plants where the power tilt was caused by either misaligning one rod from its bank or dropping only one rod into the core. Any dropped or misaligned rod will cause an x-y power tilt except the center rod.

The perturbation caused by the center RCCA or any other RCCA in the core will easily be seen by the Reactor Control and Protection System, thermocouple, or movable incore flux maps.

Perturbations caused by a dropped or misaligned RCCA lead to a local increase in the hot channel factor around the perturbation, without increasing the quadrant power by the same amount. For an x-y xenon transient or inlet temperature mismatch, the whole quadrant is raised or lowered in power. For these conditions, the increase in indicated tilt will match the increase in quadrant power and hence increase in radial hot channel factor, i.e. a slope of 1 : 1.

Two design limits are of concern with respect to quadrant tilts: the normal operating design peak kW/ft (F_Q) and the normal operating design enthalpy rise hot channel factor ($F_{\Delta H}$).

In the case of the F_Q limit, an off-center dropped rod will perturb the radial component of the peaking factor and, as shown in Figure 4.3-3, could cause an increase up to twice the observed quadrant tilt.

However, even with this reduced sensitivity, the tilt will be detectable before the design factor is violated. The alarm limit on tilt will be set as low as practicable (a setpoint of 1.02) and measurements will be increased by 5% (the measurements uncertainty) before comparison with the design limit. Even with the errors in the system, the design F_Q will not be violated for tilts up to and including the alarm point. This was demonstrated during the startup physics tests and included appropriate consideration of the worst possible axial profiles . These tests were accomplished during the Rod Withdrawal Tests and Rod Insertion Test at approximately 50% power during plant startup. They involved various misalignments of different control rods, selected to give the worst possible situations which could arise.

These tests verified that the excore detector system and quadrant tilt alarm are sufficiently sensitive to assure operation with power shapes not exceeding design limits.

The enthalpy rise hot channel factor ($F_{\Delta H}$), or rod integral power, is affected by both dropped rods and misaligned rods above the bank position. There is a design margin of 4% included in $F_{\Delta H}$, which is not included in F_Q, and the tests described above will verify that the design value of $F_{\Delta H}$ is not exceeded.

The above discussion refers to those misalignments which cause an indicated tilt. The special case of the center rod must also be considered. If the center rod is fully withdrawn with the remainder of bank D inserted to its limit, the hot channel power may increase in the affected region by as much as 4%. Even with this increase, the hot channel does not necessarily increase above the limit expected in normal operating situations, which is 8% below the design limit of 1.65. The margin in $F_{\Delta H}$ covers all cases of center rod misalignment with expected values not approaching closer than 5% to the design value of 1.65. In the rod withdrawn case, the peak local power density, or F_Q, will not increase since the core average axial peak will be below the bank D rods. If the center rod is inserted below the bank, beyond the part of the core that contains the peak local power, the radial power shape will increase, but it will not violate the design limit on F_Q or $F_{\Delta H}$, even for the worst permitted core average axial profile. This is true of reload cores, since it is a property of the optimum power shape and rod pattern.

For safety limits, the maximum linear power must be less than 22.6 kW/ft and DNBR must be greater than the DNBR limit. Since the normal operating peak linear power must be less than 16.4 kW/ft* the ratio of maximum power (safety limit) to normal power is approximately 1.38. Allowing the 2:1 factor described above, the quadrant tilt must be limited to 19% to avoid violating safety limits. The peak local power density is again more limiting than DNB with respect to quadrant tilt and safety limits.

Since the quadrant tilt alarm relies on four excore detectors for complete coverage, the conditions of operation with one of these out of service must be considered. The rod deviation alarm is a completely independent system for the detection of misaligned rods and would be expected to provide the first indication of misalignment. A secondary tilt monitoring system is available through the thermocouples or movable incore system. In the event that one excore detector is out of service, hand logging hourly of at least one thermocouple per quadrant or one movable incore trace per quadrant would be required to demonstrate core symmetry. The readings would be related to a datum previously determined from a known symmetric situation.

In the event that the statements made above, relative to meeting design hot channel factors with alarmed quadrant tilts, are not verified by measurements made on the operating plant, alternative courses of action must be the subject of a safety review. The solution will depend on the nature and severity of the discrepancy and should the maximum linear power exceed 16.4 kW/ft, immediate action will be taken to reduce the maximum linear power to less than 16.4 kW/ft and may dictate a temporary reduction in maximum core power.

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* The normal operating peak linear power is based on a F_Q limit of 2.40. Bounding the power is an uncertaining of 102% maximum power and a pellet stack height factor of 1.002. Max kW/ft = 2.40 x 1.02 x 1.002 x 6.7 kW/ft = 16.4 kW/ft where 6.7 kW/ft is the average nominal linear heat generation in the core.

4.3.2.3 Reactivity Coefficients

The response of the reactor core to plant conditions or operator adjustments during normal operation, as well as the response during abnormal or accidental transients, is evaluated by means of a detailed plant simulation. In these calculations, reactivity coefficients are required to couple the response of the core neutron multiplication to the variables which are set by conditions external to the core. Since the reactivity coefficients change during the life of the core, a range of coefficients is established to determine the response of the plant throughout life.

4.3.2.3.1 Moderator Temperature Coefficient

The moderator temperature coefficient relates a change in neutron multiplication to the change in reactor coolant temperature. Reactors employing soluble boron as a means of reactivity control possess less negative moderator temperature coefficients than cores controlled solely by movable or fixed RCCAs. There are two reasons for this:

1. Soluble boron density is decreased with the water density when the coolant temperature rises; and

2. In a chemical shim core, the control rods are only partially inserted. A deep insertion tends to increase the effective length of the core, thus causing the moderator coefficient to become more negative.

In order to reduce the dissolved boron requirement for control of excess reactivity, burnable absorber rods can be incorporated in the core design. The result is that changes in the coolant density will have less effect on the density of boron and the moderator temperature coefficient will be reduced.

Zion Units 1 and 2 with VANTAGE 5 fuel may use Integral Fuel Burnable Absorber (IFBA) rods and Wet Annular Burnable Absorber (WABA) rods. In the form of clusters, these rods are distributed throughout the core as illustrated in Figures 4.3-4 and 4.3-5. Information regarding research, development, and nuclear evaluation of the burnable absorber rods can be found in Reference 5. These rods initially control the installed excess reactivity and their addition results in a reduction of the initial hot full power boron concentration. The moderator temperature coefficient is more negative than the allowed limit at the operating coolant temperature with this boron concentration and with burnable absorber poison rods installed.

The effect of burnup on the moderator temperature coefficient is calculated. The coefficient becomes more negative with increasing burnup. This is due to the buildup of fission product with burnup and dilution of the boric acid concentration with burnup. The reactivity loss due to equilibrium xenon is controlled by removing boron as xenon builds up. With core burnup, the coefficient will become more negative as boron is removed because of a shift in the neutron energy spectrum due to the buildup of plutonium and fission products. Typical moderator temperature curves at beginning of life are illustrated in Figure 4.3-6.

4.3.2.3.2 Moderator Pressure Coefficient

The moderator pressure coefficient has an opposite sign to the moderator temperature coefficient. Its effect on core reactivity and stability is small because of the small magnitude of the pressure coefficient. A change of 50 psi in pressure has no more effect on reactivity than a half-degree change in moderator temperature. This coefficient can be determined from the moderator temperature coefficient by relating change in pressure to the corresponding change in density. The expected range in moderator pressure coefficient over the cycle is given in Table 4.3-1.

4.3.2.3.3 Moderator Density Coefficient

A uniform moderator density coefficient is defined as a change in the neutron multiplication per unit change in moderator density. The range of the moderator density coefficient from beginning of life (BOL) to end of life (EOL) is specified in Table 4.3-1.

4.3.2.3.4 Doppler Temperature and Power Coefficients

The Doppler coefficient of reactivity is that portion of the reactivity feedback due to temperature changes in the fuel. The Doppler temperature coefficient, pcm/°F, relates the reactivity change to the change in the average temperature of the fuel. The Doppler "only" power coefficient, $\Delta k/k/\%$ power, relates to the change in power which produced the temperature change.

The Doppler coefficient is primarily a measure of the Doppler broadening of the U-238 and Pu-240 resonance absorption. An increase in fuel temperature increases the effective resonance absorption cross-section of the fuel and produces a corresponding reduction in reactivity.

The Doppler coefficient changes as a function of core life, representing the combined effects of the fuel temperature reduction with burnup and the buildup of Pu-240.

The Doppler coefficient, in addition to including the effect of Doppler broadening on resonance captures in U-238 and Pu-240, also includes the effect of thermal expansion of the fuel and clad. Expansion effects increase the magnitude of the coefficient by about 10%.

The range of Doppler temperature and Doppler power coefficients over core life are provided in Table 4.3-4.

The Doppler temperature coefficient is shown in Figure 4.3-9.

Figure 4.3-10 shows the Doppler power coefficient as a function of power. The results presented do not include any moderator coefficient even though the moderator temperature changes with power level.

4.3.2.3.4.1 Total Power Coefficient

The total reactivity change with a change in power is defined as the total power coefficient. This coefficient as calculated at BOL under HFP critical boron and equilibrium xenon conditions is illustrated in Figure 4.3-18.

4.3.2.3.5 Total Power Reactivity Defect

Control rods must be available to compensate for the reactivity change incurred with a change in power level. The magnitude of this change has been established by correlating experimental results as presented in the Indian Point Unit 2 FSAR.

The average temperature of the reactor coolant is increased with power level in the reactor. Since this change is actually a part of the power dependent reactivity change, along with the Doppler effect and void formation, the associated reactivity change must be controlled by rods. The largest amount of reactivity that must be controlled is at the end of life when the moderator temperature coefficient has its most negative value. The moderator temperature coefficient range for a typical core is given in Table 4.3-1 while the cumulative reactivity change for both an all OFA core and a first VANTAGE 5 transition core is shown in the first line of Table 4.3-2. By the end of the fuel cycle, the nonuniform axial depletion causes a power redistribution at low power. The reactivity associated with this redistribution is part of the power defect.

4.3.2.4 Control Requirements

Reactivity control is provided by neutron absorbing control rods and by a soluble chemical neutron absorber (boric acid) in the reactor coolant. The concentration of boric acid is varied as necessary during the life of the core to compensate for: (1) changes in reactivity which occur with change in temperature of the reactor coolant from cold shutdown to the hot operating, zero power conditions; (2) changes in reactivity associated with changes in the fission product poisons (xenon and samarium); (3) reactivity losses associated with the depletion of fissile inventory and buildup of long-lived fission product poisons (other than xenon and samarium); and (4) changes in reactivity due to burnable absorbers depletion, if burnable absorbers are loaded.

The control rods provide reactivity control for: (1) fast shutdown; (2) reactivity changes associated with changes in the average coolant temperature above hot zero power (core average coolant temperature is increased with power level); (3) reactivity associated with any void formation; (4) reactivity changes associated with the power coefficient of reactivity.

4.3.2.4.1 Chemical Shim Control

Control to render the reactor subcritical at temperatures below the operating range is provided by a chemical neutron absorber (boron). The boron concentration during refueling has been established as shown in Table 4.3-1. This concentration, together with the control rods, provides approximately 5% shutdown margin for these operations. The concentration is also sufficient to maintain the core shutdown without any

RCCA rods during refueling. For cold shutdown, at the beginning of core life, the concentration (shown in Table 4.3-1) is sufficient for 1% shutdown with all but the highest worth rod inserted. The boron concentration (Table 4.3-1) for refueling is equivalent to less than 2%, by weight, boric acid (H_3BO_3) and is well within solubility limits at ambient temperature. This concentration is also maintained in the spent fuel pool since it is directly connected with the refueling canal during refueling operations.

Predicted for Cycle 13, the initial full power boron concentration without equilibrium xenon and peak samarium is 1214 ppm. As these fission product poisons are built up to their equilibrium values, the boron concentration would be reduced to 916 ppm. This initial boron concentration is that which permits the withdrawal of the control banks to their operational limits. The xenon-free hot shutdown (k = 0.987) with all but the highest worth rod inserted, can be maintained with the boron concentration less than the full-power operating value with equilibrium xenon.

4.3.2.4.2 Control Rod Requirements

Neutron-absorbing control rods provide reactivity control to compensate for more rapid variations in reactivity. The rods are divided into two categories according to their function. Some rods compensate for changes in reactivity due to variations in operating conditions of the reactor such as power or temperature. These rods comprise the control group of rods. The remaining rods, which provide shutdown reactivity, are termed shutdown rods. The total shutdown worth of all the rods is also specified to provide adequate shutdown with the most reactive rod stuck out of the core.

Control rod reactivity requirements at beginning and end of life are summarized in Table 4.3-2. The installed worth of the all OFA and first VANTAGE 5 transition core control rods is shown in Table 4.3-3.

The difference is available for excess shutdown upon reactor trip. The control rod requirements are discussed below.

4.3.2.5 Control Rod Patterns and Reactivity Worths

If sufficient boron is present in a chemically-shimmed core, the inherent operational control afforded by the negative moderator temperature coefficient is lessened to such a degree that the major control of transients resulting from load variations must be compensated for by control rods. The ability of the plant to accept major load variations is distinct from safety considerations, since the reactor would be tripped and the plant shut down safely if the rods could not follow the imposed load variations. In order to meet required reactivity ramp rates resulting from load changes, the control rods should be inserted a given distance into the core. The reactivity worth of this insertion has been defined as control rod bite.

The reactivity insertion rate must be sufficient to compensate for reactivity variation due to changes in power and temperature, caused either by a ramp load change of 5% per minute, or by a step load change of 10%. An insertion rate of 4 x 10^{-5} Δk per second is determined by the transient analysis of the core and plant to be adequate for the most adverse combinations of power and moderator coefficients. To obtain this minimum ramp rate one control bank of rods should remain partly inserted into the core. The reactivity associated with this bite is 0.04%.

The control requirements are nominally based on providing 1.3% shutdown at hot, zero power conditions with the highest worth rod stuck in its fully withdrawn position or to prevent return to criticality following a credible steamline break, whichever is the more limiting. The condition where excess reactivity insertion is most critical is at the end of a cycle when the steam break accident is considered. The excess control available at the end of cycle, hot zero power condition with the highest worth rod stuck out, allowing a 10%, margin for uncertainty in control rod worth, is shown in Table 4.3-3.

The complement of 53 full-length control rods, arranged in the pattern shown in Figure 4.3-11, meets the shutdown requirements. Table 4.3-3 lists the calculated worths of this rod configuration for beginning and end of the cycle. In order to be sure of maintaining a conservative margin between calculated and required rod worths, an additional amount has been added to account for uncertainties in the control rod worth calculations. The calculated reactivity worths listed are decreased in the design by 10% to account for any errors or uncertainties in the calculation. This worth is established for the condition that the highest worth rod is stuck in the fully withdrawn position in the core.

A comparison between calculated and measured rod worths in operating Westinghouse reactors from Shippingport through Robert E. Ginna Nuclear plants shows the calculation to be well within the allowed uncertainty of 10%.

4.3.3 Analytical Methods

The analysis methods and computer codes to be utilized by Commonwealth Edison Company in modeling VANTAGE 5 cores are described in References 12 and 13. The basis for confidence in the Commonwealth Edison procedures and neutronic design methods come from the comparison of these methods with many experimental results as documented in the Commonwealth Edison Company Topical Report on Benchmark of PWR Nuclear Design Methods, Docket No. 50-295/304, which was approved by the NRC in a Safety Evaluation Report dated December 2, 1983 (Docket No. 50-295-304). Additional confidence is obtained from the fact that the Commonwealth Edison procedures and neutronic design methods are based on the Westinghouse procedures and design methods which were validated against a number of experimental results. These experiments include criticals and other measured data from operating power reactors. A summary of the results and discussion of the agreement between calculated and measured values is given in other safety analysis reports such as the FSAR for Indian Point Unit 2, Docket No. 50-247, Section 3.2.1 and the FSAR for Zion Unit 1, Docket No. 50-295, Section 4.3.

Extensive analyses on the threshold to xenon instabilities as a function of variation in core parameters (power coefficient, etc.) have been reported in Reference 2.

4.3.3.1 Tests to Confirm Reactor Core Characteristics

A detailed series of startup physics tests are performed from zero power up to and including 100% power. As part of these tests, a series of core power distribution measurements are made over the entire range of operation in terms of control rod assembly configuration and power level by means of the incore movable detector system. These measurements are analyzed and the results compared with the analytical predictions upon which safety analyses were based.

4.3.3.2 Tests Performed During Operation

To detect and eliminate possible errors in the calculation of the initial reactivity of the core and the reactivity depletion rate, the predicted relation between fuel burnup and the boron concentration is compared to actual core conditions. This comparison is continuously updated and evaluated. Any reactivity anomaly greater than 1% would be unexpected and its occurrence would be thoroughly investigated and evaluated.

If desired, the predicted relation can be normalized to accurately reflect actual core conditions at the start of a cycle. When full power is reached initially, and with the control groups in the desired positions, the boron concentration is measured and the predicted curve is adjusted to this point. As power operation continues, the measured boron concentration is compared with the predicted concentration and the slope of the curve relating burnup and reactivity is compared with that predicted and corrected if necessary. This normalization should be completed after about 10% of the total core burnup has occurred. Thereafter, actual boron concentration can be compared with the predicted concentration, and the reactivity prediction of the core can be continuously evaluated and adjusted.

4.3.4 Changes

4.3.4.1 Onsite Review of Reload Transition to VANTAGE 5 Fuel Assemblies (From Ref. 1)

Beginning with the Cycle 13 cores, Zion Station Units 1 and 2 loaded Westinghouse 15x15 VANTAGE 5 fuel. Prior to Cycle 13, Westinghouse OFA fuel was used in the cores. As a result, transition core loadings range from approximately 50% to 70% OFA and 30% to 50% VANTAGE 5 to an all VANTAGE 5 fueled core. The 15x15 VANTAGE 5 fuel assembly was designed as a modification to the OFA design. This improved fuel design was approved by the NRC for application in the Zion Units 1 and 2 cores (Reference 15).

The Zion cores began operating with the following VANTAGE 5 features beginning with Cycle 13:

1. IFBAs;
2. Reconstitutable top nozzle;
3. Extended burnup; and
4. Axial blankets.

In addition, the Zion VANTAGE 5 fuel assemblies contain the Debris Filter Bottom Nozzle (DFBN).

Beginning with the Cycle 14 cores, the Zion VANTAGE 5 fuel assemblies will contain Intermediate Flow Mixer (IFM) grids and low pressure drop Zircaloy mid-grids.

In addition to the fuel assembly design change, the burnable absorber poison design, Wet Annular Burnable Absorber (WABA), may be utilized. Zion Station has reviewed the impact of using VANTAGE 5 and WABA for all future Zion Units 1 and 2 reloads and the required Technical Specification changes.

4.3.4.1.1 VANTAGE 5 vs OFA

The VANTAGE 5 design feature changes relative to OFA fuel, as described in Section 4.2, are summarized as follows:

1. Integral Fuel Burnable Absorber - The IFBA features a zirconium diboride coating on the fuel pellet surface on the central portion of the enriched UO_2 pellet stack. In a typical reload core, approximately one third of the fuel rods in the feed region are expected to include IFBAs. IFBAs provide power peaking and moderator temperature coefficient control.

2. Reconstitutable Top Nozzle - A mechanical disconnect feature facilitates the top nozzle removal. Changes in the design of both the top and bottom nozzles increase burnup margins by providing additional plenum space and room for fuel rod growth.

3. Extended Burnup - The VANTAGE 5 fuel design will be capable of achieving extended burnups. The basis for designing to extended burnup is contained in the approved Westinghouse extended burnup topical report (Reference 16).

4. Blankets - The axial blanket consists of a nominal six inches of natural UO_2 pellets at each end of the fuel stack to reduce neutron leakage and to improve uranium utilization. For VANTAGE 5 reload cores, low leakage loading patterns (burned blankets) are shown to further improve uranium utilization and provide additional pressurized thermal shock margin.

5. Intermediate Flow Mixer Grids - (Cycle 14 and subsequent cycles) Three IFM grids located between the three upper most Zircaloy grids provide increased DNB margin. Increased margin permits an increase in the design basis F_Q and $F_{\Delta H}$.

6. Low Pressure Drop Zircaloy Mid-Grids - (Cycle 14 and subsequent cycles) Five Zircaloy mixing vane structural grids that utilize a reduced grid height and diagonal springs to decrease the pressure drop of the mid-grids.

VANTAGE 5 fuel is mechanically compatible with OFA assemblies, reactor internals interfaces, and fuel handling and refueling equipment.

4.3.4.1.2 Nuclear Design

Evaluation of the effect of VANTAGE 5 fuel on nuclear characteristics was done by modeling the Zion core with one VANTAGE 5 region, two VANTAGE 5 regions, and all VANTAGE 5 fuel. A comparison of these results to the all OFA core shows all core nuclear characteristics differences were within the range normally seen from cycle to cycle with the exception that the VANTAGE 5 cores were designed to the increased peaking factors and higher boron concentrations as allowed by the positive moderator temperature coefficient limit (+7 pcm/°F at Hot Zero Power). The comparison was made for each of the neutronics safety parameters which are normally checked during the reload safety evaluation process. Those parameters which exceeded the OFA limits were provided as inputs to the VANTAGE 5 safety analysis. It was found that the core with one region of VANTAGE 5 fuel and the full VANTAGE 5 core bounded the two regions of VANTAGE 5 fuel core results. An example of the expected radial and axial peaking factors is given in Figure 4.3-12.

4.3.4.1.3 Thermal and Hydraulic Design

Section 4.4 describes the calculational methods used for the thermal-hydraulic analysis, the DNB performance, and the hydraulic compatibility during the transition from mixed-fuel cores to an all VANTAGE 5 core. The thermal-hydraulic design parameters for the Zion Station Units 1 and 2 used in these analyses are provided in Table 4.4-1. The thermal-hydraulic design criteria and methods used for VANTAGE 5 fuel remain the same as those for previous fuel types with the exceptions noted in Section 4.4. All of the current UFSAR thermal-hydraulic design criteria are satisfied.

4.3.4.1.4 Accident Analysis and Evaluations

All incidents analyzed and reported in the UFSAR have been reviewed by Commonwealth Edison Company for the effect of transition cores and an all VANTAGE 5 core on the results. Accounted for in the review, were four changes being made for the mixed VANTAGE 5/OFA core:

1. An increase to the Nuclear Hot Channel Factor, $F^N_{\Delta H}$, limit and Nuclear Heat Flux Hot Channel Factor, F^N_Q , including a revised K(Z) curve.

2. An increase in the moderator temperature coefficient limit to a positive +7.0 pcm/°F.

3. A change in the shutdown margin limit to 1.3% $\Delta k/k$.

4. An allowance for higher boron concentrations to accommodate the positive moderator temperature coefficient limit.

4.3.4.1.4.1 (Deleted)

4.3.4.1.4.2 (Deleted)

4.3.4.1.4.3 (Deleted)

4.3.4.1.4.4 LOCA Evaluation

The large break loss-of-coolant accident (LOCA) analysis for the Zion Station Units 1 and 2, applicable to a full core of 15x15 VANTAGE 5 fuel assemblies, was performed to develop Zion specific peaking factor limits and supports the increase in the nuclear heat flux hot channel factor, F_Q, and the nuclear enthalpy rise hot channel factor, $F_{\Delta H}$.

4.3.4.1.5 WABA

In addition to the VANTAGE 5 fuel, it may be necessary to use a discrete burnable absorber rod design. The rod design known as WABA will be utilized (Reference 14). The WABA design has annular aluminum oxide-boron carbide (Al_2O_3-B_4C) absorber pellets contained within two concentric Zircaloy tubes with water flowing through the center tube as well as around the outer tube. The WABA design satisfies all performance and design requirements for an 18,000 effective full-power hour (EFPH) irradiated life.

4.3.4.2 Fuel and Core Loading Errors

4.3.4.2.1 Incident Description

Fuel and core loading errors can arise from the inadvertent loading of one or more fuel assemblies into improper positions, loading a fuel rod during manufacture with one or more pellets of the wrong enrichment, or the loading of a full fuel assembly during manufacture with pellets of the wrong enrichment will lead to increased heat fluxes if the error results in placing fuel in core positions calling for fuel of lesser enrichment. Also included among possible core loading errors is the inadvertent loading of one or more fuel assemblies requiring burnable absorber rods into a new core without burnable absorber rods.

Any error in enrichment, beyond the normal manufacturing tolerances, can cause power shapes which are more peaked than those calculated with the correct enrichments. There is a 5% uncertainty margin included in the design value of power peaking factor assumed in the analysis of Condition I and Condition II transients. The incore system of moveable flux detectors which is used to verify power shapes at the start of life is capable of revealing any assembly enrichment error or loading error which causes power shapes to be peaked in excess of the design value.

To reduce the probability of core loading errors, each fuel assembly is marked with an identification number and loaded in accordance with a core loading diagram. After completion of core loading, the identification number will be checked on each assembly. Serial numbers read during the physical inventory are subsequently recorded on the loading diagram as a further check on proper placing after the loading is completed.

The power distortion due to any combination of misplaced fuel assemblies would significantly raise peaking factors and would be readily observable with incore flux monitors. In addition to the flux monitors, thermocouples are located at the outlet of about one-third of the fuel assemblies in the core. There is a high probability that these thermocouples would also indicate any abnormally high coolant enthalpy rise. Incore flux measurements are taken during the startup subsequent to every refueling operation.

Power distribution in the x-y plane of the core, and resulting thermal-hydraulic conditions, are analyzed with the steady state computer programs briefly discussed in Section 4.4. A discrete representation is used wherein each individual fuel rod is described by a mesh interval. The assembly-wide power distributions in the x-y plane for a correctly loaded core are also given.

For each core loading error case analyzed, the percent deviations from detector readings for a normally loaded core are shown at all incore detector locations (Figures 4.3-13 to 4.3-17).

4.3.4.2.2 <u>Cases Analyzed</u>

<u>Case A -</u>

A Region 1 assembly is interchanged with a Region 3 assembly. The particular case considered was the interchange of two adjacent assemblies near the periphery of the core (Figure 4.3-13).

<u>Case B -</u>

A Region 1 assembly is interchanged with a neighboring Region 2 fuel assembly. Two analyses have been performed for this case.

Case B-1: the interchange is assumed to take place with the burnable absorber rods transferred with the Region 2 assembly mistakenly loaded into Region 1 (Figure 4.3-14).

Case B-2: the interchange is assumed to take place closer to core center and with burnable absorber rods located in the correct Region 2 position but in a Region 1 assembly mistakenly loaded into the Region 2 position (Figure 4.3-15).

Case C -

Enrichment error: Case in which a Region 2 fuel assembly is loaded in the core central position (Figure 4.3-16).

Case D -

Case in which a Region 2 fuel assembly instead of a Region 1 assembly is loaded near the core periphery (Figure 4.3-17).

4.3.4.2.3 Conclusions

Fuel assembly enrichment errors would be prevented by administrative procedures implemented in fabrication.

In the event that a single rod or pellet has a higher enrichment than the nominal value, the consequences in terms of reduced DNBR and increased fuel and clad temperatures will be limited to the incorrectly loaded rods or pellets.

Fuel assembly loading errors are prevented by administrative procedures implemented during core loading. In the unlikely event that a loading error occurs, analyses in this section confirm that resulting power distribution effects will either be readily detected by the incore moveable detector system or will cause a sufficiently small perturbation to be acceptable within the uncertainties allowed between nominal and design power shapes.

4.3.5 References, Section 4.3

1. Letter from S.F. Stimac (CECo) to T.E. Murley (NRC) requesting changes to the Zion Units 1 and 2 Technical Specifications due to VANTAGE 5 fuel, July 10, 1991.

2. Poncelet, C.G. and Christie, A.M., "Xenon-Induced Spatial Instabilities in Large Pressurized Water Reactors," WCAP-3680-20 (1968).

3. A.M. Ross and Stoute, R.D., "Heat Transfer Coefficient Between UO2 and Zircaloy-2," AECL-1552, June 1962.

4. Westinghouse letter R.S. Grimm to C. Reed "Zion Station Constant Axial Offset Control," July 21, 1975 (CWS 75-92).

5. Wood, P.M., Bassler, E.A., et al, "Use of Burnable Poison Rods in Westinghouse Pressurized Water Reactors," WCAP-7113 (October 1967).

6. (Deleted)

7. (Deleted)

8. (Deleted)

9. (Deleted)

10. (Deleted)

11. (Deleted)

12. "Benchmark of PWR Design Methods Using the PHOENIX-P and Advanced Nodal Code (ANC)," Commonwealth Edison Company Topical Report, NFSR-0081, July 1990.

13. Letter from R.M. Polsifer (NRC) to T.J. Kovach (Commonwealth Edison), March 11, 1991, SER for "Topical Report on Benchmark of PWR Nuclear Design Methods Using PHOENIX-P and ANC Computer Codes."

14. Skaritka, J. et al., "Westinghouse Wet Annular Burnable Absorber Evaluation Report," WCAP-10377-NP-A, Revision 1, October 1983.

15. Letter from C.P. Patel (NRC) to T.J. Kovach (CECo), "Issuance of Amendments (TAC Nos. M80044, M80045, M81061, M81062, M83241, and M83242), June 29, 1992.

16. Davison, S.L. et al., "Extended Burnup Evaluation of Westinghouse Fuel," WCAP-10125-P-A, December 1985.

TABLE 4.3-1 (1 of 3)

NUCLEAR DESIGN DATA

STRUCTURAL CHARACTERISTICS	
Fuel Weight (UO_2), lbs	216,600*
Zircaloy Weight, lbs	44,547*
Core Diameter, inches	133.7*
Core Height, inches	143.4*
REFLECTOR THICKNESS AND COMPOSITION	
Top - Water Plus Steel	~10 in.*
Bottom - Water Plus Steel	~10 in.*
Side - Water Plus Steel	~15 in.*
H_2O/U, (cold) Core	4.09*
Number of Fuel Assemblies	193*
UO_2 Rods per Assembly	204*
PERFORMANCE CHARACTERISTICS	
Heat Output, MWt (initial rating)	3,250*
Equilibrium Cycle Fuel Burnup, MWD/MTU	14,300
Maximum Enrichment	4.65
Equilibrium Enrichment (Typical)	3.60
Nuclear Heat Flux Hot Channel Factor, F_Q^N	2.40
Nuclear Enthalpy Rise Hot Channel Factor, $F_{\Delta H}^N$	1.65

TABLE 4.3-1 (2 of 3)

NUCLEAR DESIGN DATA

CONTROL CHARACTERISTICS

Absorber Material	5% Cd; 15%, In; 80%, Ag*
Full Length, Number	53*
Part Length, Number	None
Number of Absorber Rods per RCC Assembly	20*
Total Rod Worth (less worst stuck rod)	(See Table 4.3-3)
Boron Concentration	
Fuel Loading Shutdown; Rods in (k_{eff} = 0.95)	2000 ppm
Rods in (k_{eff} = 1.00)	1400 ppm
To Maintain k_{eff} = 1 at Hot Full Power, No Rods Inserted:	

TABLE 4.3-1 (3 of 3)

NUCLEAR DESIGN DATA

No Xenon	1920 ppm**
Equilibrium Xenon	1600 ppm**
Shutdown, All But One Rod Inserted, Clean, Cold (k_{eff} = 0.99)	1480 ppm**
Shutdown, All Rods Out, Clean, Hot (k_{eff} = 0.987)	2200 ppm**

KINETIC CHARACTERISTICS

Moderator Temperature Coefficient at Full Power, $\frac{\Delta k}{k}$ /°F	0.0×10^{-4} to $- 3.2 \times 10^{-4}$
Moderator Pressure Coefficient, $\frac{\Delta k}{k}$ /psi	0.0×10^{-6} to 3.0×10^{-6}
Moderator Density Coefficient, $\frac{\Delta k}{k}$ /gm/cm^3	$- 1.0 \times 10^{-5}$ to $+ 0.3 \times 10^{-5}$

* Data is nonreload related
** Maximum expected with positive moderator temperature coefficient design

TABLE 4.3-2

REACTIVITY REQUIREMENTS FOR CONTROL RODS

Requirements	% Δk/k Cycle 12	% Δk/k Cycle 13
Control		
Power Defect	2.43	2.49
Control Rod Bite & Operational Maneuvering Band	0.50	0.50
Total Control	2.93	2.99

TABLE 4.3-3

CALCULATED ROD WORTHS
% (ΔK/K)

Core Condition	Rod Configuration	Worth	Less 10%,*	Design Reactivity Require-ments	Shutdown Margin
HZP Cycle 12	52 rods in; Highest Worth Rod Stuck Out	5.64	5.08	2.93	2.15
HZP Cycle 13	52 rods in; Highest Worth Rod Stuck Out	5.98	5.38	2.99	2.39**

BOL = Beginning of Life
EOL = End of Life
HZP = Hot Zero Power

* Calculated rod worth is reduced by 10% to allow for uncertainties
** The design basis minimum shutdown margin beginning with Cycle 13 is 1.30%

TABLE 4.3-4

RANGE OF KEY SAFETY PARAMETERS

Safety Parameter	Zion Station Units 1 and 2 VANTAGE 5 Transition	Zion Station Unit 1 Cycle 12 OFA
Reactor Core Power (MWt)	3250	3250
Core Average Coolant Temperature HFP (°F)	563.9	563.9
Coolant System Pressure (psia)	2250	2250
Core Average Linear Heat Rate (kW/ft)	6.70	6.70
Most Positive Moderator Temperature Coefficient (pcm/°F)	+7.0	0.0
Most Positive Moderator Density Coefficient (Δk/g/cc)	0.40	0.312
Doppler Temperature Coefficient (pcm/°F)	-0.91 to -2.9	-1.0 to -1.6
Dopper Only Power Coefficient (pcm/% Power) Least Negative	-12.6 to -7.2	-12.6 to -7.2
Doppler Only Power Coefficient (pcm/% Power) Most Negative	-25.6 to -13.2	-25.6 to -13.2
Beta-Effective	0.0044 to 0.0075	0.0045 to 0.0070
Boron Worth (pcm/ppm)	-7 to -16	-7 to -16
Shutdown Margin (% delta-rho)	1.0 to 1.3	1.0 to 1.6
Nuclear Design $F^N_{\Delta H}$	1.527*	1.435

* Limit for VANTAGE 5 fuel which bounds OFA fuel contained in the transition core.

0.916					
1.348	0.976				
1.177	1.362	0.979			
1.137	1.074	1.328	0.887		
1.170	1.347	0.996	1.322	0.960	
1.109	0.996	1.090	1.148	1.204	0.658
1.284	1.279	1.248	1.142	0.595	0.208
0.602	0.582	0.540	0.406		

AP — ASSEMBLY POWER

NOTE: NEAR BEGINNING OF LIFE, UNRODDED CORE, HOT FULL POWER, EQUILIBRIUM XENON

ZION STATION UFSAR

Figure 4.3-1

TYPICAL NORMALIZED POWER DENSITY DISTRIBUTION

JULY 1993

0.876					
1.341	0.974				
1.179	1.369	0.983			
1.125	1.071	1.334	0.885		
1.119	1.337	0.997	1.313	0.918	
1.103	0.997	1.097	1.148	1.192	0.649
1.301	1.297	1.266	1.154	0.595	0.204
0.612	0.592	0.549	0.411		

AP — ASSEMBLY POWER

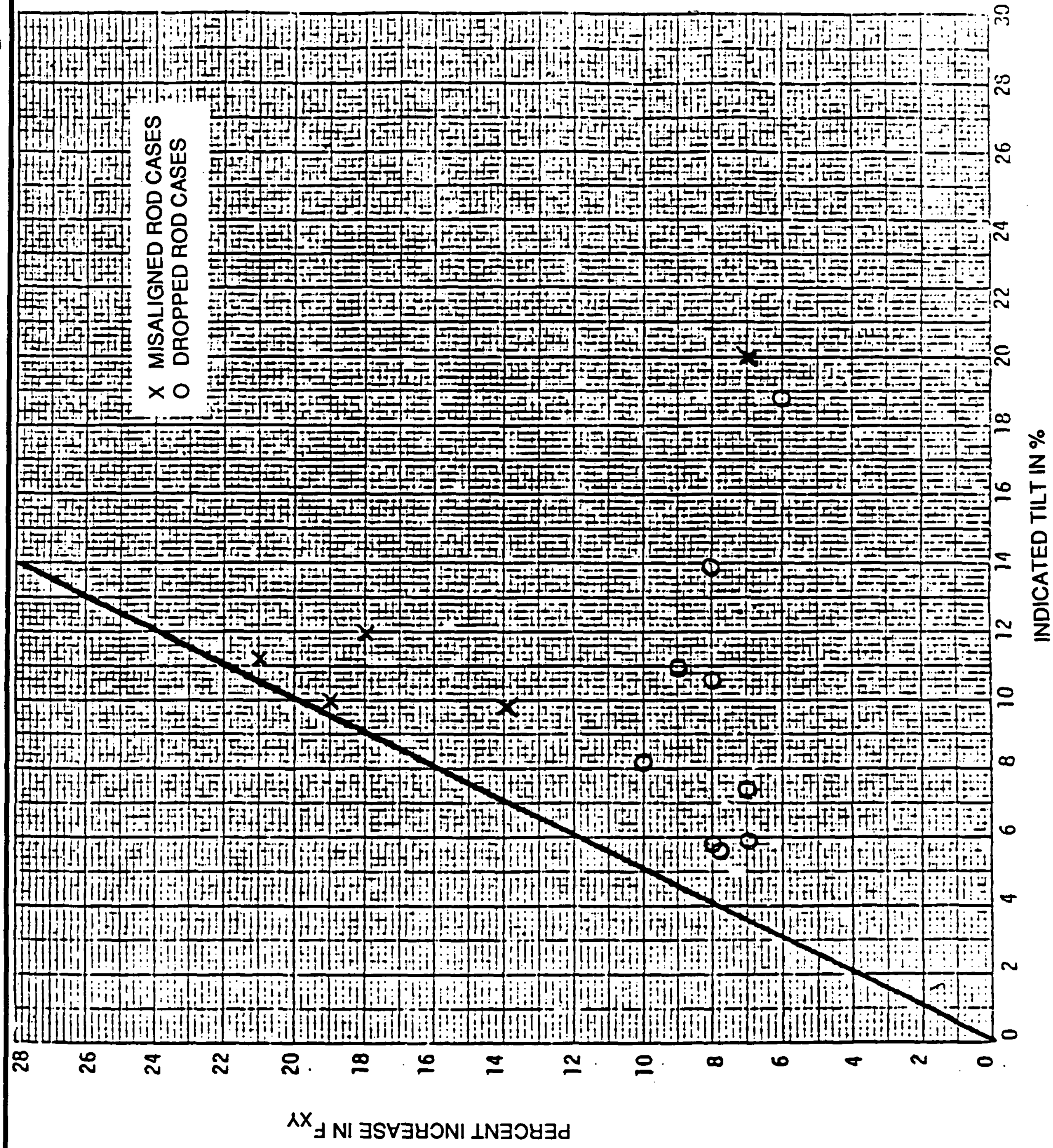
NOTE: NEAR BEGINNING OF LIFE, BANK D AT INSERTION LIMIT, HOT FULL POWER, EQUILIBRIUM XENON

ZION STATION UFSAR

Figure 4.3-2

TYPICAL NORMALIZED POWER DENSITY DISTRIBUTION

JULY 1993



ZION STATION UFSAR

Figure 4.3-3

PERCENT INCREASE IN F_{XY}
VS. INDICATED TILT

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1															
2			20FS		48I	100I	20W	80I	20W	100I	48I		20FS		
3		20FS		80I		116I		4SSA		116I		80I		20FS	
4			80I				116I		116I				80I		
5		48I				116I		20W		116I				48I	
6		100I	116I		116I		20W		20W		116I		116I	100I	
7		20W		116I		20W		100I		20W		116I		20W	
8		80I			20W		100I		100I		20W			80I	
9		20W		116I		20W		100I		20W		116I		20W	
10		100I	116I		116I		20W		20W		116I		116I	100I	
11		48I				116I		20W		116I				48I	
12			80I				116I		116I				80I		
13		20FS		80I		116I		4SSA		116I		80I		20FS	
14			20FS		48I	100I	20W	80I	20W	100I	48I		20FS		
15															

90°

0°

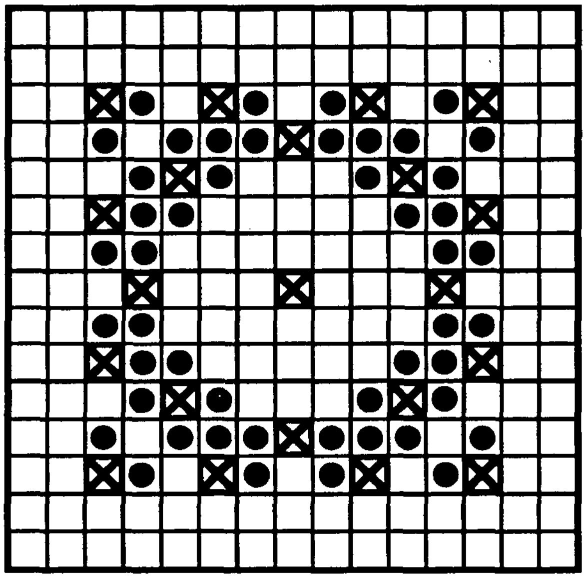
TYPE		TOTAL
##W	(NUMBER OF WABA RODLETS)	400
###I	(NUMBER OF IFBA RODS)	5328
##FS	(NUMBER OF FLUX SUPPRESSION RODLETS)	160
#SSA	(NUMBER OF SECONDARY SOURCE RODLETS)	8

ZION STATION UFSAR

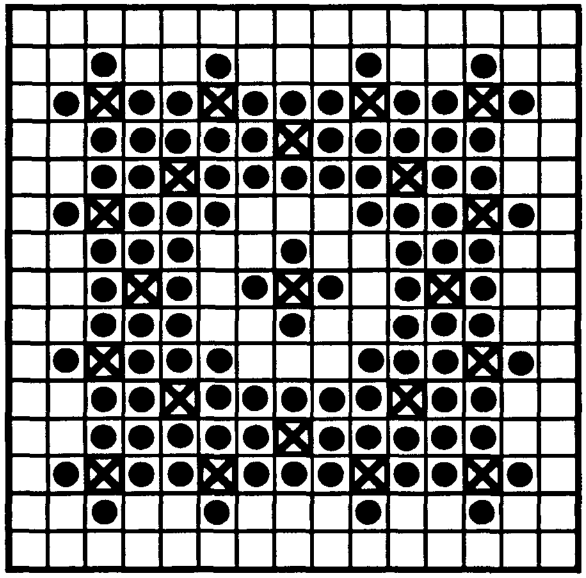
Figure 4.3-4

ZION 1 CYCLE 13 BURNABLE ABSORBER CORE LOADING PATTERN

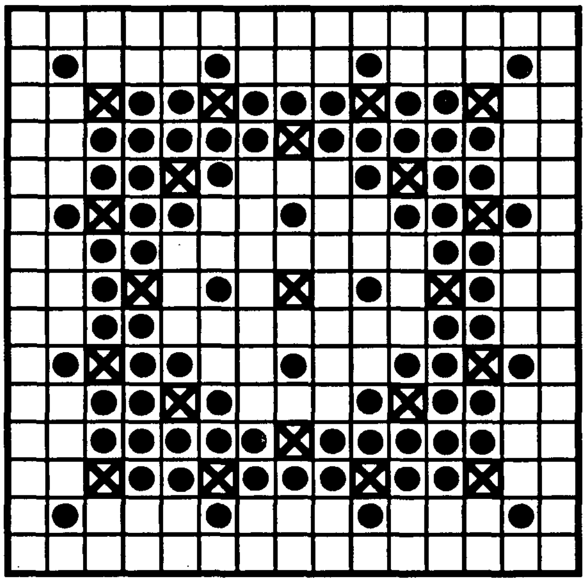
JULY 1993



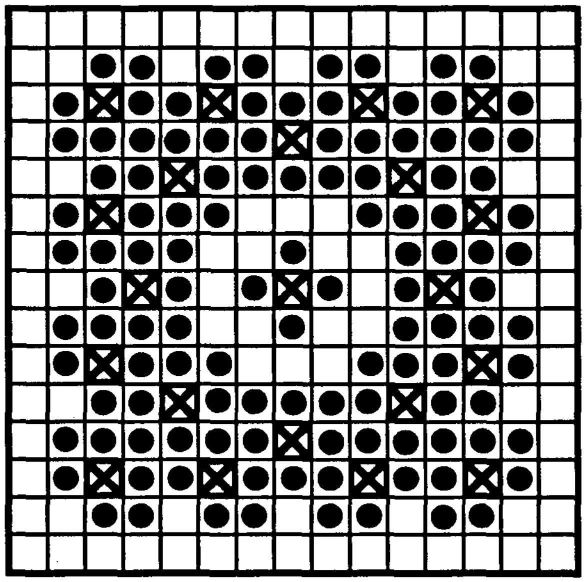
48 IFBA ASSEMBLY



100 IFBA ASSEMBLY



80 IFBA ASSEMBLY



116 IFBA ASSEMBLY

LEGEND:

☐ FUEL ROD

☒ GUIDE TUBE OR INSTRUMENTATION TUBE

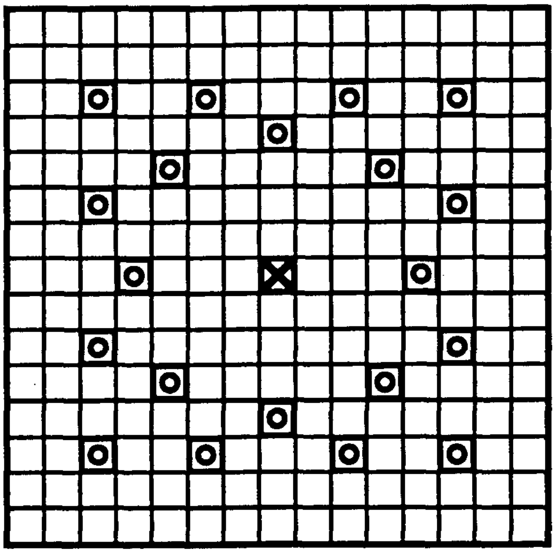
⦿ IFBA ROD

ZION STATION UFSAR

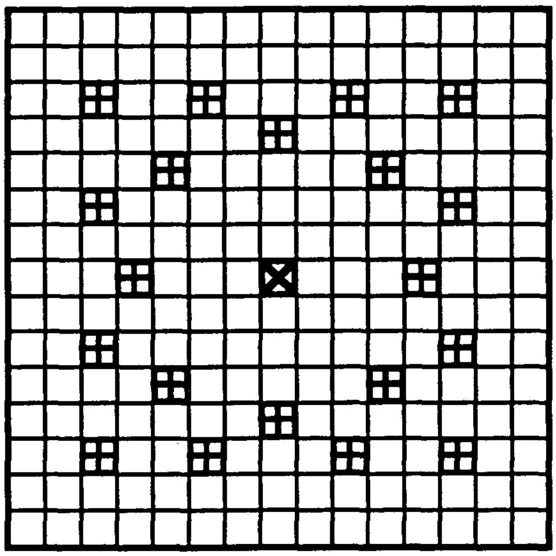
Figure 4.3-5A

ZION UNIT 1, CYCLE 13
IFBA ASSEMBLY CONFIGURATIONS

JULY 1993



20 WABA ASSEMBLY



20 FLUX SUPPRESSION ASSEMBLY

LEGEND:

FUEL ROD

INSTRUMENTATION TUBE

WABA ROD

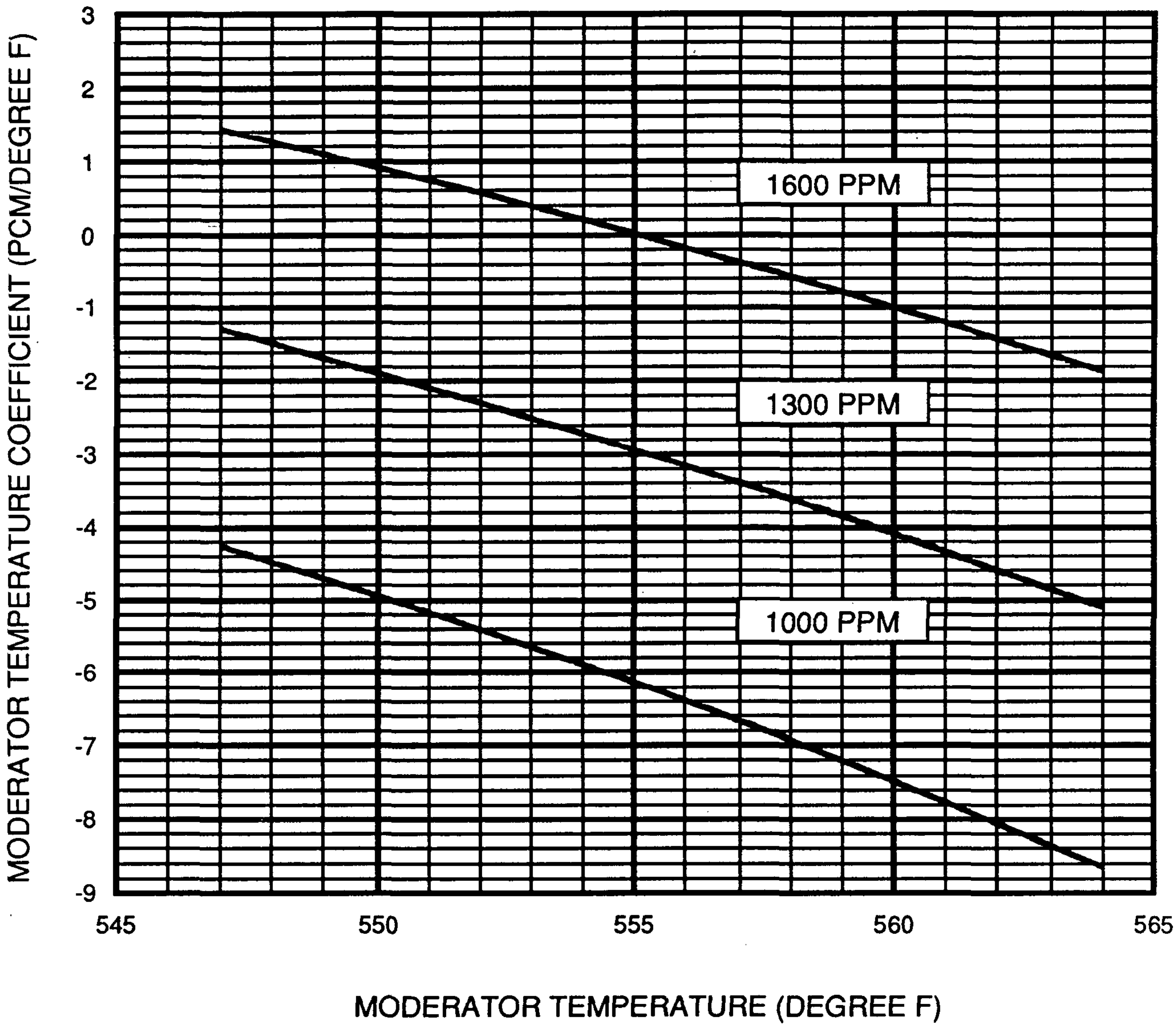
PRESSURIZED THERMAL SHOCK (PTS) ROD

ZION STATION UFSAR

Figure 4.3-5B

ZION UNIT 1, CYCLE 13
BURNABLE ABSORBER ASSEMBLY CONFIGURATION

JULY 1993



NOTE: BOL, NO RODS INSERTED,
HFP EQXE

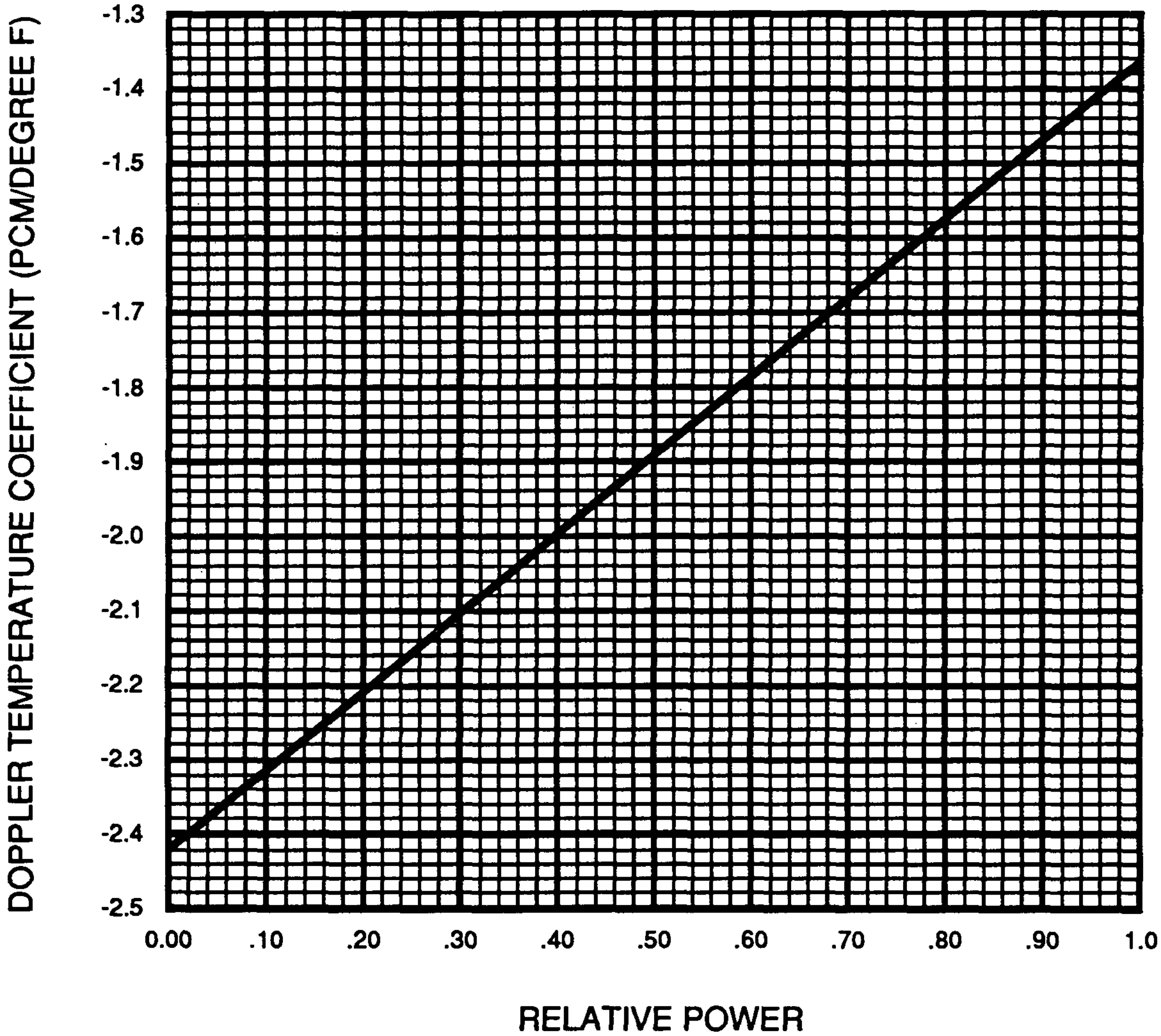
ZION STATION UFSAR

Figure 4.3-6
MODERATOR TEMPERATURE COEFFICIENT VERSUS MODERATOR TEMPERATURE

JULY 1993

- Figure 4.3-7 was Intentionally Deleted -

- Figure 4.3-8 was Intentionally Deleted -

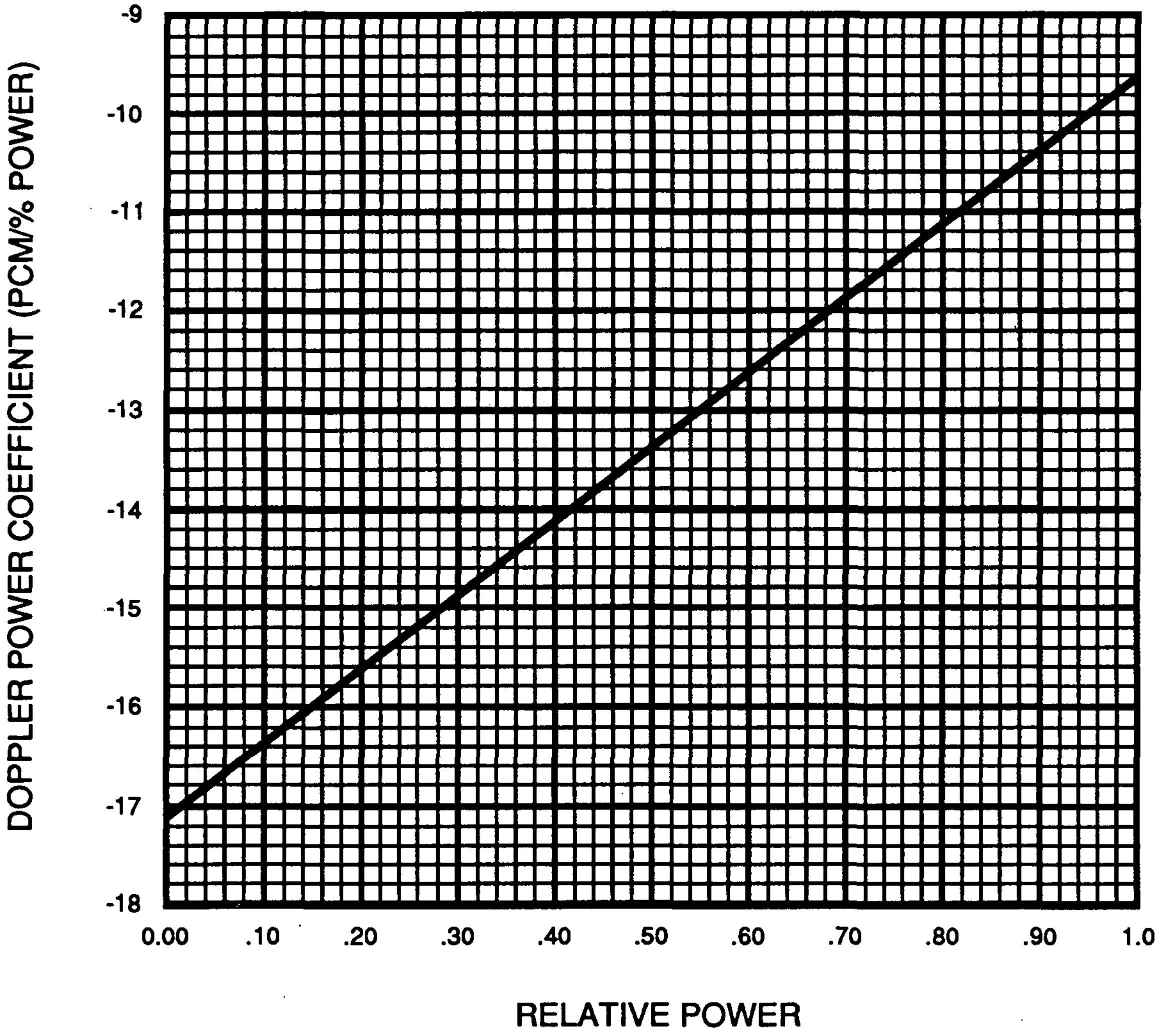


ZION STATION UFSAR

Figure 4.3-9

DOPPLER TEMPERATURE COEFFICIENT VERSUS RELATIVE POWER

JULY 1993

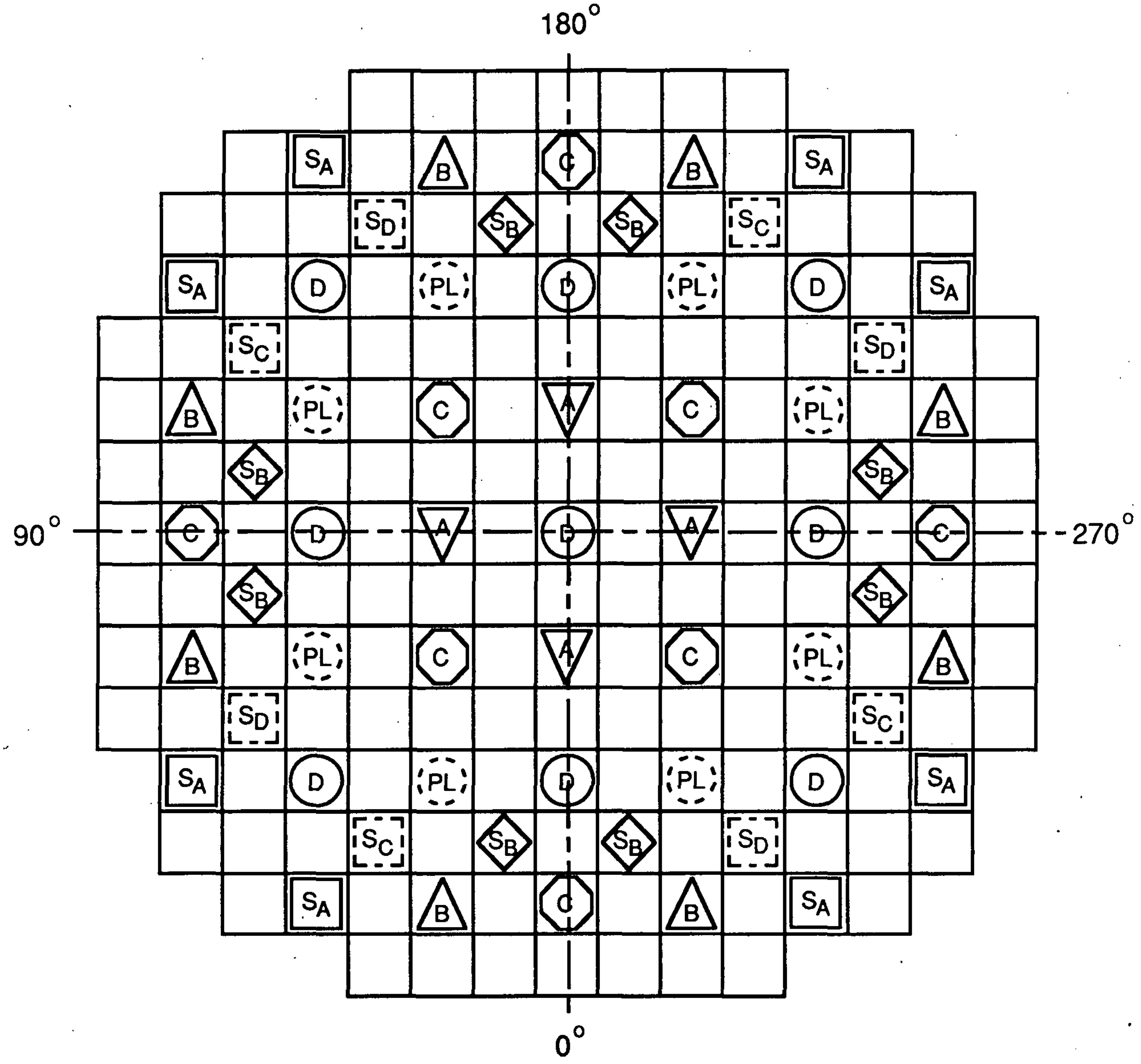


ZION STATION UFSAR

Figure 4.3-10

DOPPLER POWER COEFFICIENT VERSUS RELATIVE POWER

JULY 1993



Bank	Symbol	Number of Rod Clusters
S_A	□	8
S_B	◇	8
S_C & S_D	⬚	4 & 4
A	▽	4
B	△	8
C	⬡	8
D	○	9
PL	◌	8

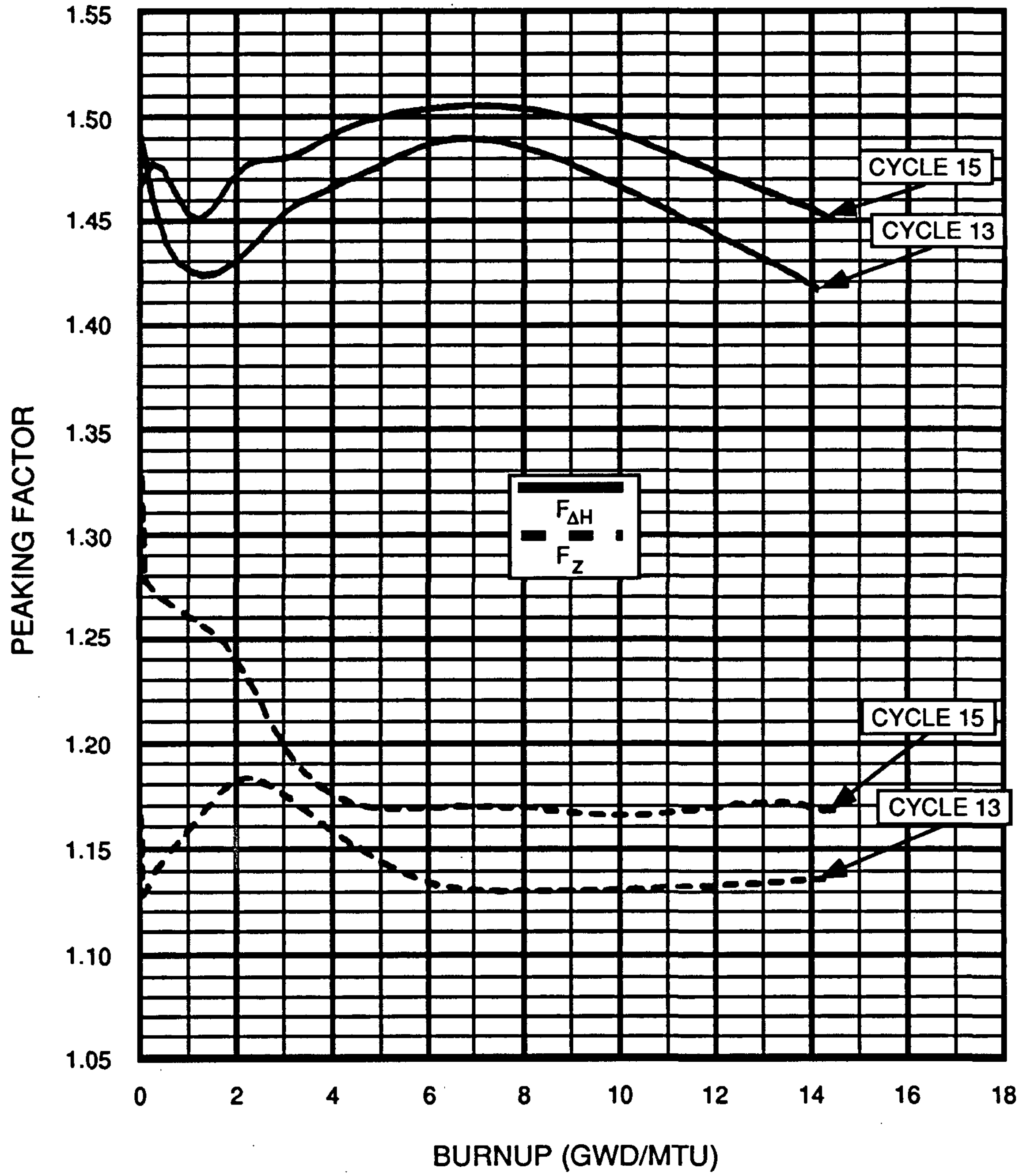
LEGEND:
S = SHUTDOWN BANK
A,B,C,D = CONTROL ROD BANKS
PL = PART-LENGTH ROD ASSEMBLIES

ZION STATION UFSAR

Figure 4.3-11

ROD CLUSTER GROUPS

JULY 1993



ZION STATION UFSAR

Figure 4.3-12

VANTAGE 5 COMPARISON OF AXIAL AND RADIAL PEAKING FACTORS

JULY 1993

4026-385

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1							-8.9			-7.4					
2			-5.6			-9.1		-8.5							
3								-8.2		-6.8		-4.1		0.2	
4		-7.9	-8.2					-7.7							
5					-8.4				-6.0		-3.8		-1.8		
6	-8.5		-8.4			-7.4		-5.5						-0.3	
7				-7.7			-5.0			-1.2			-1.0		
8	-7.7		-7.3		-5.9		-3.2			1.5		3.2	3.4	3.6	
9		-6.9							2.7		5.9				6.0
10					-3.4		0.7					10.6			
11	-5.3				-1.8			5.9			17.1				11.4
12						1.3			12.3			24.6			
13			0.1		0.7			7.7			X			23.6	
14			2.5				4.7			11.1	X	17.6			
15					2.1			6.5							

CASE A

ZION STATION UFSAR

Figure 4.3-13

INTERCHANGE BETWEEN REGION 1 AND REGION 3 ASSEMBLY

4026-386

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1					0.3			1.5							
2			3.2				0.8			3.2		6.0			
3			1.2		0.0			1.6						10.3	
4						0.0			2.9			6.5			
5	-2.2				-1.0			2.2			6.9				6.6
6					-1.7		0.5					8.8			
7		-3.2							5.2		16.7				5.4
8	-3.5		-3.4		-2.6		-0.7			11.4		11.3	5.8	4.4	
9				-3.6			-2.0			-2.3			2.2		
10	-3.8		-3.8			-3.6		-2.9						0.5	
11					-3.9				-4.3		-4.6		-1.5		
12		-2.8	-3.1					-4.5							
13								-4.8		-4.4		-2.6		1.4	
14			-0.4			-4.8		-4.8							
15							-4.8			-4.5					

CASE B-1

ZION STATION UFSAR

Figure 4.3-14

INTERCHANGE BETWEEN REGION 1 AND REGION 2 ASSEMBLY, BURNABLE POISON RODS BEING RETAINED BY THE REGION 2 ASSEMBLY

4026-387

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1							1.0			1.1					
2			5.1			1.0		1.0							
3								1.1		1.1		1.9		4.3	
4		1.7	1.7					1.4							
5					1.1				1.8		1.1		0.7		
6	0.0		0.2			1.8		3.9						4.0	
7				0.0			5.2			2.2			-0.3		
8	-0.7		-0.6		0.3		5.1			1.5		-0.3	-0.6	-0.7	
9		-1.0							-1.1		-0.8				-0.9
10					-1.4		-3.1					-1.3			
11	-0.9				-1.7						-1.7				-0.3
12						-2.5			-2.9			-1.1			
13			0.7		-1.9			-2.3						2.5	
14			2.3				-2.8			-2.4		-0.8			
15					-2.1			-2.8							

CASE B-2

ZION STATION UFSAR

Figure 4.3-15

INTERCHANGE BETWEEN REGION 1 AND REGION 2 ASSEMBLY, BURNABLE POISON RODS BEING RETAINED BY THE REGION 1 ASSEMBLY

4026-388

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1							-2.2			-2.1					
2			2.0			-2.0		-2.1							
3								-1.5		-1.6		-1.0		2.0	
4		-0.9	-1.0					-0.4							
5					-0.4				1.2		-0.5		-1.4		
6	-2.1		-1.6			2.3		5.7						-2.0	
7				-3.2			9.7			4.4			-1.7		
8	-2.3		-1.6		1.8		13.6	X		5.6		-0.4	-1.6	-2.1	
9		-2.2							9.7		1.1				-2.2
10					0.3		4.5					-0.9			
11	-1.9				-0.4			1.8			-0.5				-1.9
12						-0.9			-0.6			-1.1			
13			0.4		-1.4			-1.5						2.0	
14			2.0				-2.1			-2.0		-0.9			
15					-1.9			-2.2							

CASE C

ZION STATION UFSAR

Figure 4.3-16

ENRICHMENT ERROR: A REGION 2 ASSEMBLY LOADED INTO THE CORE CENTRAL POSITION

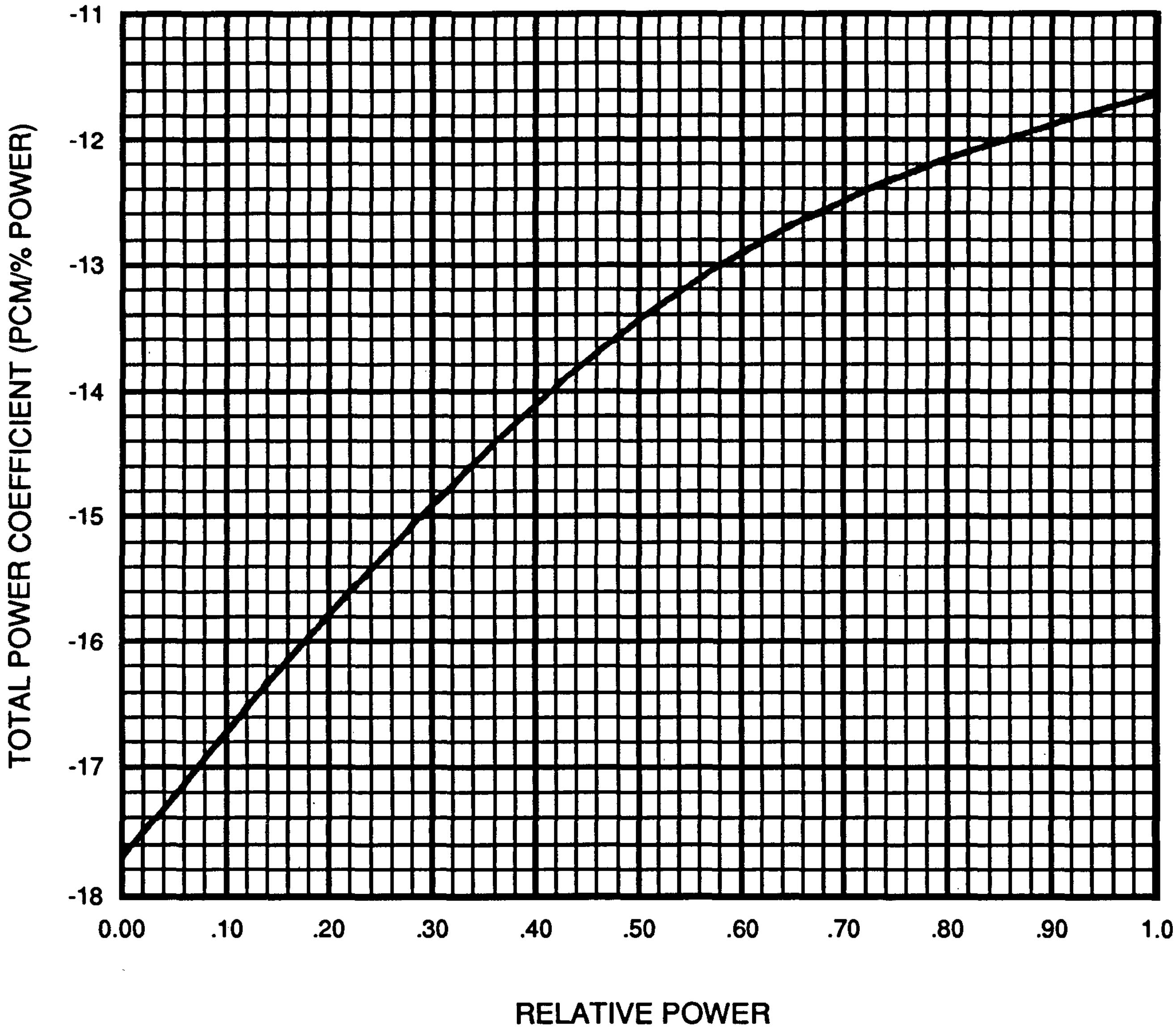
4026-389

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1							-11			-14					
2			0.4			-9.2		-12							
3								-12		-14		-15		-13	
4		3.2	1.2					-11							
5					-1.5				-12		-15		-16		
6	9.8		7.1			-1.6		-8.0						-16	
7				9.2			-2.3			-12			-14		
8	20.0		17.8		10.8		0.8			-10		-14	-15	-16	
9		27.2							-5.5		-11				-15
10					20.7		5.8					-12			
11	42.0		X		23.6			1.9			-8.6				-13
12						14.0			-1.7			-8.9			
13			38.6		20.4			2.8						-7.0	
14			35.9				7.0			-3.3		-6.3			
15					15.3			2.9							

CASE D

ZION STATION UFSAR

Figure 4.3-17
LOADING A REGION 2 ASSEMBLY INTO A REGION 1 POSITION NEAR CORE PERIPHERY



NOTE: BOL, NO RODS INSERTED,
HFP EQXE

ZION STATION UFSAR

Figure 4.3-18

TOTAL POWER COEFFICIENT VERSUS RELATIVE POWER

JULY 1993

4.4 THERMAL AND HYDRAULIC DESIGN

4.4.1 Design Basis

The overall objective of the thermal and hydraulic design of the reactor core is to provide adequate heat transfer which is compatible with the heat generation distribution in the core. In order to satisfy this requirement, the following design bases have been established for the thermal and hydraulic design of the reactor core.

4.4.1.1 Departure from Nucleate Boiling (DNB) Design Basis

4.4.1.1.1 Basis

There will be at least a 95% probability that DNB will not occur on the limiting fuel rod during normal operation and operational transients and any transient conditions arising from faults of moderate frequency (Condition I and II events) at a 95% confidence level.

4.4.1.1.2 Discussion

By preventing DNB, adequate heat transfer is assured between the fuel cladding and the reactor coolant, thereby preventing fuel damage as a result of inadequate cooling.

The design method employed to meet the DNB design basis for the VANTAGE 5 with and without Intermediate Flow Mixer (IFM) fuel assemblies is the Revised Thermal Design Procedure (RTDP), Reference 29. With the RTDP methodology, uncertainties in plant operating parameters, nuclear and thermal parameters, fuel fabrication parameters, computer codes and DNB correlation predictions are considered statistically to obtain DNB uncertainty factors. Based on the DNB uncertainty factors, RTDP design limit DNB ratio (DNBR) values are determined such that there is at least a 95% probability at a 95% confidence level that DNB will not occur on the most limiting fuel rod during normal operation and operational transients and during transient conditions arising from faults of moderate frequency (Condition I and II events). Since the parameter uncertainties are considered in determining the RTDP design limit DNBR values, the plant safety analyses are performed using input parameters at their nominal values.

The RTDP design limit DNBR values are 1.23 and 1.22 for the typical and thimble cells, respectively, for both the VANTAGE 5 with and without IFMs.

To maintain DNBR margin to offset DNB penalties such as those due to fuel rod bow and transition core, the safety analyses were performed to DNBR limits higher than the design limit DNBR values. The difference between the design limit DNBRs and the safety analysis limit DNBRs results in available DNBR margin. The net DNBR margin, after consideration of all penalties, is available for operating and design flexibility.

The Standard Thermal Design Procedure (STDP) is used for those analyses where RTDP is not applicable. In the STDP method, the parameters used in the analysis are treated in a conservative way from a DNBR standpoint. The parameter uncertainties are applied directly to the plant safety analyses input values to give the lowest minimum DNBR. The DNBR limit for STDP is the appropriate DNB correlation limit increased by sufficient margin to offset the applicable DNBR penalties.

4.4.1.2 Fuel Temperature Design Basis

4.4.1.2.1 Basis

During modes of operation associated with Condition I and II events, the maximum fuel temperature shall be less than the melting temperature of UO_2. The UO_2 melting temperature for at least 95% of the peak kW/ft fuel rods will not be exceeded at the 95% confidence level. Melting temperature of UO_2 is taken as 5080 °F unirradiated (Reference 7) and reducing 58°F per 10,000 MWD/MTU. By precluding UO_2 melting, the fuel geometry is preserved and possible adverse effects of molten UO_2 are eliminated. To preclude center melting, and as a basis for overpower protection system setpoints, a calculated centerline fuel temperature of 4700°F has been selected as the overpower limit.

4.4.1.2.2 Discussion

Fuel rod thermal evaluations are performed at rated power, maximum overpower and during transients at various burnups. These analyses assure that this design basis, as well as the fuel integrity design bases, are met. They also provide input for the evaluation of Condition III and IV faults given in Chapter 15.

4.4.2 Description of Thermal and Hydraulic Design of the Reactor Core

Table 4.4-1 provides the thermal and hydraulic design parameters for the reactor core.

4.4.2.1 Central Temperature of the Hot Pellet

The temperature distribution in the pellet is mainly a function of the uranium dioxide thermal conductivity and the local power density. The surface temperature of the pellet is affected by the cladding temperature and the thermal conductance of the gap between the pellet and the cladding.

The occurrence of nucleate boiling maintains maximum cladding surface temperature below about 660°F at nominal system pressure. The contact conductance between the fuel pellet and cladding is a function of the contact pressure and the composition of the gas in the gap (References 1 and 2) and may be calculated by the following equation:

$$h = 0.6P + \frac{k}{\delta_r}$$

where:

h	=	contact conductance in Btu/hr-ft^2-°F
P	=	contact pressure in psi
k	=	thermal conductivity of the gas mixture in the rod including a correction factor for the accommodation coefficient in BTU/hr-ft-°F
δ_r	=	effective gap spacing due to surface roughness in feet

The thermal conductivity of uranium dioxide was evaluated from data reported in Reference 3 and References 30 through 41.

At the higher temperatures, thermal conductivity is best obtained utilizing the integral conductivity to melt which can be determined with more certainty. From an examination of the data, it has been concluded that the best estimate for the value of the integral conductivity to melt is 93 W/cm. This conclusion is based on the integral values reported in Reference 5 and References 41 through 45.

The design curve for the thermal conductivity is shown in Figure 4.4-1. The section of the curve at temperatures between 0°C and 1300°C is in excellent agreement with the recommendation of the International Atomic Energy Agency (IAEA) panel (Reference 46). The section of the curve above 1300°C is derived for an integral value of 93 W/cm (References 5, 41 and 45).

Thermal conductivity for UO_2 at 95% theoretical density can be presented best by the following equation:

$$K = \frac{1}{11.8 + 0.0238T} + 8.775 \times 10^{-13}T^3$$

where:

K = W/cm-°C
T = °C

Based upon the above considerations, the maximum central temperature of the hot pellet at steady-state and overpower conditions is below the melting temperature of irradiated UO_2 which is assumed to be about 4700°F.

4.4.2.2 Critical Heat Flux Ratios

The main objective of reactor core thermal-hydraulic analysis is to verify that the fuel rods in the reactor core will not experience DNB during normal operation and anticipated transient conditions. DNB is characterized by a heat transfer condition where a sudden decrease in the heat transfer capability occurs due to a vapor blanket surrounding the fuel rod surface, causing a rapid increase in the fuel cladding temperature. DNB is of concern in design because of the possibility of fuel rod failure resulting from the increased temperature.

A design basis is established in terms of a minimum departure from nucleate boiling ratio (MDNBR) to assure that there is an adequate heat transfer between the fuel cladding and the reactor coolant. DNBR is defined as the ratio of the predicted heat flux at which DNB occurs (known as the critical heat flux (CHF)) to the local heat flux of the fuel rod. MDNBR is a figure of merit for most pressurized water reactor (PWR) transients. If the calculated MDNBR remains greater than an imposed design DNBR limit, it is assumed that there is thermal margin to the design basis. Thus, the purpose of reactor core thermal-hydraulic analysis is to accurately calculate MDNBR for assessment and quantification of the core thermal margin.

DNB is a function of hydrodynamic and heat transfer phenomena and is affected by the local and upstream conditions including the flux distribution.

In reactor design, the heat flux associated with DNB and the location of DNB are both important. The magnitude of the local fuel rod temperature after DNB depends upon the axial location where DNB occurs. The WRB-1 DNB correlation (see Reference 12) used for the VANTAGE 5 fuel analysis, incorporates local and system parameters in predicting the local DNB heat flux. This correlation includes the nonuniform flux effect, and the upstream effect which includes inlet enthalpy and length. The local DNB heat flux ratio (defined as the ratio of the DNB heat flux to the local heat flux) is indicative of the contingency available in the local heat flux without reaching DNB.

4.4.2.2.1 (Deleted)

4.4.2.2.2 W-3 DNB Correlation

The W-3 DNB correlation, References 47 and 14, is used where the primary DNBR correlation is not applicable. The WRB-1 correlation was developed based on mixing vane data and, therefore, is only applicable in the heated rod spans above the first mixing vane grid. The W-3 correlation, which does not take credit for mixing vane grids, is used to calculate DNBR values in the heated region below the first mixing vane grid. In addition, the W-3 correlation is applied in the analysis of accident conditions where the system pressure is below the range of the primary correlation. For system pressures in the range of 500 to 1000 psia, the W-3 correlation limit is 1.45 (Reference 48). For system pressures greater than 1000 psia, the W-3 correlation limit is 1.30. A cold wall factor, Reference 15, is applied to the W-3 DNB correlation to account for the presence of the unheated thimble surfaces.

4.4.2.2.2.1 Local Nonuniform DNB Flux

The WRB-1 and W-3 correlations give the equivalent uniform DNB heat flux, $q''_{DNB,EU}$, for a given set of system and local conditions. The heat distribution upstream of the DNB point affects the value of the DNB flux. This influence is accounted for by the F-factor (see Reference 9). The nonuniform DNB heat flux, $q''_{DNB,N}$, is given by

Equation (1)

$$q''_{DNB,N} = \frac{q''_{DNB,EU}}{F}$$

The DNB heat flux ratio (DNBR) is defined as

Equation (2)

$$DNBR = \frac{q''_{DNB,N}}{q''_{loc}} = \frac{q''_{DNB,EU}}{(F)\ (q''_{loc})}$$

where q''_{loc} is the actual local heat flux.

The F-factor may be considered as a hot spot factor, applicable to DNB, due to the non-uniform axial heat flux distribution.

4.4.2.2.2.2 (Deleted)

4.4.2.2.3 WRB-1 DNB Correlation

The primary DNB correlation used for the analysis of the VANTAGE 5 fuel, with and without IFMs, is the WRB-1 correlation (Reference 12).

The WRB-1 correlation was developed based exclusively on the large bank of mixing vane grid rod bundle critical heat flux data (in excess of 1100 points) that Westinghouse has collected. The WRB-1 correlation, based on local fluid conditions, represents the rod bundle data with better accuracy over a wide range of variables than the previously used W-3 correlation. This correlation accounts directly for both typical and thimble cold wall cell effects, uniform and nonuniform heat flux profiles, and variations in rod heated length and grid spacing.

The applicable range for each variable is listed in Table 4.4-4.

Figure 4.4-3 shows measured critical heat flux plotted against predicted critical heat flux using the WRB-1 correlation.

4.4.2.3 Hot Channel Factors

The total hot channel factors for heat flux and enthalpy rise are defined as the maximum-to-core average ratios of these quantities. The heat flux factors consider the local maximum at a point (the "hot spot" - maximum linear power densities), and the enthalpy rise factors consider the maximum integrated value along a channel (the "hot channel").

Each of the total hot channel factors is the product of a nuclear hot channel factor describing the neutron flux distribution and an engineering hot channel factor to allow for variations from design conditions. The engineering hot channel factors account for the effects of flow conditions and fabrication tolerances. These factors are made up of subfactors accounting for the influence of the variations of fuel pellet diameter, density and enrichment, inlet flow distribution, flow redistribution, and flow mixing.

4.4.2.3.1 Height Dependent Heat Flux Hot Channel Factor ($F_Q(Z)$)

The Height Dependent Heat Flux Hot Channel Factor is defined as the maximum local heat flux on the surface of a fuel rod at core elevation Z divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods.

4.4.2.3.2 Nuclear Heat Flux Hot Channel Factor (F_Q^N)

The Nuclear Heat Flux Hot Channel Factor is defined as the maximum local fuel rod linear power density divided by the average fuel rod linear power density, assuming nominal fuel pellet and rod dimensions.

4.4.2.3.3 Nuclear Enthalpy Rise Hot Channel Factor ($F_{\Delta H}^N$)

The Nuclear Enthalpy Rise Hot Channel Factor is a parameter that accounts for rod-to-rod variations in enrichment and density. It is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power.

4.4.2.3.3.1 Effects of Fuel Rod Bow on DNBR

The phenomenon of fuel rod bowing, as described in Reference 49, must be accounted for in the DNBR safety analysis of Condition I and Condition II events for each plant application. In the upper spans of the VANTAGE 5 with IFM assembly, additional restraint is provided with the Intermediate Flow Mixer (IFM) grids such that the grid-to-grid spacing in DNB limiting spans is approximately 13 inches compared to approximately 26 inches in the VANTAGE 5 without IFM assembly. Using the rod bow topical report methods in Reference 49 and the NRC approved scaling factor (L^2/EI) results in predicted channel closure in the limiting spans of less than 50% closure; therefore, no rod bow DNBR penalty is required in the 13 inch spans in the VANTAGE 5 safety analyses. In the lower assembly spans of the VANTAGE 5 with IFM assembly and in the VANTAGE 5 without IFM assembly, rod bow is accounted for with available DNBR margin.

The maximum rod bow penalties accounted for in the design safety analyses are based on an assembly average burnup of 24,000 MWD/MTU. At burnups greater than 24,000 MWD/MTU, credit is taken for the effect of $F_{\Delta H}$ burndown, due to the decrease in fissionable isotopes and buildup of fission product inventory. No additional rod bow penalty is required above 24,000 MWD/MTU.

4.4.2.3.4 Engineering Heat Flux Hot Channel Factor (F^E_Q)

The engineering heat flux hot channel factor is used to evaluate the maximum linear heat generation rate in the core. This subfactor is determined by statistically combining the fabrication variations for fuel pellet diameter, density, and enrichment and has a value of 1.03 at the 95% probability level with 95% confidence. As shown in Reference 50, no DNB penalty need be taken for the short, relatively low intensity heat flux spikes caused by variations in the above parameters, as well as fuel pellet eccentricity and fuel rod diameter variation.

4.4.2.3.5 Engineering Enthalpy Rise Hot Channel Factor ($F^E_{\Delta H}$)

The effect of variations in flow conditions and fabrication tolerances on the hot channel enthalpy rise is directly considered in the THINC core thermal subchannel analysis under any reactor operating condition. The items considered contributing to the enthalpy rise engineering hot channel factor are discussed below.

1. Pellet diameter, density and enrichment - Variations in pellet diameter, density, and enrichment are considered statistically in establishing the limit DNBRs for the RTDP employed in this application. Uncertainties in these variables are determined from sampling of manufacturing data.

2. Inlet Flow Maldistribution - Data from several 1/7 scale hydraulic reactor model tests indicate that a conservative design basis is to consider a 5% reduction in the flow to the hot assembly. This design basis is used in the THINC analysis.

3. Flow Redistribution - The flow redistribution accounts for the reduction in flow in the hot channel because of the local or bulk boiling. The effect of the nonuniform power distribution is inherently considered in the THINC analysis for every operating condition which is evaluated.

4. Flow Mixing - The subchannel mixing model incorporated in the THINC code and used in reactor design is based on experimental data (Reference 51). The mixing vanes incorporated in the spacer grid design induce additional flow mixing between the various flow channels in a fuel assembly, as well as between adjacent assemblies. This mixing reduces the enthalpy rise in the hot channel resulting from local power peaking of unfavorable mechanical tolerances.

4.4.2.4 Transition Core DNB Methodology

The Westinghouse transition core (OFA to VANTAGE 5) DNB methodology is given in References 52, 53, and 54. Using this methodology, transition cores are analyzed as if the entire core consisted of one assembly type (full VANTAGE 5 without IFM or full VANTAGE 5 with IFM).

The VANTAGE 5 without IFM fuel assembly has a higher mixing vane grid loss coefficient relative to the VANTAGE 5 with IFM fuel assembly mixing vane grid loss coefficient. The higher grid loss coefficient introduces localized flow redistribution from the VANTAGE 5 without IFM fuel assembly into the VANTAGE 5 with IFM fuel assembly at the lower axial zones near the mixing vane grid. The localized flow redistribution actually benefits the VANTAGE 5 with IFM fuel assembly in the lower axial zones.

The maximum transition core DNBR penalty for VANTAGE 5 without IFM fuel assembly was calculated using the methodology described in References 52 and 53. Sufficient DNBR margin is maintained in the VANTAGE 5 without IFM fuel DNBR analyses to completely offset this transition core penalty in the lower axial zones. No transition core DNBR penalty is required in the upper axial zones for VANTAGE 5 without IFM fuel assembly.

The VANTAGE 5 with IFM fuel assembly has IFM grids located in spans between mixing vane grids. No grid exists between mixing vane grids in the VANTAGE 5 without IFM fuel assembly. The additional grids introduce localized flow redistribution from the VANTAGE 5 with IFM fuel assembly into the VANTAGE 5 without IFM fuel assembly at the axial zones near the IFM grid position in a transition core. Between the IFM grids, the tendency for velocity equalization in parallel open channels causes flow to return to the VANTAGE 5 with IFM fuel assembly. The localized flow redistribution described above actually benefits the VANTAGE 5 without IFM fuel assembly in the upper axial zones.

The VANTAGE 5 with IFM transition core DNBR penalty is a function of VANTAGE 5 with IFM fuel assemblies in the core based on the methodology described in Reference 54. Sufficient DNBR margin is maintained in the VANTAGE 5 with IFM fuel DNBR analyses to completely offset the transition core penalty in the upper axial zones. No transition core DNBR penalty is required in the lower axial zones for VANTAGE 5 with IFM fuel assembly.

Therefore, according to the Westinghouse analyses discussed above, no transition core DNBR penalty is required during the transition from Optimized Fuel Assembly (OFA) fuel to VANTAGE 5 fuel.

4.4.2.5 Core Pressure Drops and Hydraulic Loads

The total pressure loss across the reactor core is listed in Table 4.4-1. These values include a 10% uncertainty factor.

4.4.3 Description of the Thermal and Hydraulic Design of the Reactor Coolant System

The thermal and hydraulic design of the Reactor Coolant System is described in Chapter 5.

4.4.4 Evaluation

4.4.4.1 Critical Heat Flux

4.4.4.1.1 (Deleted)

4.4.4.1.2 Local Nonuniform DNB Flux

The local nonuniform $q''_{DNB,N}$ is calculated as follows:

$$q''_{DNB,N} = q''_{DNB,EU}/F$$

where:

Equation (4)

$$F = \frac{C}{q''_{local}\,[1 - e^{-C\ell_{DNB,EU}}]} \int_0^{\ell_{DNB,N}} q''(z)\, e^{-C(\ell_{DNB,N} - z)}\, dz$$

ℓ_{DNB} = distance from the inception of local boiling to the point of DNB.

z = distance from the inception of local boiling measured in the direction of flow.

The empirical constant, C (see Reference 9), has been updated through the use of more recent nonuniform DNB data. However, the revised expression does not significantly influence (< 1% deviation from that of Reference 9) the value of the F-factor and the DNBR. It does provide a better prediction of the location of DNB. The new expression is

Equation (5)

$$C = 0.15\,\frac{(1 - X_{DNB})^{4.31}}{(G/10^6)^{0.478}}\ inch^{-1}$$

G = mass velocity lb/hr-ft^2

X_{DNB} = quality of the coolant at the location where DNB flux is calculated.

In determining the F-factor, the value of q''_{local} at ℓ_{DNB} in equation (4) was measured as $z = \ell_{DNB}$, the location where the DNB flux is calculated. For a uniform flux, F becomes unity so that $q''_{DNB,N}$ reduces to $q''_{DNB,EU}$ as expected. The criterion for determining the predicted location of DNB is to evaluate the ratio of the predicted DNB flux to the local heat flux along the length of the channel. The location of the minimum DNB ratio is considered to be the location of DNB.

4.4.4.1.3 Application of the WRB-1 Correlation in Design

During steady-state operation at the nominal design conditions, the DNB ratios are determined. Under other operating conditions, particularly overpower transients, more limiting conditions develop than those existing during steady-state operation. The DNB correlations are sensitive to several parameters. In addition, thermal flux generated under transient conditions is also sensitive to many parameters. Therefore, a combination of the significant parameters is used to determine design limit DNBR values. These parameters include:

1. Reactor coolant system pressure;
2. Reactor coolant system temperature;
3. Reactor power (determined from secondary plant calorimetrics); and
4. Core power distribution (hot channel factors).

For transient accident conditions where the power level, system pressure, and core temperature may increase, the DNBR is limited to a minimum value equal to the design limit DNBR as described in Section 4.4.1.1. The Reactor Control and Protection System is designed to prevent any credible combination of conditions from occurring which would result in a lower DNB ratio.

4.4.4.1.4 (Deleted)

4.4.4.1.5 Effects of DNB on Neighboring Rods

Westinghouse has never observed DNB to occur in a group of neighboring rods in a rod bundle as a result of DNB in one rod in the bundle. Westinghouse has conducted DNB with physical burnout tests in a 25-rod bundle where physical burnout occurred with one rod (see Reference 19). After this occurrence, the 25-rod test section was used for several days to obtain more DNB data from the other rods in the bundle. The burnout and deformation of the rod did not affect the performance of neighboring rods in the test section during the burnout or the validity of the subsequent DNB data points as predicted by the W-3 correlation. No occurrences of flow instability or other abnormal operation were observed.

4.4.4.1.6 DNB With Return to Nucleate Boiling

Additional DNB tests have been conducted by Westinghouse in 19- and 21-rod bundles (see Reference 20). In these tests, DNB without physical burnout was experienced more than once on single rods in the bundles for short periods of time. Each time, a reduction in power of approximately 10% was sufficient to re-establish nucleate boiling on the surface of the rod. During these and subsequent tests, no adverse effects were observed on this rod or any other rod in the bundle as a consequence of operating in DNB.

4.4.4.1.7 Hydrodynamic and Flow Power Coupled Instability

Boiling flow may be susceptible to thermo-hydrodynamic instabilities. These instabilities are undesirable in reactors since they may cause a change in thermo-hydraulic conditions that may lead to a reduction in the DNB heat flux. Two specific types of flow instabilities are considered for Westinghouse PWR operation. These are the Ledinegg or flow excursion type of static instability and the density wave type of dynamic instability.

A Ledinegg instability involves a sudden change in flowrate from one steady state to another. This instability occurs (Reference 55) when the slope of the reactor coolant system pressure drop-flowrate curve becomes algebraically smaller than the loop supply (pump head) pressure drop-flowrate curve. The Westinghouse pump head curve has a negative slope whereas the reactor coolant system pressure drop-flowrate curve has a positive slope over Condition I and II operational ranges. Thus, the Ledinegg instability will not occur.

The mechanism of density wave oscillations in a heated channel has been described by Lahey and Moody (Reference 56). A simple model has been developed by Ishii (Reference 57) for parallel, closed-channel systems to evaluate whether a given condition is stable with respect to the density wave type of dynamic instability. This method has been used to assess the stability of typical Westinghouse reactor designs under Condition I and II operation. The closed channel model is conservative relative to the parallel open-channel feature of Westinghouse PWR cores. The results indicate that a large margin to density wave instability exists.

4.4.4.2 THINC Thermal Hydraulic Analysis

The THINC computer program as approved by the NRC (References 58 and 59) is used to determine coolant density, mass velocity, enthalpy, vapor void, static pressure, and DNBR distributions along parallel flow channels within a reactor core under all expected operating conditions. The THINC code is described in detail in References 59, 60, and 61, including models and correlations used. In additoin, a disucssion on experimental verificiation of THINC is given in Reference 61. The core region being studied is considered to be made up of a number of contigious elements in a rectangular array extending the full length of the core. An element may represent any region of the core, from a single assembly to a subchannel. The THINC analysis is based on a knowledge and understanding of the heat transfer and hydrodynamic behavior of the coolant flow and mechanical characteristics of the fuel elements. The use of the THINC analysis provides a realistic evaluation of the core performance.

4.4.4.3 VIPRE-01 Thermal Hydraulic Analysis Code

Subchannel analysis has been widely used in the design and safety analysis of reactor cores. Traditionally core thermal-hydraulic analysis is performed by using a multistage method. A core analysis in which each fuel assembly is modeled as a single lumped flow channel is performed to calculate crossflow boundary conditions to be used in the subsequent subchannel analysis. In subchannel analysis, the hot assembly is modelled separately as an array of subchannels which consists of a flow channel surrounded by four fuel rods or three fuel rods and a thimble tube. The cross flow boundary condition calculated in the first stage analysis is then used in the second stage subchannel analysis to simulate the effects of the surrounding fuel assemblies on the subchannel flows. The multistage method has previously been used primarily because of limitations in computer core memory, computational speed and the thermal-hydraulic codes which allow a limited number of channels to be modelled. However, development of new and faster computers with large core memory enables the use of state-of-the-art thermal-hydraulic computer codes such as VIPRE-01 (Reference 11), with which hundreds of channels can be modelled. Thus, it is now possible to perform core thermal-hydraulic analysis in one stage using the same radial nodalization as used in the traditional multistage analyses. The accuracy of this approach has been verified through comparisons with multistage analysis.

This approach has been applied by Commonwealth Edison Company (CECo) in the development of a PWR core thermal-hydraulic analysis capability. This capability is based upon a single stage analysis using the VIPRE-01 thermal-hydraulic analysis code and the WRB-1 CHF correlation (Reference 12). The VIPRE-01 thermal-hydraulic code has been approved by the Commission for all PWR core thermal-hydraulic analyses except a loss-of-coolant accident (LOCA) (Reference 13). The use of the WRB-1 CHF correlation with the VIPRE-01 code for the analysis of the Westinghouse OFA has been approved by the Commission for use by CECo. For a detailed description of the VIPRE-01 code, see Reference 11.

4.4.5 Testing and Verification

4.4.5.1 Thermal and Hydraulic Tests and Inspections

General hydraulic tests on models have been used to confirm the design flow distributions and pressure drops (see References 25 and 26). Fuel assemblies and control and drive mechanisms are also tested in this manner. Appropriate onsite measurements are made to confirm the design flow rates. In addition, the individual analytical models in THINC Link 2, which are used for predicting the coolant conditions in the assembly by assembly analysis, were demonstrated to be conservative. The overall conservatism of the analysis was demonstrated as a portion of the Zion Startup Test Program in which assembly exit incore thermocouple measurements were compared to those predicted by THINC. When the actual tests were performed, core coolant conditions were obtained which were representative of those described in the Zion Technical Specifications. This minimized the extent of extrapolation.

4.4.6 Instrumentation Requirements

This subsection title has been created in order to implement the UFSAR format delineated by Regulatory Guide 1.70, Revision 3. However, the section is not used due to the level of detail required at the time of license application and subsequent revisions.

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TABLE 4.4-1

THERMAL AND HYDRAULIC DESIGN PARAMETERS

Total Heat Output, MWt	3250	
Total Heat Output, Btu/hr	$11{,}092 \times 10^6$	
Heat Generated in Fuel, %	97.4	
Nominal System Pressure, psia	2250	
$F^N_{\Delta H}$ (RTDP)	1.59 [1 + 0.3 (1-P)]	
(STDP)	1.65 [1 + 0.3 (1-P)]	
	RTDP	STDP
Coolant Flow		
Total Flow Rate, 10^6 lbs/hr	137.6	135.0
Average Mass Velocity, 10^6 lb/hr-ft^2	2.52	2.43
Coolant Temperature, °F		
Design Nominal Inlet	530.7	530.2
Average Rise in Vessel	63.0	64.1
Average Rise in Core	66.4	68.5
Average in Core	565.2	565.9
Average in Vessel	562.2	562.2

Heat Transfer		
Active Heat Transfer Surface Area, ft^2		52,089
Average Heat Flux, Btu/hr-ft^2		207,410
Peak Linear Power for Normal Operation, kW/ft+		16.1
Maximum Clad Surface Temperature BOL at Nominal Pressure, °F		~ 660
Fuel Central Temperatures for nominal fuel rod dimensions, °F		
Maximum at 100% Power		3900
Maximum at Over Power		< 4700
DNB Ratio		
Minimum DNB Ratio at nominal RTDP conditions		
Typical Flow Channel	OFA, VANTAGE 5 without IFMs	2.53
	VANTAGE 5 with IFMs	> 2.53
Thimble (Cold Well) Flow Channel	OFA, VANTAGE 5 without IFMs	2.39
	VANTAGE 5 with IFMs	> 2.39
Pressure Drop, psi*		
Across Core	OFA, VANTAGE 5 without IFMs	26.4 ± 2.6
	VANTAGE 5 with IFMs	26.8 ± 2.7

+ Based on 2.40 F_Q peaking factor
* Based on 379,200 GPM

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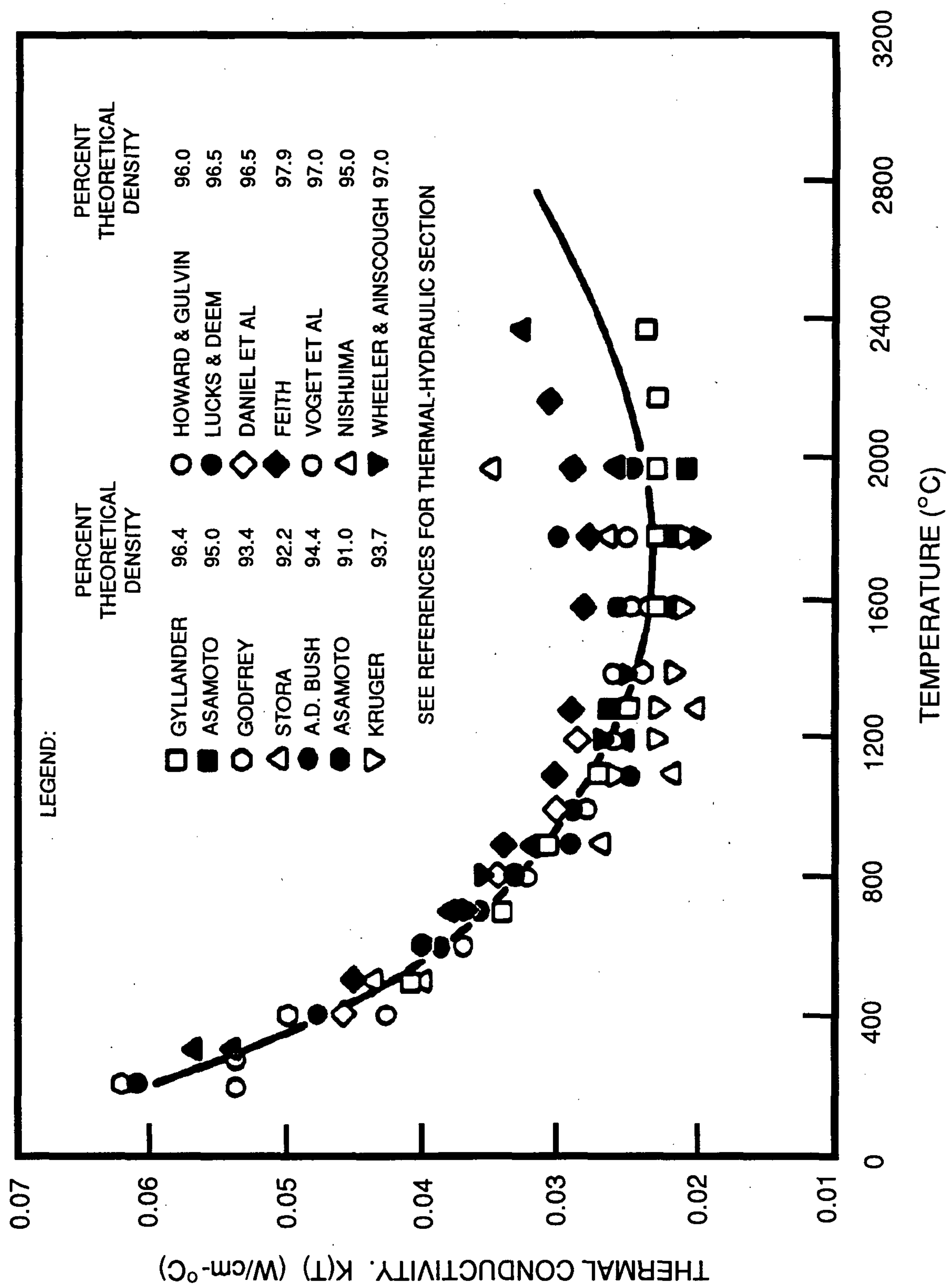

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Figure 4.4-1
THERMAL CONDUCTIVITY OF UO_2
(DATA CORRELATED TO 95%
THEORETICAL DENSITY)

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4.5 REACTOR MATERIALS

4.5.1 Control Rod Drive System Structural Materials

The control rods, or rod cluster control assemblies (RCCAs) each consist of a group of individual absorber rods fastened at the top end to a common hub or spider assembly. These assemblies, which are shown in Figures 4.5-1 and 4.5-2, are provided to control the reactivity of the core under operating conditions. RCCA specifications are in Table 4.2-1.

The absorber material used in the control rods is silver-indium-cadmium alloy which is essentially "black" to thermal neutrons and has sufficient additional resonance absorption to significantly increase its worth. The alloy is in the form of extruded single length rods which are sealed in stainless steel tubes to prevent the rods from coming in direct contact with the coolant.

The overall control rod length is such that when the assembly has been withdrawn through its full travel, the tip of the absorber rods remain engaged in the guide thimbles so that alignment between rods and thimbles is always maintained. Since the rods are long and slender, they are relatively free to conform to any small misalignments with the guide thimble. Prototype tests have shown that the RCCAs are very easily inserted and not subject to binding even under conditions of severe misalignment.

The spider assembly is in the form of a center hub with radial vanes supporting cylindrical fingers from which the absorber rods are suspended. Handling detents, and detents for connection to the drive shaft, are machined into the upper end of the hub. A spring pack is assembled into a skirt integral to the bottom of the hub to stop the RCCA and absorb the impact energy at the end of a trip insertion. The radial vanes are joined to the hub, and the fingers are joined to the vanes by furnace brazing. A centerpost which holds the spring pack and its retainer is threaded into the hub within the skirt and welded to prevent loosening in service. All components of the spider assembly are made from Type 304 stainless steel except for the springs which are Inconel X-750 alloy and the retainer which is of 17-4 PH material.

The absorber rods are secured to the spider so as to assure trouble-free service. The rods are first threaded into the spider fingers and then pinned to maintain joint tightness, after which the pins are welded in place.

In construction, the silver-indium-cadmium rods are inserted into coldworked stainless steel tubing which is then sealed at the bottom and the top by welded end plugs. The bottom plugs are made bullet-nosed to reduce the hydraulic drag during a reactor trip and to guide smoothly into the dashpot section of the fuel assembly guide thimbles. The upper plug is threaded for assembly to the spider and has a reduced end section to permit flexing of the rods to correct for small operating and assembly misalignments. Sufficient diametral and end clearances are provided to accommodate relative thermal expansions and to limit the internal pressure to acceptable levels.

Stainless steel clad silver-indium-cadmium alloy absorber rods are resistant to radiation and thermal damage thereby ensuring their effectiveness under all operating conditions. Rods of similar design have been successfully used in the Saxton, SELNI and Indian Point 1 reload core.

In March of 1988, several original design RCCAs were replaced with an enhanced performance RCCA manufactured by Westinghouse. Although the basic design of the assemblies did not change from the description above, there are three changes.

The first was to utilize a high purity stainless steel in the manufacture of the absorber clad. This steel reduces the cobalt content by approximately 38% over the original design. The end result is to reduce the likelihood that the rod will fail due to irradiation-assisted stress-corrosion cracking.

The second change was to add a very thin coat of chrome plating on the absorber clad. This plating is expected to significantly reduce fretting wear on the rods from the guide tube guide cards. The coating thickness ranges from 0.2 to 0.75 mils and the subsequent increase in the outside diameter of the control rod is within original design specifications. Therefore, there is no concern that the rod could hang up in the guide tube upon initiation of a reactor trip.

The final change was to increase the diametral gap between the absorber material and the cladding material at the lower extremity of the rodlets. This gap was increased by reducing the absorber diameter. The gap, which is increased by more than a factor of two, is provided in the region of highest neutron fluence which the control rod experiences in service, in order to minimize absorber-cladding interaction and mitigate absorber induced strain of the rodlet tubing. The change in reactivity worth due to the reduction in tip material is negligible, and well within the calculation uncertainties of the original design.

Because the new RCCAs satisfy all functional criteria of the original RCCAs, any combination of new and old RCCAs can be used at any time in the reactors.

4.5.1.1 Full-Length Control Rod Drive Mechanism Design Description

The control rod drive mechanisms are used for withdrawal and insertion of the rod cluster control assemblies into the reactor core and to provide sufficient holding power for stationary support. Fast total insertion (reactor trip) is obtained by simply removing the electrical power allowing the rods to fall by gravity.

The complete drive mechanism, shown in Figure 4.5-3, consists of the internal (latch) assembly, the pressure vessel, the operating coil stack, the drive shaft assembly, and the rod position indicator coil stack.

Each assembly is an independent unit which can be dismantled or assembled separately. Each mechanism pressure housing is threaded onto an adaptor on top of the reactor pressure vessel and seal welded. The operating drive assembly is connected to the control rod (directly below) by means of a grooved drive shaft. The upper section of the drive shaft is suspended from the working components of the drive mechanism. The drive shaft and control rod remain connected during reactor operation, including tripping of the rods.

Main coolant fills the pressure containing parts of the drive mechanism. All moving components and the shaft are immersed in the main coolant.

Three magnetic coils, which form a removable electrical unit and surround the rod drive pressure housing, induce magnetic flux through the housing wall to operate the working components. The magnets move two sets of latches which lift, lower and hold the grooved drive shaft. The three magnets are turned on and off in a fixed sequence by solid-state switches for the full length rod assemblies. The sequencing of the magnets produces step motion over the 144 inches of normal control rod travel.

The mechanism develops a lifting force approximately two times the static lifting load. Therefore, extra lift capacity is available for overcoming mechanical friction between the moving and the stationary parts. Gravity provides the drive force for rod insertion and the weight of the whole rod assembly is available to overcome any resistance.

The mechanisms are designed to operate in water at 650°F and 2485 psig. The temperature at the mechanism head adaptor will be much less than 650°F because it is located in a region where there is limited flow of water from the reactor core, while the pressure is the same as in the reactor pressure vessel.

A multiconductor cable connects the mechanism operating coils to the 125-Vdc power supply. The dc power supply is used only during maintenance operations. The RCCA ac power supply is described in Section 7.7.1.1.1.2.

4.5.1.1.1 Latch Assembly

The latch assembly contains the working components which withdraw and insert the drive shaft and attached control rod. It is located within the pressure housing and consists of the pole pieces for three electromagnets. The electromagnets actuate two sets of latches which engage the grooved section of the drive shaft. The upper set of latches move up or down to raise or lower the drive rod by $^{5}/_{8}$ inch. The lower set of latches have a maximum $^1/_{16}$-inch axial movement to shift the weight of the control rod from the upper to the lower latches.

4.5.1.1.2 Pressure Vessel

The pressure vessel consists of the pressure housing and rod travel housing. The pressure housing is the lower portion of the vessel and contains the latch assembly. The rod travel housing is the upper portion of the vessel. It provides space for the drive shaft during its upward movement as the control rod is withdrawn from the core.

4.5.1.1.3 Operating Coil Stack

The operating coil stack is an independent unit which is installed on the drive mechanism by sliding it over the outside of the pressure housing. It rests on a pressure housing flange without any mechanical attachment and can be removed and installed while the reactor is pressurized.

The three operating coils are made of round copper wire which is insulated with a double layer of filament type glass yarn. The design operating temperature of the coils is 200°C. Average coil temperature can be determined by resistance measurement. Forced air cooling along the outside of the coil stack maintains a coil casing temperature of approximately 120°C or lower.

4.5.1.1.4 Drive Shaft Assembly

The main function of the drive shaft is to connect the control rod to the mechanism latches. Grooves for engagement and lifting by the latches are located throughout the 144 inches of control rod travel. The grooves are spaced $^5/_8$ inch apart to coincide with the mechanism step length and have 45° angle sides.

The drive shaft is attached to the control rod by the coupling. The coupling has two flexible arms which engage the grooves in the spider assembly.

A $^1/_4$-inch diameter disconnect rod runs down the inside of the drive shaft. It utilizes a locking button at its lower end to lock the coupling and control rod. At its lower end, there is a disconnect assembly. For remote disconnection of the drive shaft assembly from the control rod, a button at the top of the drive rod actuates the connect/disconnect assembly. The drive shaft assembly can be attached and removed from the control rod only when the reactor vessel head is removed.

4.5.1.1.5 Position Indicator Coil Stack

The position indicator coil stack slides over the rod travel housing section of the pressure vessel. It detects drive rod position by means of cylindrically wound differential transformer which spans the normal length of the rod travel (144 inches).

4.5.1.1.6 Drive Mechanism Materials

All parts exposed to reactor coolant, such as the pressure vessel, latch assembly and drive rod, are made of metals which resist the corrosive action of the water.

Three types of metals are used exclusively: stainless steels, Inconel X-750, and cobalt based alloys. Wherever magnetic flux is carried by parts exposed to the main coolant, 400 series stainless steel is used. Cobalt based alloys are used for the pins, latch tips, and bearing surfaces.

The control rod drive shaft material is non-nitrided, non-heat treated 410 stainless steel.

Inconel X-750 is used for the springs of both latch assemblies and 304 stainless steel is used for all pressure containment. Hard chrome plating provides wear surfaces on the sliding parts and prevents galling between mating parts (such as threads) during assembly.

Outside of the pressure vessel, where the metals are exposed only to the reactor plant containment environment and cannot contaminate the main coolant, carbon and stainless steels are used. Carbon steel, because of its high permeability, is used for flux return paths around the operating coils. It is zinc-plated 0.001-inch-thick to prevent corrosion.

4.5.1.1.7 Principles of Operation

The drive mechanisms shown schematically in Figure 4.5-4 withdraw and insert their respective control rods as electrical pulses are received by the operating coils.

ON and OFF sequence, repeated by switches in the power programmer, causes either withdrawal or insertion of the control rod. Position of the control rod is indicated by the differential transformer action of the position indicator coil stack surrounding the rod travel housing. The differential transformer output changes as the top of the ferromagnetic drive shaft assembly moves up the rod travel housing.

Generally, during plant operation, the drive mechanisms hold the control rods withdrawn from the core in a static position, and only one coil, the stationary gripper coil is energized on each mechanism.

4.5.1.1.7.1 Control Rod Withdrawal

The control rod is withdrawn by repeating the following sequence:

1. Movable Gripper Coil - ON

   The movable gripper armature raises and swings the movable gripper latches into the drive shaft groove.

2. Stationary Gripper Coil - OFF

   Gravity causes the stationary gripper latches and armature to move downward until the load of the drive shaft is transferred to the movable gripper latches. Simultaneously, the stationary gripper latches then swing out of the shaft groove.

3. Lift Coil - ON

   The $^5/_8$-inch gap between the lift armature and the lift magnet pole closes and the drive rod raises one step length.

4. Stationary Gripper Coil - ON

   The stationary gripper armature raises and closes the gap below the stationary gripper magnetic pole, swinging the stationary gripper latches into a drive shaft groove. The latches contact the shaft and lift it $^1/_{16}$ inch. The load is transferred from the movable to the stationary gripper latches.

5. Movable Gripper Coil - OFF

   The movable gripper armature separates from the lift armature under the force of the spring and gravity. Three links, pinned to the movable gripper armature, swing the three movable gripper latches out of the groove.

6. Lift Coil - OFF

   The gap between the lift armature and the lift magnet pole opens. The movable gripper latches drop $^5/_8$ inch to a position adjacent to the next groove.

4.5.1.1.7.2 <u>Control Rod Insertion</u>

The sequence for control rod insertion is similar to that for control rod withdrawal:

1. Lift Coil - ON

   The movable gripper latches are raised to a position adjacent to a shaft groove.

2. Movable Gripper Coil - ON

   The movable gripper armature raises and swings the movable gripper latches into a groove.

3. Stationary Gripper Coil - OFF

   The stationary gripper armature moves downward and swings the stationary gripper latches out of the groove.

4. Lift Coil - OFF

   Gravity and spring force separates the lift armature from the lift magnet pole and the control rod drops down $^5/_8$ inch.

5. Stationary Gripper Coil - ON

6. Movable Gripper Coil - OFF

The sequences described above are termed as one step or one cycle and the control rod moves $^5/_8$ inch for each cycle. Each sequence can be repeated at a rate of up to 72 steps per minute and the control rods can therefore be withdrawn or inserted at a rate of up to 45 inches per minute.

4.5.1.1.7.3 Control Rod Tripping

During operations, the control rod position is held with the stationary gripper coil. Removing current to the stationary gripper coil would open the stationary latches allowing the control rods to fall. If power to the movable gripper coil is cut off while rods are in motion, as for tripping, the combined weight of the drive shaft and the rod cluster control assembly is sufficient to move the latches out of the shaft groove. The control rod falls by gravity into the core. The tripping occurs as the magnetic field, holding the movable gripper armature against the lift magnet, collapses and the movable gripper armature is forced down by the weight acting upon the latches.

4.5.2 Reactor Internals Materials

The reactor internal components are designed to withstand the stresses resulting from startup, steady state operation with any number of pumps running, and shutdown conditions. No damage to the reactor internals occurs as a result of loss of pumping power.

Lateral deflection and torsional rotation of the lower end of the core barrel is limited to prevent excessive movements resulting from seismic disturbances and thus prevent interference with rod control cluster assemblies. Core drop in the event of failure of the normal supports is limited so that the rod cluster control assemblies do not disengage from the fuel assembly guide thimbles.

The internals are further designed to maintain their functional integrity in the event of a major loss-of-coolant accident. The dynamic loading resulting from the pressure oscillations associated with a LOCA does not cause sufficient deformation to prevent RCCA insertion.

The reactor core and reactor vessel internals are shown in cross-section in Figure 4.5-5 and in elevation in Figure 4.5-6. The core, consisting of the fuel assemblies, control rods, source rods, burnable poison rods, and guide thimble plugging devices, provides and controls the heat source for the reactor operation. The internals, consisting of the upper and lower core support structure, are designed to support, align, and guide the core components, direct the coolant flow to and from the core components, and to support and guide the incore instrumentation. A listing of the core mechanical design parameters is given in Table 4.2-1.

The fuel assemblies are arranged in a roughly circular cross-sectional pattern. The assemblies are all identical in configuration, but contain fuel of different enrichments depending on the location of the assembly within the core. Additional information concerning the fuel assemblies can be found in Sections 4.2.3 and 4.3.4.

The fuel is in the form of slightly enriched uranium dioxide ceramic pellets. The pellets are stacked to an active height of 144 inches within Zircaloy-4 tubular cladding which is plugged and seal welded at the ends to encapsulate the fuel. The fuel rods are internally pressurized with helium during fabrication. The enrichments of the fuel for the various regions in the initial core are given in Table 4.2-1. Heat generated by the fuel is removed by demineralized light water which flows upward through the fuel assemblies and acts as both moderator and coolant.

The core is typically divided into regions of three different enrichments. The loading arrangement for the initial cycle is indicated on Figure 4.5-7. Refueling takes place generally in accordance with an inward loading schedule. Beginning with Unit 2, Cycle 6 a low leakage loading pattern (LLLP) has been used. LLLP places some fresh fuel inboard and some irradiated fuel generally at positions with two sides on the baffle. The reason for using a LLLP is better utilization of uranium. Additionally, LLLP reduces embrittlement of the Reactor Pressure Vessel beltline region by reducing the neutron fluence in that region.

The control rods, or RCCAs, consist of groups of individual absorber rods which are held together by a spider at the top end and actuated as a group. In the inserted position, the absorber rods fit within hollow guide thimbles in the fuel assemblies. The guide thimbles are an integral part of the fuel assemblies and occupy locations within the regular fuel rod pattern where fuel rods have been deleted. In the withdrawn position, the absorber rods are guided and supported laterally by guide tubes which form an integral part of the upper core support structure. Figure 4.5-1 shows a typical RCCA.

As shown in Figure 4.5-6, the fuel assemblies are positioned and supported vertically in the core between the upper and lower core plates. The core plates are provided with pins which index into closely fitting mating holes in the fuel assembly top and bottom nozzles. The pins maintain the fuel assembly alignment which permits free movement of the RCCAs from the fuel assembly into the guide tubes in the upper support structure without binding or restriction between the rods and their guide surfaces.

Operational or seismic loads imposed on the fuel assemblies are transmitted through the core plates to the upper and lower support structures and ultimately to the internals support ledge at the pressure vessel flange in the case of vertical loads or to the lower radial support and internals support ledge in the case of horizontal loads. The internals also provide a form fitting baffle surrounding the fuel assemblies which confines the upward flow of coolant in the core area to the fuel bearing region.

4.5.2.1 Reactor Internals Design Description

The reactor internals are designed to support and orient the reactor core fuel assemblies and RCCAs, absorb the control rod dynamic loads and transmit these and other loads to the reactor vessel flange, provide a passageway for the reactor coolant, and support incore instrumentation. The reactor internals are shown in Figure 4.5-6.

The internals are designed to withstand the forces due to weight, preload of fuel assemblies, control rod dynamic loading, vibration, and LOCA blowdown coincident with earthquake accelerations. These internals are analyzed in a manner similar to Connecticut Yankee, San Onofre, Zorita, Saxton and Yankee. Under the loading conditions, including conservative effects of design earthquake loading, the structure satisfies stress values prescribed in Section III, ASME Nuclear Vessel Code.

The reactor internals are equipped with bottom-mounted incore instrumentation supports. These supports are designed to sustain the applicable loads outlined above.

The components of the reactor internals are divided into three parts consisting of the lower core support structure (including the entire core barrel and thermal shield), the upper core support structure and the incore instrumentation support structure.

Zion 1 and 2 core support structures were evaluated with a prototype reactor. The test program was performed on the IPP-II plant and the results obtained from the IPP-II $^{1}/_{7}$th scale model have a direct application to the Zion plant because they are similar 4-loop plants. Measurements from the IPP plant will provide direct stress information (frequencies and amplitudes) from strain gages, and complementary data from accelerometers, displacement gages and pressure gages to be used for a correlation study of the reactor component's vibration.

The IPP-II vibration hot functional test included a wide range of temperature and flow conditions (room temperature, operating temperature and intermediate temperature levels; one-, two-, three- and four-loop operations). During these tests, coolant flow is 10% higher than during normal operation and consequently, oscillatory-forcing forces will be 20% higher. This circumstance assures the severity of this test from the vibration point of view. Temperature effects, in particular differential thermal expansion of components, have no dynamic implications, and local differences due to non-uniformity in temperature distribution are static. Static effects plus vibration amplitudes are considered when applying Section III of the ASME Code to compare with allowables. The test program is principally directed toward the confirmation of vibration levels.

Vibration analysis of reactor internals for normal operation is performed using experimental data and correlation between results obtained from models and full-size plants. For Zion, the study is supported by the instrumentation program in progress for the IPP-II reactor, the lead 4-loop plant. In the final analysis, the "vibrational" hot functional tests followed by careful inspection is the most meaningful confirmatory test. Allowable stress amplitude for vibration is established on the basis of the material fatigue properties for infinite cycles (endurance limit). Since infinite cycle fatigue is a criterion, no limits are then necessary for frequency. Displacement amplitudes for reactor internals vibration are not governing. Stress limits are more restrictive.

Lower Core Support Structure

The major containment and support member of the reactor internals is the lower core support structure, shown in Figure 4.5-8. This support structure assembly consists of the core barrel, the core baffle, and lower core plate and support columns, the thermal shield, the intermediate diffuser plate, and the bottom support plate which is welded to the core barrel. All the major material for this structure is Type 304 Stainless Steel. The core support structure is supported at its upper flange from a ledge in the reactor vessel head flange, and its lower end is restrained in its transverse movement by a radial support system attached to the vessel wall. Within the core barrel are axial baffle and former plates which are attached to the core barrel wall and form the enclosure periphery of the assembled core. The lower core plate is positioned at the bottom level of the core below the baffle plates and provides support and orientation for the fuel assemblies.

The lower core plate is a 2-inch-thick member through which the necessary flow distributor holes for each fuel assembly are machined. Fuel assembly locating pins (two for each assembly) are also inserted into this plate. Columns are placed between this plate and the bottom support plate of the core barrel in order to provide stiffness and to transmit the core load to the bottom support plate. Positioned between the support plate and lower core support plate is a perforated plate to uniformly diffuse the coolant flowing into the core.

The one-piece thermal shield is fixed to the core barrel at the top with rigid bolted connections. The bottom of the thermal shield is connected to the core barrel by means of axial flexures. This bottom support allows for differential axial growth of the shield/core barrel but restricts radial or horizontal movement of the bottom of the shield. Rectangular tubing in which material samples can be inserted and irradiated during reactor operation are welded to the thermal shield and extend to the top of the thermal shield. These samples are held in the rectangular tubing by a preloaded spring device at the top and bottom.

The lower core support structure and principally the core barrel serve to provide passageways and control for the coolant flow. Inlet coolant flow from the vessel inlet nozzles proceeds down the annulus between the core barrel and the vessel wall, flows on both sides of the thermal shield, and then into a plenum at the bottom of the vessel. It then turns and flows up through the lower support plate, passes through the intermediate diffuser plate and then through the lower core plate. The flow holes in the diffuser plate and the lower core plate are arranged to give a very uniform entrance flow distribution to the core. After passing through the core the coolant enters the area of the upper support structure and then flows generally radially to the core barrel outlet nozzles and directly through the vessel outlet nozzles.

A small amount of water also flows between the baffle plates and core barrel to provide additional cooling of the barrel. Similarly, a small amount of the entering flow is directed into the vessel head plenum to provide cooling of the head. Both these flows eventually are directed into the upper support structure plenum and exit through the vessel outlet nozzles.

Vertically downward loads from weight, fuel assembly preload, control rod dynamic loading and earthquake acceleration are carried by the lower core plate partially into the lower core plate support flange on the core barrel shell and partially through the lower support columns to the lower core support and thence through the core barrel shell to the core barrel flange supported by the vessel head flange. Transverse loads from earthquake acceleration, coolant cross flow, and vibration are carried by the core barrel shell to be distributed to the lower radial support to the vessel wall, and to the core barrel flange. Transverse acceleration of the fuel assemblies is transmitted to the core barrel shell by direct connection of the lower core plate to the barrel wall and by a radial support type connection of the upper core plate to slab sided pins pressed into the core barrel.

The main radial support system of the core barrel is accomplished by "key" and "keyway" joints to the reactor vessel wall. At equally spaced points around the circumference, an Inconel block is welded to the vessel I.D. Another Inconel block is bolted to each of these blocks, and has a "keyway" geometry. Opposite each of these is a "key" which is attached to the internals. At assembly, as the internals are lowered into the vessel, the keys engage the keyways in the axial direction. With this design, the internals are provided with a support at the furthest extremity, and may be viewed as a beam fixed at the top and simply supported at the bottom.

Radial and axial expansions of the core barrel are accommodated but transverse movement of the core barrel is restricted by this design. With this system, cycle stresses in the internal structures are within the ASME Section III limits. This eliminates any possibility of failure of the core support.

In the event of downward vertical displacement of the internals, energy absorbing devices limit the displacement by contacting the vessel bottom head. The load is transferred through the energy devices of the internals.

The energy absorbers, cylindrical in shape, are contoured on their bottom surface to the reactor vessel bottom head geometry. Their number and design are determined so as to limit the forces imposed to less than yield. Assuming a downward vertical displacement, the potential energy of the system is absorbed mostly by the strain energy of the energy absorbing devices.

The free fall in the hot condition is on the order of $^1/_2$ inch, and there is an additional strain displacement in the energy-absorbing devices of approximately $^3/_4$ inch. Alignment features in the internals prevent cocking of the internals structure during this postulated drop. The control rods are designed to provide assurance of control rod insertion capabilities under this assumed drop of internals condition. The drop distance of about $1^1/_4$ inch is not enough to cause the tips of the shutdown group of RCCAs to come out of the guide tubes in the fuel assemblies.

<u>Upper Core Support Assembly</u>

The upper core support assembly, shown in Figure 4.5-9, consists of the top support plate, deep beam sections, and upper core plate, between which are contained 48 support columns and 61 guide tube assemblies. The support columns establish the spacing between the top support plate, deep beam sections, and the upper core plate and are fastened at top and bottom to these plates and beams. The support columns transmit the mechanical loadings between the two plates and serve the supplementary function of supporting thermocouple guide tubes. The guide tube assemblies, shown on Figure 4.5-10, sheath and guide the control rod drive shafts and control rods and provide no other mechanical functions. They are fastened to the top support plate and are guided by pins in the upper core plate for proper orientation and support. Additional guidance for the control rod drive shafts is provided by the control rod shroud tube which is attached to the upper support plate and guide tube.

The upper core support assembly, which is removed as a unit during refueling operation, is positioned in its proper orientation with respect to the lower support structure by flat-sided pins pressed into the core barrel which in turn engage in slots in the upper core plate. At an elevation in the core barrel where the upper core plate is positioned, the flat-sided pins are located at angular positions of 0°, 90°, 180°, and 270°. Four slots are milled into the core plate at the same positions. As the upper support structure is lowered into the main internals, the slots in the plate engage the flat-sided pins in the axial direction. Lateral displacement of the plate and of the upper support assembly is restricted by this design. Fuel assembly locating pins protrude from the bottom of the upper core plate and engage the fuel assemblies as the upper assembly is lowered into place. Proper alignment of the lower core support structure, the upper core support assembly, the fuel assemblies and control rods is thereby assured by this system of locating pins and guidance arrangement. The upper core support assembly is restrained from any axial movements by a large circumferential spring which rests between the upper barrel flange and the upper core support assembly and is compressed by the reactor vessel head flange.

Vertical loads from weight, earthquake acceleration, hydraulic loads, and fuel assembly preload are transmitted through the upper core plate via the support columns to the deep beams and top support plate and then the reactor vessel head. Transverse loads from coolant cross flow, earthquake acceleration, and possible vibrations are distributed by the support columns to the top support plate and upper core plate. The top support plate is particularly stiff to minimize deflection.

Incore Instrumentation Support Structures

The incore instrumentation support structures consist of an upper system to convey and support thermocouples penetrating the vessel through the head and a lower system to convey and support flux thimbles penetrating the vessel through the bottom.

The upper system utilizes the reactor vessel head penetrations. Instrumentation port columns are slip-connected to in-line columns that are in turn fastened to the upper support plate. These port columns protrude through the head penetrations. The thermocouples are carried through these port columns and the upper support plate at positions above their readout locations. The quick disconnects for the thermocouples are stainless steel. The thermocouple conduits are supported from the columns of the upper core support system. The thermocouple conduits are sealed stainless steel tubes.

In addition to the upper incore instrumentation, there are reactor vessel bottom port columns which carry the retractable, cold worked stainless steel flux thimbles that are pushed upward into the reactor core. Conduits extend from the bottom of the reactor vessel down through the concrete shield area and up to a thimble seal table. The minimum bend radii are about 144 inches and the trailing ends of the thimbles (at the seal table) are extracted approximately 15 feet during refueling of the reactor in order to avoid interference within the core. The thimbles are closed at the leading ends and serve as the pressure barrier between the reactor pressurized water and the containment atmosphere.

Mechanical seals between the retractable thimbles and the conduits are provided at the seal table. During normal operation, the retractable thimbles are stationary and move only during refueling or for maintenance, at which time a space of approximately 15 feet above the seal table is cleared for the retraction operation. Section 7.7.1.2 contains more information on the layout of the incore instrumentation system.

The conduits are supported at several locations. These supports are tied to the concrete shield wall with members having sufficient flexibility to absorb the thermal movement of the conduits.

The incore instrumentation support structure is designed for adequate support of instrumentation during reactor operation and is rugged enough to resist damage or distortion under the conditions imposed by handling during the refueling sequence.

4.5.2.2 Evaluation of Core Barrel and Thermal Shield

The internals design is based on analysis, test and operational information. Troubles in previous Westinghouse PWRs have been evaluated and information derived has been considered in this design. For example, the new Westinghouse design uses a one-piece thermal shield which is attached rigidly to the core barrel at one end and flexured at the other. The early designs that malfunctioned were multi-piece thermal shields that rested on vessel lugs and were not rigidly attached at the top.

Early core barrel designs that have malfunctioned in service, now abandoned, employed threaded connections such as tie rods, joining the bottom support to the bottom of the core barrel, and a bolted connection that tied the core barrel to the upper barrel. The malfunctioning of core barrel designs in earlier service was believed to have been caused by the thermal shield which was oscillating, thus creating forces on the core barrel. Other forces were induced by unbalanced flow in the lower plenum of the reactor. In today's RCCA design there are no fuel followers to necessitate a large bottom plenum in the reactor. The elimination of these fuel followers enabled Westinghouse to build a shorter core barrel.

The Connecticut Yankee, Indian Point #2 and the Zorita reactor core barrels are of the same construction as the Zion reactor core barrel. Deflection measuring devices employed in the Connecticut Yankee reactor during the hot-functional test, and deflection and strain gages employed in the Zorita reactor during the hot-functional test have provided important information that has been used in the design of the present day internals, including that for Zion. When the Connecticut Yankee thermal shield was modified to the same design as for Southern California Edison, it, too, operated satisfactorily as was evidenced by the examination after the hot-functional test. After these hot-functional tests on all of these reactors, a careful inspection of the internals was provided. All the main structural welds were examined, nozzle interfaces were examined for any differential movement, upper core plate inside supports were examined, the thermal shield attachments to the core barrel including all lockwelds on the devices used to lock the bolts were checked: no malfunctions were found.

Substantial scale model testing was performed at WNES. This included tests which involved a complete full scale fuel assembly which was operated at reactor flow, temperature and pressure conditions. Tests were run on a $^{1}/_{7}$th scale model of the Indian Point Unit 2 reactor. Measurements taken from these tests indicate very little shield movement, on the order of a few mils when scaled up to Indian Point Unit 2. Strain gage measurements taken on the core barrel also indicate very low stresses. Testing to determine thermal shield excitation due to inlet flow disturbances have been included. Information gathered from these tests was used in the design of the thermal shield and core barrel.

In order to provide further confirmation of the internals design, Indian Point Unit 2 has deflection gages mounted on the thermal shield top and bottom for the hot-functional test. Six such gages are mounted in the top of the thermal shield equidistant between the fixed supports and eight located at the bottom, equidistant between the six flexures, and two next to flexure supports. The internals inspection, just before the hot-functional test, includes looking at mating bearing surfaces, main welds and welds that are used on bolt locking devices. At the conclusion of the hot-functional test, measurement readings are taken from the deflectometers on the shield and the internals are re-examined at all key areas for any evidence of malfunction. It can be concluded from the testing programs, analyses and the experience gained that the design as employed on the Zion Plant is adequate.

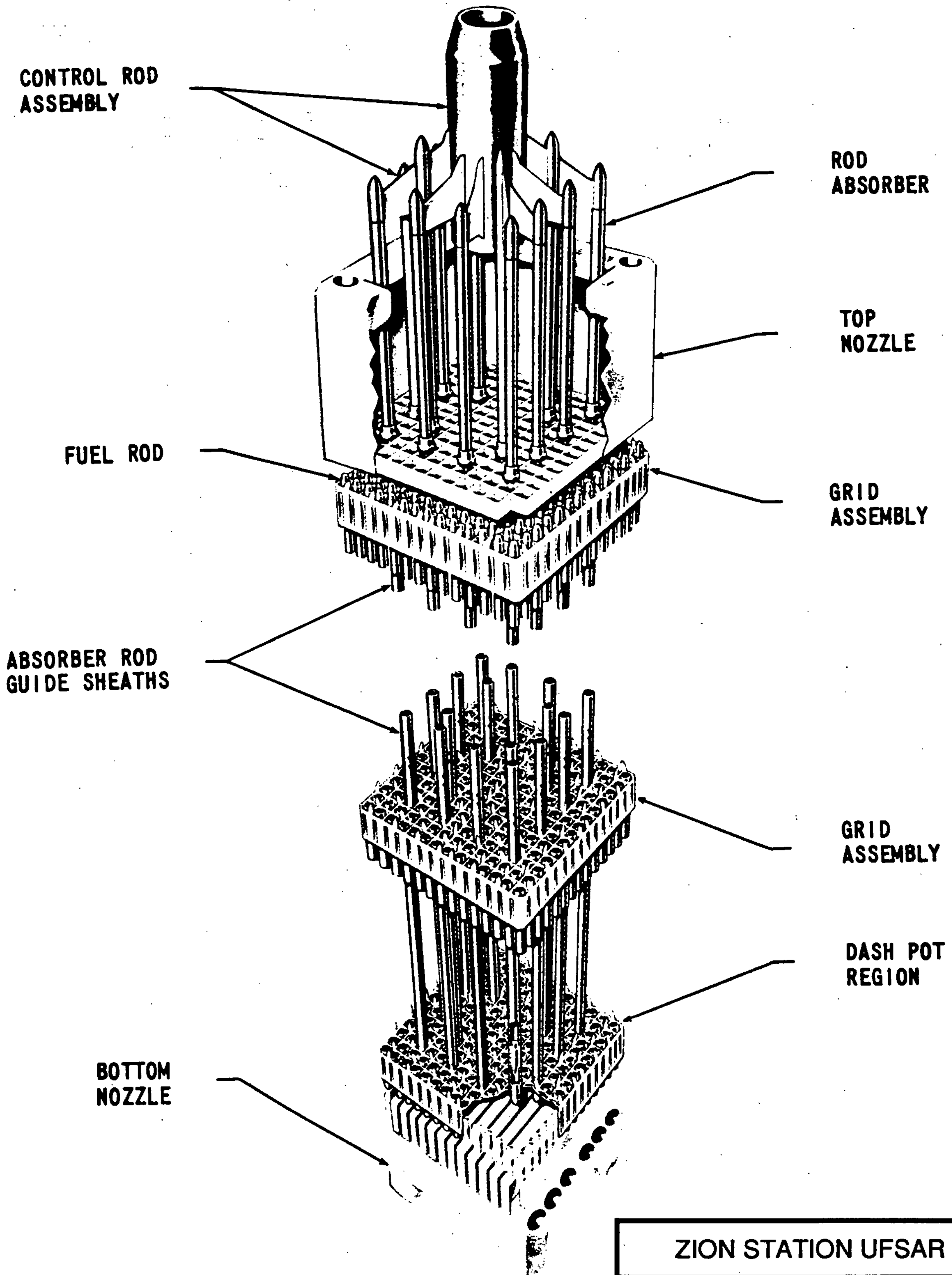
4.5.2.3 Core Component Quality Assurance

To ensure that all materials, components, and assemblies conform to the design requirements, a release point program is established with the manufacturer. This requires surveillance of all raw materials, special processes (i.e, welding, heat treating, nondestructive testing, etc.) and parts which directly affect the assembly and alignment of the reactor internals. The surveillance is accomplished by the issuance of an Inspection Release by quality control organization after conformance has been verified.

A resident quality control representative performs a surveillance/audit program at the manufacturer's facility and witnesses the required tests and inspections and issues the inspection releases. An example is the radiographic examination of the welds joining core barrel shell courses.

Components and materials supplied by Westinghouse to the assembly manufacturer are subjected to a similar program. Quality Control engineers develop inspection plans for all raw materials, components and assemblies. Each level of manufacturing is evaluated by a qualified inspector for conformance (i.e., witnessing the ultrasonic testing of core plate raw material). Upon completion of specified events, all documentation is audited prior to releasing the material or component for further manufacturing. All documentation and inspection releases are maintained in the quality control central records section. All materials are traceable to the mill heat number.

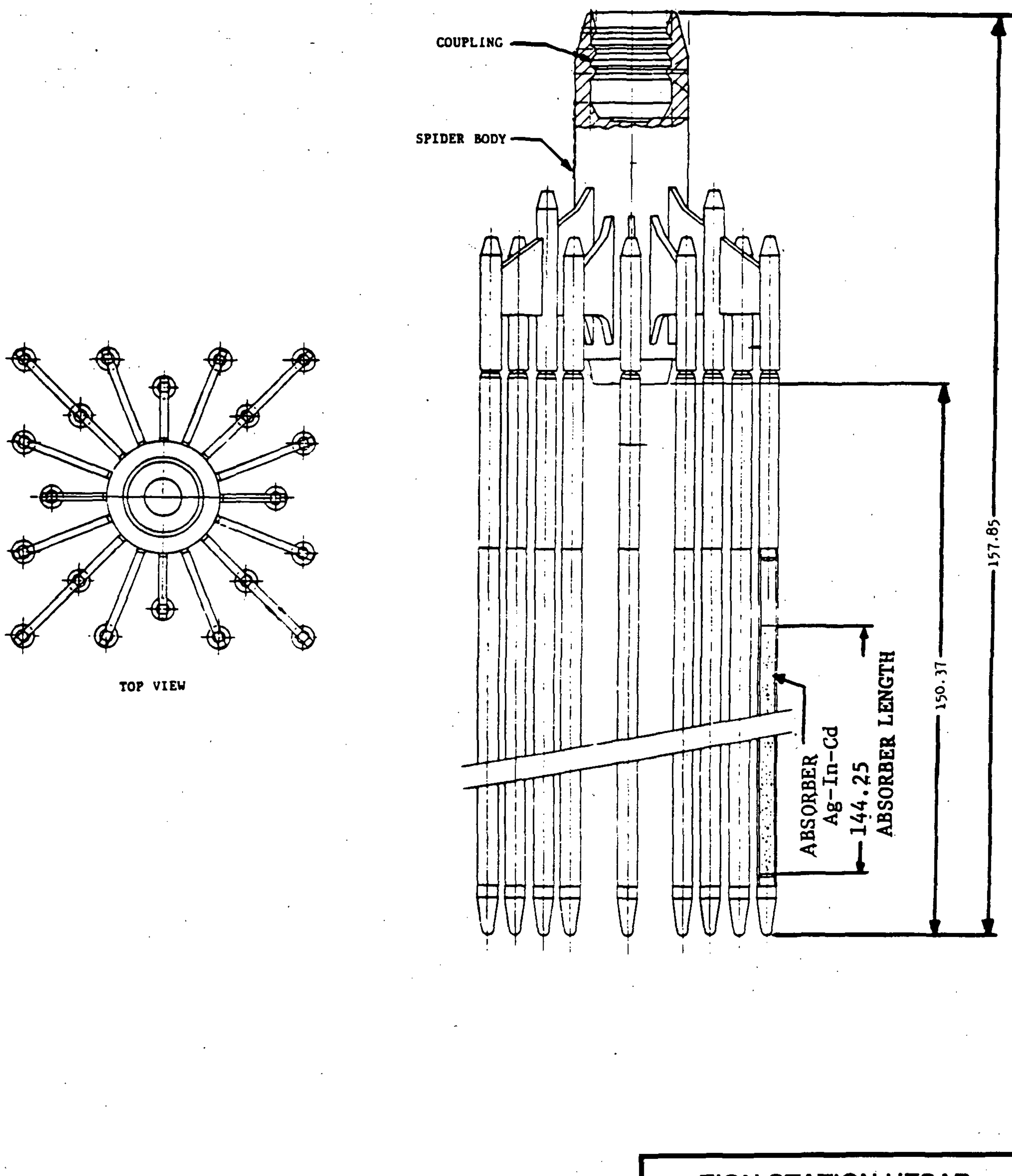
In conclusion, a set of "as built" dimensions are taken to verify conformance to the design requirements and assure proper fitup between the reactor internals and the reactor pressure vessel.



ZION STATION UFSAR

Figure 4.5-1

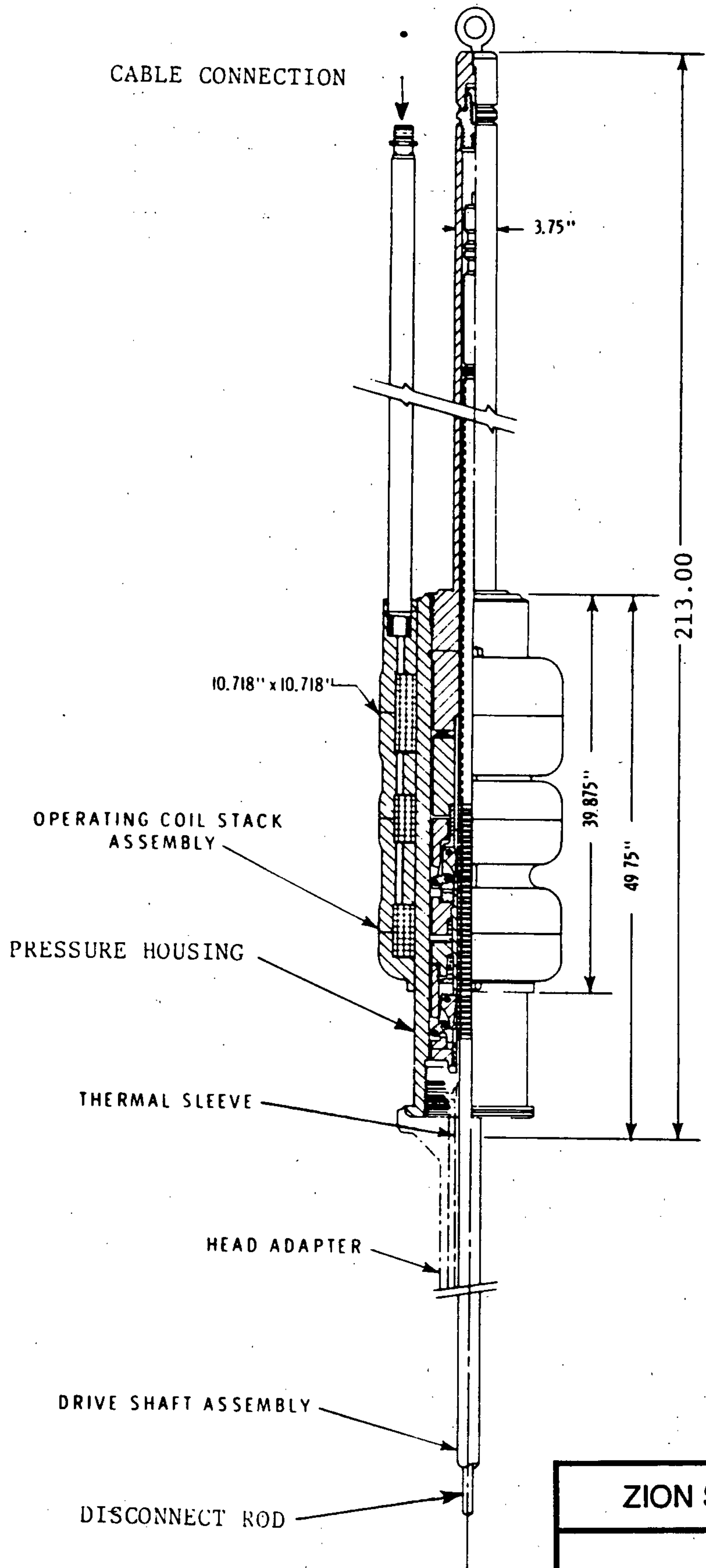
TYPICAL ROD CLUSTER CONTROL ASSEMBLY



ZION STATION UFSAR

Figure 4.5-2

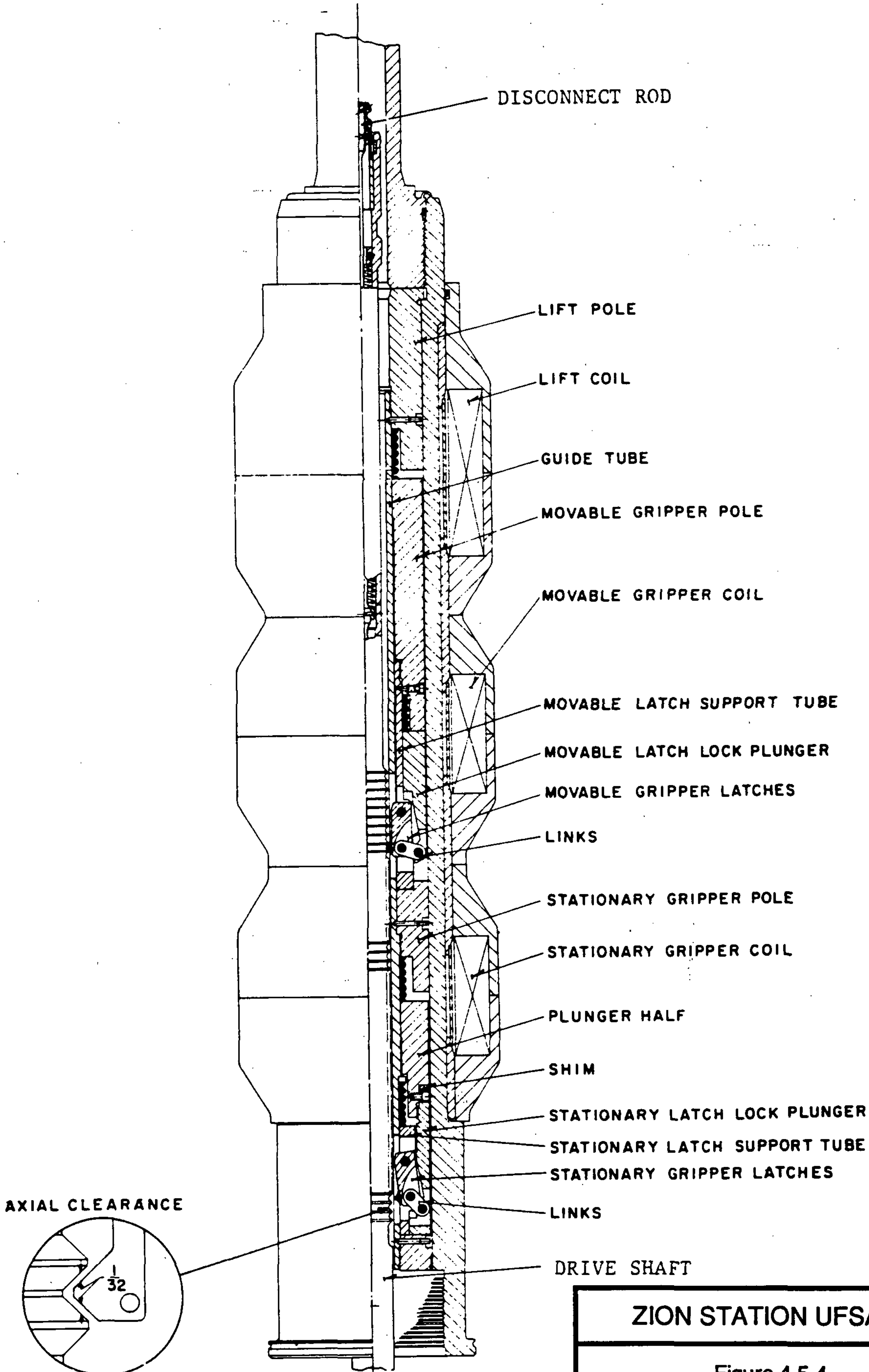
ROD CLUSTER CONTROL ASSEMBLY OUTLINE



ZION STATION UFSAR

Figure 4.5-3

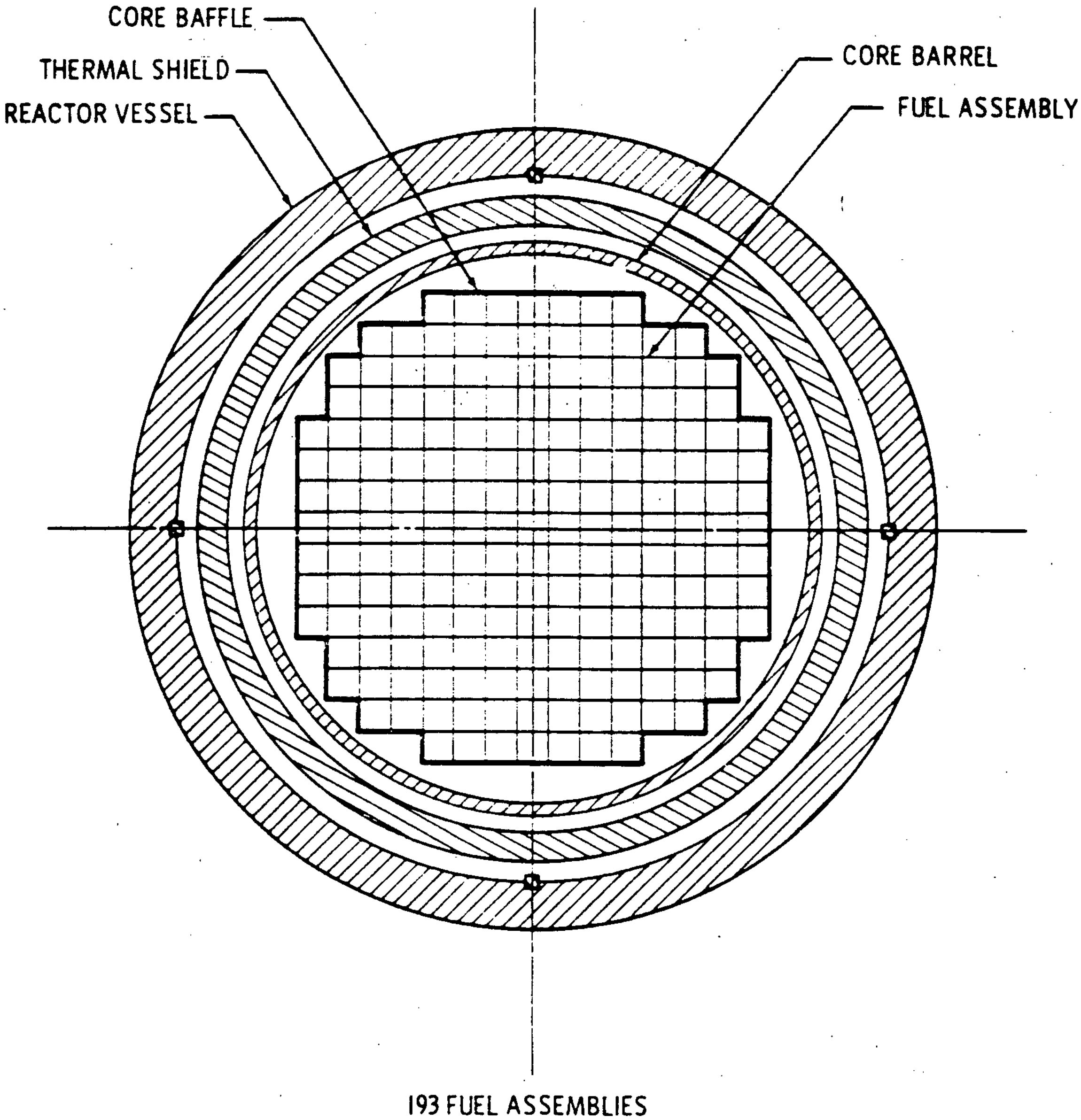
CONTROL ROD DRIVE MECHANISM ASSEMBLY



ZION STATION UFSAR

Figure 4.5-4

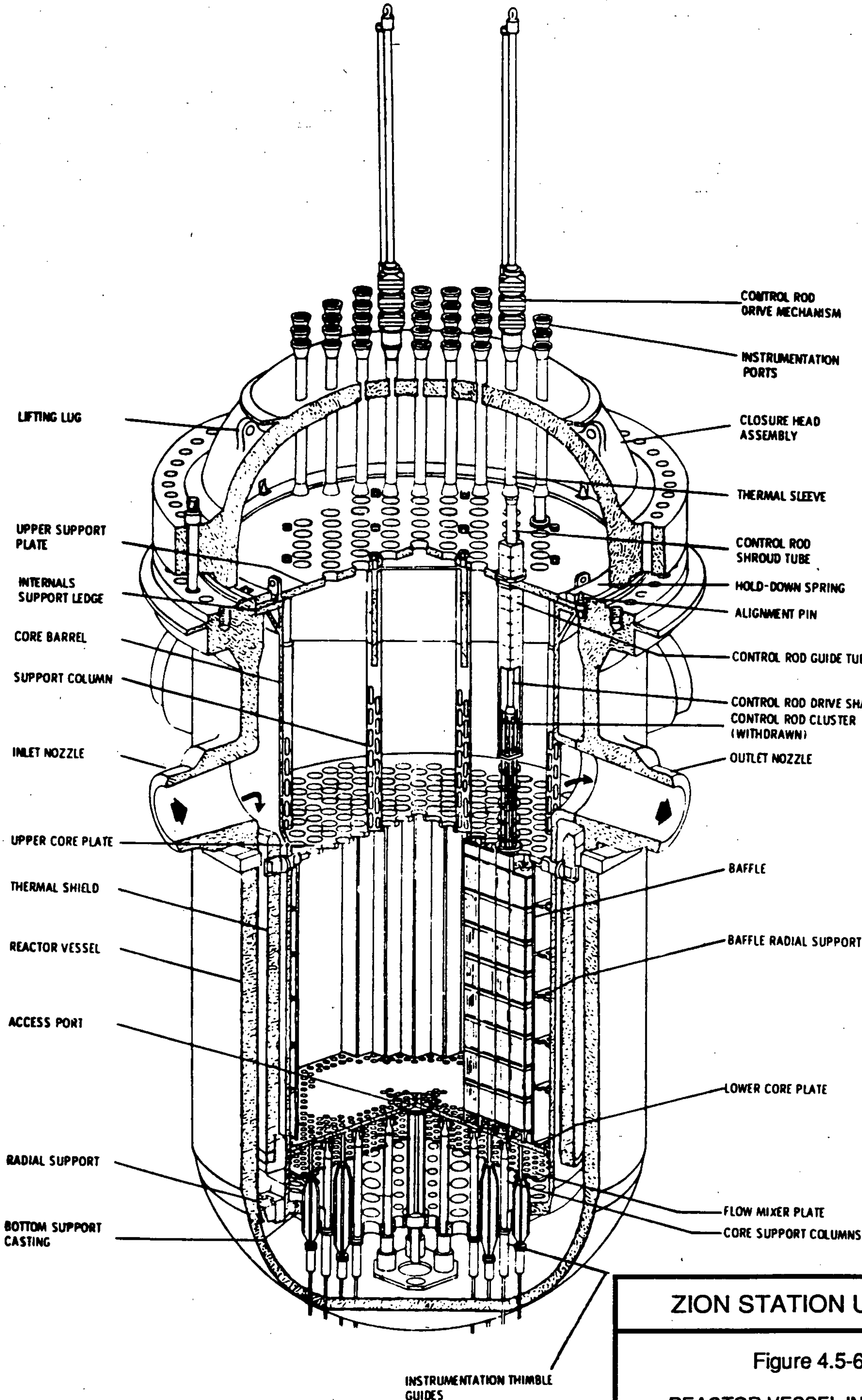
CONTROL ROD DRIVE MECHANISM SCHEMATIC



ZION STATION UFSAR

Figure 4.5-5

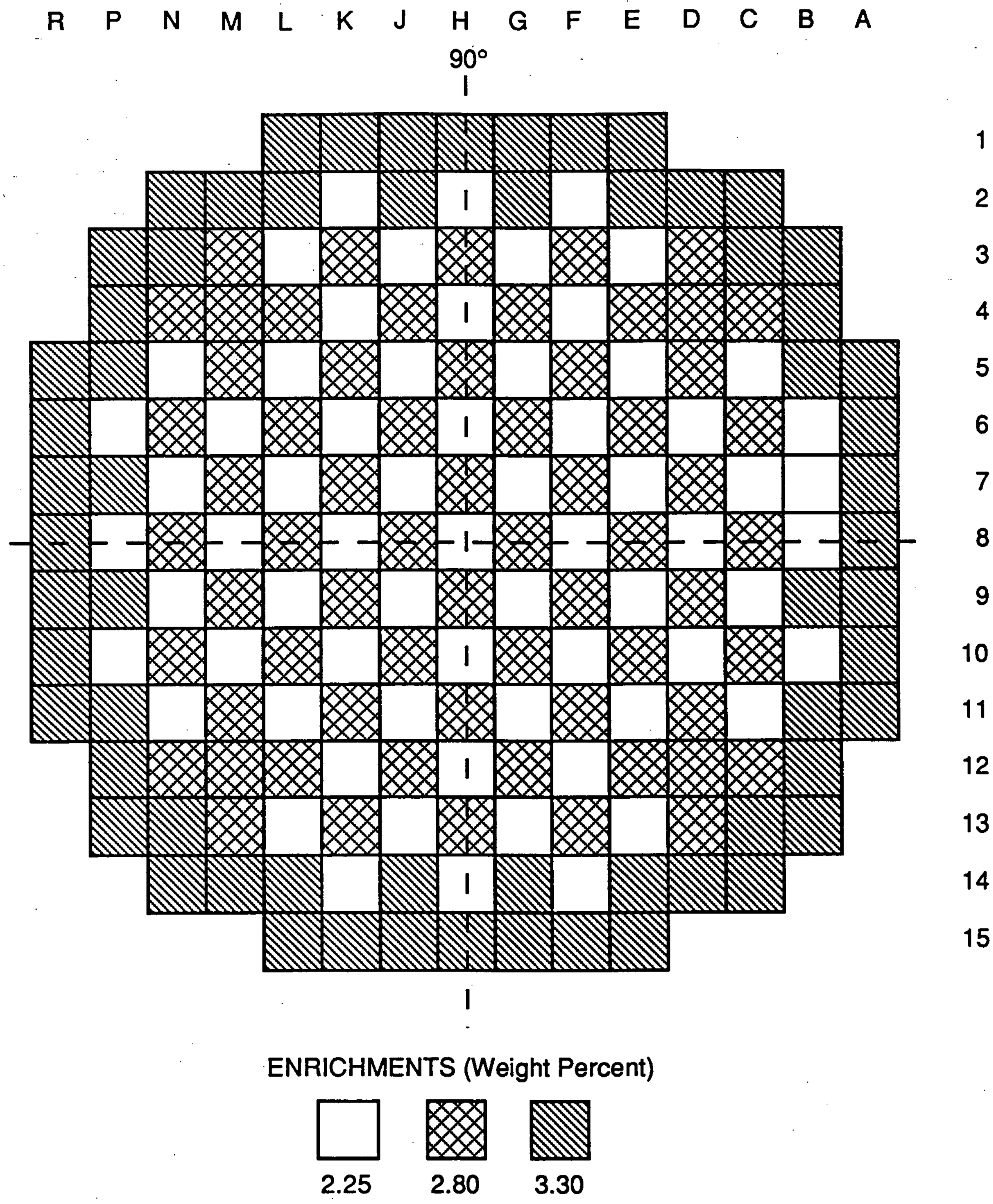
CORE CROSS SECTION



ZION STATION UFSAR

Figure 4.5-6

REACTOR VESSEL INTERNALS



ZION STATION UFSAR

Figure 4.5-7

FIRST CORE LOADING ARRANGEMENT

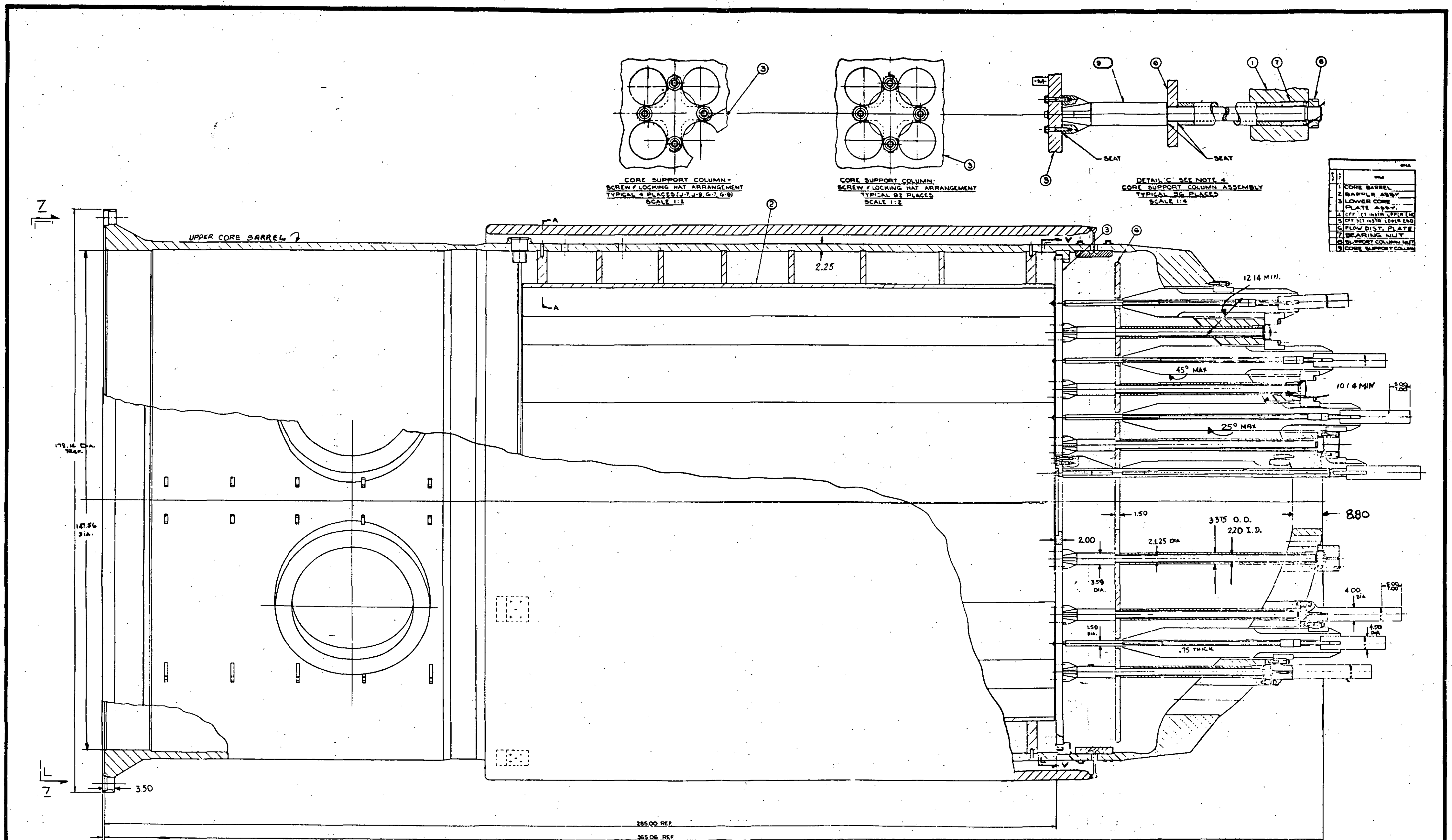


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Figure 4.5-8

CORE BARREL ASSEMBLY

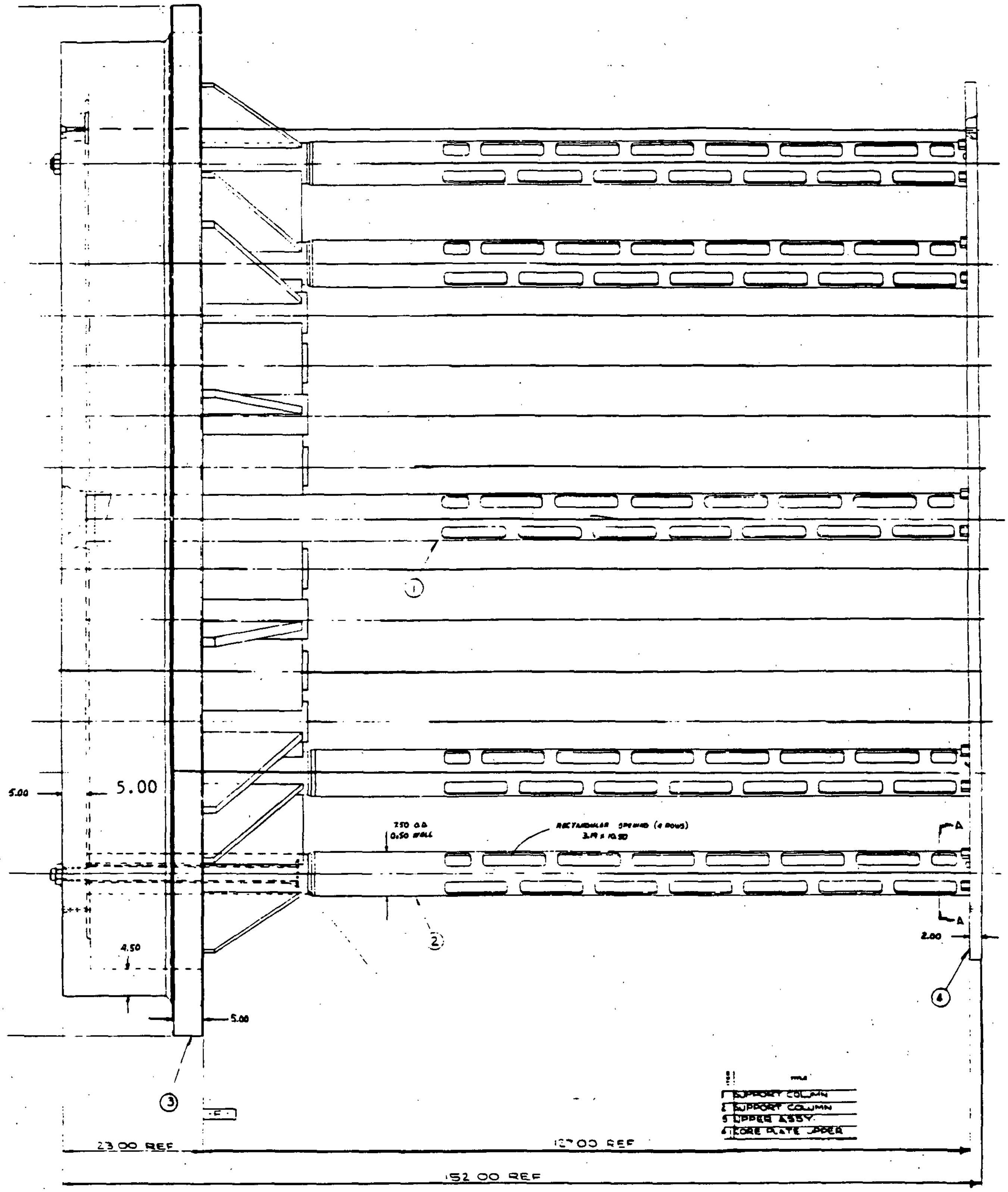

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Figure 4.5-9
UPPER CORE SUPPORT STRUCTURE

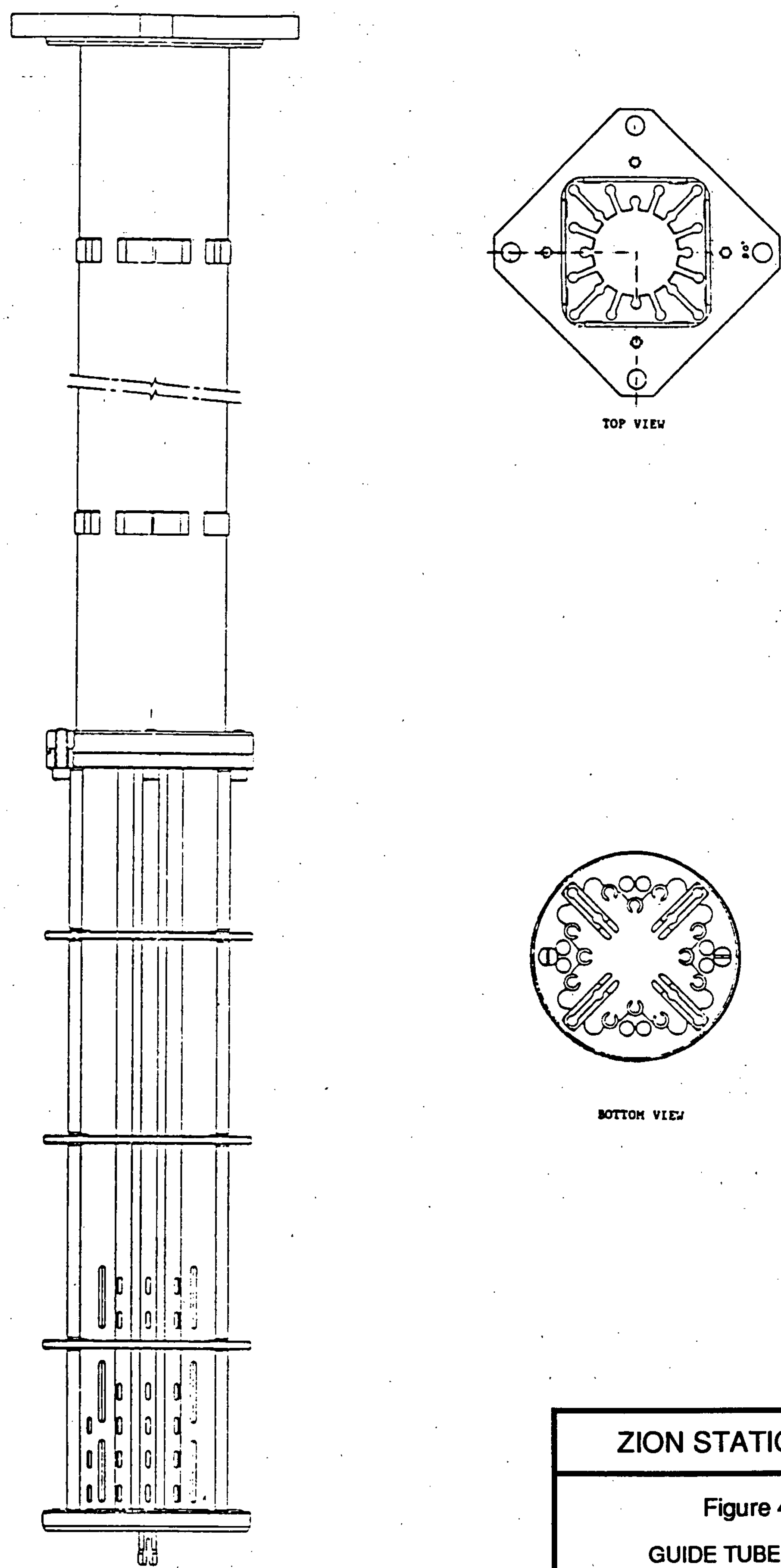

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Figure 4.5-10

GUIDE TUBE ASSEMBLY

4.6 FUNCTIONAL DESIGN OF REACTIVITY CONTROL SYSTEMS

4.6.1 Information for Control Rod Drive (CRD) System

4.6.1.1 Control Rod Drive Assemblies

Each control rod drive assembly is designed as a hermetically sealed unit to prevent leakage of reactor coolant. All pressure-containing components are designed to meet the requirements of the ASME Code, Section III, Nuclear Vessels for Class A vessels.

The control rod drive assemblies for the full-length rods provide rod cluster control assembly (RCCA) insertion and withdrawal rates consistent with the required reactivity changes for reactor operational load changes. This rate is based on the worths of the various rod groups, which are established to limit power-peaking flux patterns to design values. The maximum reactivity addition rate is specified to limit the magnitude of a possible nuclear excursion resulting from a control system or operator error malfunction. Also, the control rod drive assemblies provide a fast insertion rate during a "trip" of the RCCAs which results in a rapid shutdown of the reactor for conditions that cannot be handled by the reactor control system. For further information on this subject refer to Section 4.5.1.

4.6.2 Evaluations of the CRD System

This subsection title has been created in order to implement the UFSAR format delineated by Regulatory Guide 1.70, Revision 3. However, the section is not used due to the level of detail required at the time of license application and subsequent revisions.

4.6.3 Testing and Verification of the CRD System

This subsection title has been created in order to implement the UFSAR format delineated by Regulatory Guide 1.70, Revision 3. However, the section is not used due to the level of detail required at the time of license application and subsequent revisions.

4.6.4 Information for Combined Performance of Reactivity Systems

This subsection title has been created in order to implement the UFSAR format delineated by Regulatory Guide 1.70, Revision 3. However, the section is not used due to the level of detail required at the time of license application and subsequent revisions.

4.6.5 Evaluations of Combined Performance

This subsection title has been created in order to implement the UFSAR format delineated by Regulatory Guide 1.70, Revision 3. However, the section is not used due to the level of detail required at the time of license application and subsequent revisions.

APPENDIX 4A: FUEL DENSIFICATION - ZION STATION UNIT NO. 1

This Appendix has been deleted. See References 12 and 13 of Section 4.2 for information.

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5. REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

5.1 SUMMARY DESCRIPTION

The Reactor Coolant System (RCS) consists of four similar heat transfer loops connected in parallel to the reactor vessel. Each loop contains a steam generator, a pump, two isolation valves, loop piping, and instrumentation. The system also includes a pressurizer, connecting piping to one of the loops, pressurizer safety and relief valves, and a relief tank necessary for operational control. Auxiliary system piping connections into the reactor coolant piping are provided as necessary.

RCS design data are listed in Tables 5.1-1 and 5.1-2. Connected systems and components of the RCS are discussed in detail in Section 5.4.

Pressure in the system is controlled by the pressurizer through the use of electrical heaters and sprays. Steam can either be formed by the heaters or condensed by a pressurizer spray to minimize pressure variations due to contraction and expansion of the coolant. Instrumentation used in the pressure control system is described in Chapter 7. Spring-loaded safety valves and power-operated relief valves are connected to the pressurizer and discharge to the pressurizer relief tank, where the discharged steam is condensed and cooled by mixing with water.

The RCS transfers the heat generated in the core to the steam generators, where steam is generated to drive the turbine generator.

Demineralized light water is circulated at the flow rate and temperature consistent with achieving the reactor core thermal hydraulic performance presented in Chapter 4. The water also acts as a neutron moderator, a neutron reflector, and a solvent for the neutron absorber used in chemical shim control.

The water chemistry is selected to provide the necessary boron content for reactivity control and to minimize corrosion of RCS surfaces. Periodic analyses of the coolant chemical composition are performed to monitor the adherence of the system to the desired reactor coolant water quality listed in Table 5.2-4. Maintenance of the water quality to minimize corrosion is accomplished using the Chemical and Volume Control System and the Sampling System which are described in Chapter 9.

The RCS provides a boundary for containing the coolant under operating temperature and pressure conditions. It serves to confine radioactive material and to limit its uncontrolled release to the secondary system and to other parts of the plant under normal or abnormal operating conditions. During transient operation, the system's heat capacity attenuates thermal transients generated by the core or the steam generators. The RCS accommodates coolant volume changes within the protection system limits of the reactor as presented in Chapter 7.

By appropriate selection of the inertia of the reactor coolant pumps, the thermal hydraulic effects are reduced to a safe level during the pump coastdown in a loss-of-flow situation. The layout of the system assures the natural circulation capability following a loss of flow to permit decay heat removal without overheating the core. Part of the system piping serves as part of the Emergency Core Cooling System (ECCS) when delivering cooling water to the core during a loss-of-coolant accident (LOCA).

5.1.1 Schematic Flow Diagram

A simplified schematic of the RCS is not contained in the current revision of the Zion UFSAR. Figure 5.1-1 (Sheets 1 and 2) shows a detailed diagram of the RCS and interconnecting systems.

5.1.2 Piping and Instrumentation Diagram

A piping and instrument diagram of the RCS is shown in Figure 5.1-1 (Sheets 1 and 2). The diagram also shows RCS connections to other systems. A discussion of RCS instrumentation is contained in Chapter 7.

5.1.3 Elevation Drawing

This subsection title has been created in order to implement the UFSAR format delineated by Regulatory Guide 1.70, Revision 3. However, the section is not used due to the level of detail required at the time of license application and subsequent revisions.

TABLE 5.1-1

SYSTEM DESIGN AND OPERATING PARAMETERS

Plant design life, years	40
Number of heat transfer loops	4
Design pressure, psig	2485
Nominal operating pressure, psig	2235
Total system volume including pressurizer* and surge line, ft^3 (ambient conditions)	12,710
System liquid volume, including pressurizer* and surge line, ft^3 (ambient conditions)	11,990
Total heat output (100% power) Btu/hr	$11{,}089 \times 10^6$
Reactor vessel coolant temperature at full power:	
Inlet, nominal, °F	530.2
Outlet, °F	594.2
Coolant temperature rise in vessel at full power, avg, °F	64.0
Total coolant flow rate, lb/hr	135.0×10^6
Steam pressure, psia*	720

* At full power operation with zero steam generator tubes plugged.

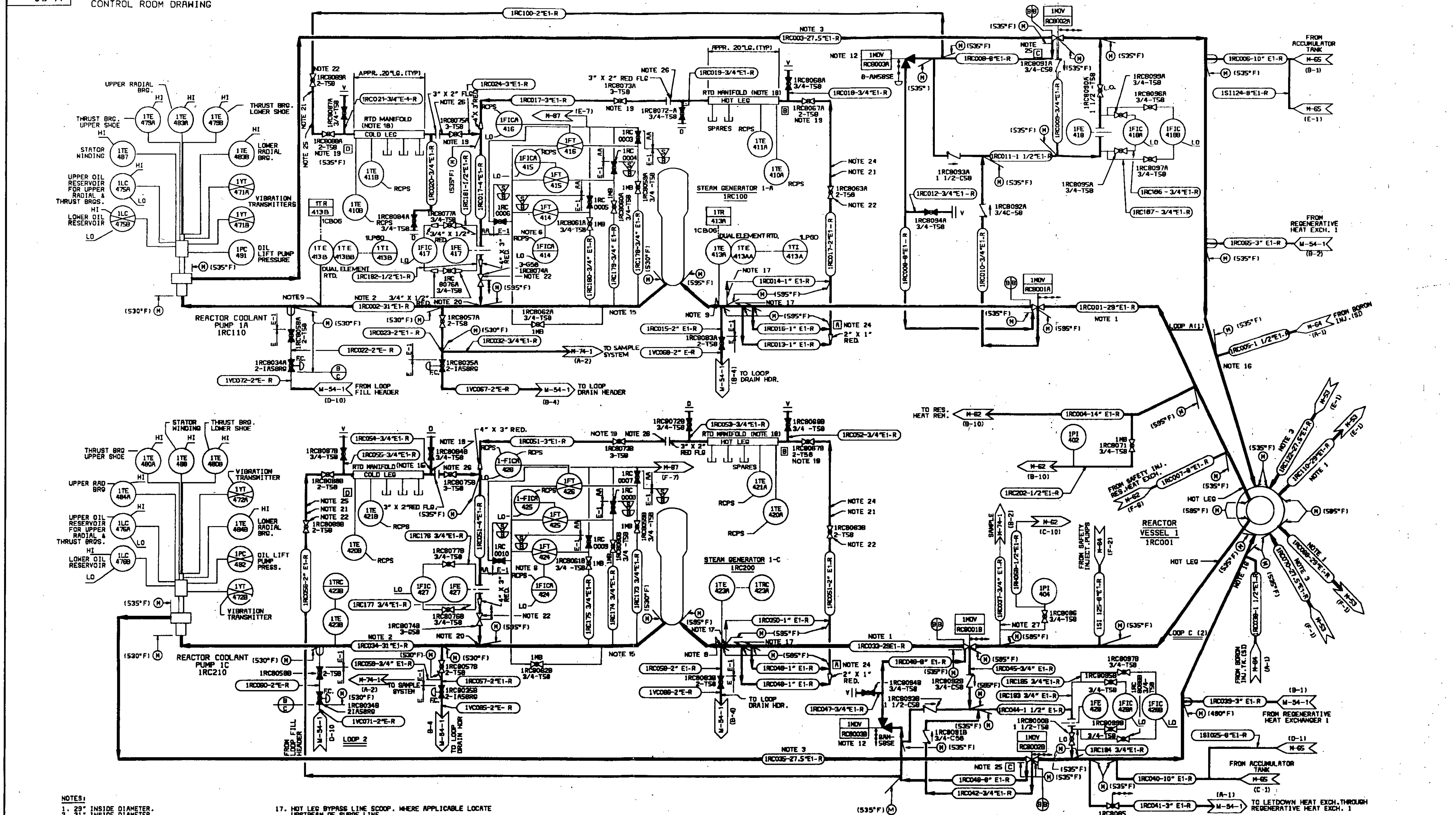
TABLE 5.1-2

REACTOR COOLANT SYSTEM DESIGN PRESSURE SETTINGS (PSIG)

Design Pressure	2485
Operating Pressure	2235
Safety Valves	2485
Power Relief Valves *	2335
Pressurizer Spray Valves (Begin to Open)	2260
Pressurizer Spray Valves (Full Open)	2310
High Pressure Trip	2385
High Pressure Alarm	2335
Low Pressure Trip	1825
Low Pressure Alarm	2210
Hydrostatic Test Pressure	3107
Backup Heaters On	2185
Proportional Heaters (Begin to Operate)	2250
Proportional Heaters (Full Operation)	2220

* Setpoint is at normal operating temperature. The setpoint is reduced to provide overpressurization protection at low temperatures.

CONTROL ROOM DRAWING



NOTES:

1. 29" INSIDE DIAMETER.
2. 31" INSIDE DIAMETER.
3. 27.5" INSIDE DIAMETER.
4. GLOBE VALVES NORMALLY INSTALLED WITH FLOW UNDER SEAT. EXCEPTIONS ARE:
5. SPRAY LINE SCOOP.
6. ELBOW FLOW METERS.
7. VENT PIPE FURNISHED WITH REACTOR VESSEL HEAD.
8. HEAD GASKET MONITORING CONNECTIONS FURNISHED WITH REACTOR VESSEL.
9. RTD INSTALLED IN WELL.
10. LOCATE CONNECTION IN BOTTOM HALF OF REACTOR COOLANT PIPING ON 45° ANGLE TO VERTICAL.
11. ALL ⊛ ITEM NUMBERS IN RCS ARE SHOWN WITHOUT PREFIX.
12. 0.25" HOLE THROUGH VALVE PLUG.
13. VALVE WELDED DIRECTLY TO PUMP.
14. LOCATE BELOW ELEVATION OF REACTOR VESSEL FLANGE.
15. LOCATE FLOWMETER IN 10" ELBOW DOWNSTREAM OF STEAM GENERATORS.
16. PLACE AS CLOSE AS POSSIBLE TO REACTOR VESSEL NOZZLE.
17. HOT LEG BYPASS LINE SCOOP. WHERE APPLICABLE LOCATE UPSTREAM OF SURGE LINE.
18. RTD MANIFOLD - PIPE & RTD'S SUPPLIED AS A PACKAGE MANIFOLD APPROX. 20 INCHES LONG.
19. LOCATE RTD MANIFOLD ISOLATION VALVES APPROX. 8 INCHES FROM MANIFOLD.
20. LOCATE CONNECTION ON UPPER 180° OF PIPE CIRCUMFERENCE.
21. ALL BYPASS LOOP PIPING AND THE RTD MANIFOLD SHALL HAVE REMOVABLE INSULATION UP TO LOOP ROOT VALVES.
22. LOCATE ROOT VALVE ABOVE ELEVATION OF REACTOR VESSEL NOZZLES, IF POSSIBLE.
23.
24. LENGTH OF HOT LEG 2 INCH PIPE UPSTREAM OF RTD $\boxed{A}$ TO $\boxed{B}$ TO BE A MAXIMUM OF 3 FEET.
25. LENGTH OF COLD LEG 2" & 6" PIPE UPSTREAM OF RTD MANIFOLD $\boxed{C}$ TO $\boxed{D}$ TO BE A MAXIMUM OF 6 FEET.
26. FLANGES ARE INSTALLED FOR INSERTION OF FLOW LIMITING ORIFICES IF REQUIRED. ORIFICE PLATE TO BE DRILLED TO MATCH PIPE I.D.
27. 1.05 O.D. - .234 I.D.

NOTE:
INSULATION IS METALLIC REFLECTIVE TYPE

ASME CODE RELATED AND NUCLEAR SAFETY RELATED ITEMS ARE SHOWN ON THIS DRAWING

CONTROL ROOM DRAWING

ZION STATION UFSAR

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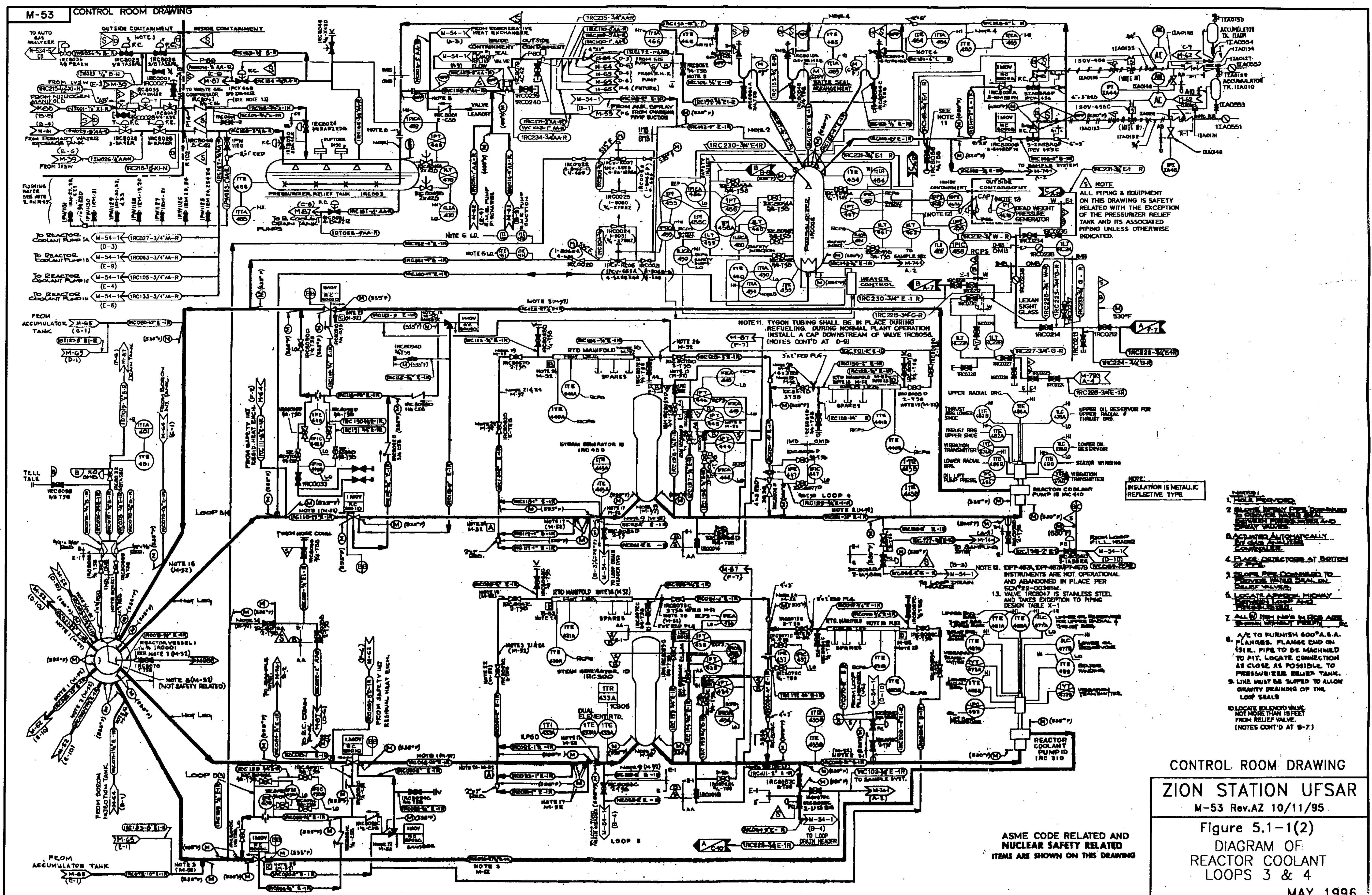
Figure 5.1-1(1)

REACTOR COOLANT DIAGRAM LOOPS 1 AND 2 (UNIT 1)

MAY 1996



CONTROL ROOM DRAWING

ZION STATION UFSAR

M-53 Rev.AZ 10/11/95

Figure 5.1-1(2)

DIAGRAM OF REACTOR COOLANT LOOPS 3 & 4

MAY 1996

5.2 INTEGRITY OF REACTOR COOLANT PRESSURE BOUNDARY

5.2.1 Compliance with Codes and Code Cases

All primary pressure-containing components of the Reactor Coolant System (RCS) are designed, fabricated, inspected, and tested in conformance with the applicable codes listed in Table 5.2-1, and are Seismic Class I design.

To establish the service life of the RCS components as required by the ASME Boiler and Pressure Vessel (B&PV) Code, Section III, for Class "A" vessels, the unit operating conditions have been established for the 40-year design life. These operating conditions include the cyclic application of pressure loadings and thermal transients. The number of thermal and loading cycles used for design purposes is listed in Table 5.2-2.

Environmental protection is afforded by close adherence to the water chemistry limits set forth in the Technical Specifications and by the absence of any deleterious conditions in the Containment environment, piping, and component insulation.

Maintenance standards will comply with the applicable codes and standards and with appropriate quality levels. Operating procedures will be established and adhered to in accordance with the Zion Quality Assurance Plan.

5.2.1.1 Design Criteria

5.2.1.1.1 Quality Standards

Criterion: Those systems and components of reactor facilities which are essential to the prevention, or the mitigation of the consequences, of nuclear accidents which could cause undue risk to the health and safety of the public shall be identified and then designed, fabricated, and erected to quality standards that reflect the importance of the safety function to be performed. Where generally recognized codes and standards pertaining to design, materials, fabrication, and inspection are used, they shall be identified. Where adherence to such codes or standards does not suffice to assure a quality product in keeping with the safety function, they shall be supplemented or modified as necessary. Quality assurance programs, test procedures, and inspection acceptance criteria to be used shall be identified. An indication of the applicability of codes, standards, quality assurance programs, test procedures, and inspection acceptance criteria used is required. Where such items are not covered by applicable codes and standards, a showing of adequacy is required.

The RCS is of primary importance with respect to its function in protecting the health and safety of the public.

Quality standards of material selection, design, fabrication, and inspection conform to the applicable provisions of recognized codes and good nuclear practice. Details of the quality assurance programs, test procedures and inspection acceptance levels are given in Sections 5.2.4 and 5.3.3. Particular emphasis is placed on the assurance of quality of the reactor vessel to obtain material whose properties are uniformly within tolerances appropriate to the application of the design methods of the code delineated in Section 5.2.1.

5.2.1.1.2 Performance Standards

Criterion: Those systems and components of reactor facilities which are essential to the prevention or to the mitigation of the consequences of nuclear accidents which could cause undue risk to the health and safety of the public shall be designed, fabricated, and erected to performance standards that will enable such systems and components to withstand, without undue risk to the health and safety of the public the forces that might reasonably be imposed by the occurrence of an extraordinary natural phenomena such as earthquake, tornado, flooding condition, high wind or heavy ice. The design bases so established shall reflect: (a) appropriate consideration of the most severe of these natural phenomena that have been officially recorded for the site and the surrounding area and (b) an appropriate margin for withstanding forces greater than those recorded to reflect uncertainties about the historical data and their suitability as a basis for design.

RCS piping and components containing operating pressure, and supporting structures thereto, are designed as Seismic Class I. Details are given in Section 3.7.1.

The RCS is located in the Containment where design, in addition to being a Seismic Class I structure, also considers accidents or other applicable natural phenomena. Details of the containment design are given in Section 3.8.1.

Code records will be maintained for the mandatory period, and thereafter, either by Westinghouse or the Commonwealth Edison Company (CECo).

5.2.1.1.3 Records Requirements

Criterion: The reactor licensee shall be responsible for assuring the maintenance throughout the life of the reactor of records of the design, fabrication, and construction of major components of the plant essential to avoid undue risk to the health and safety of the public.

Records that should be maintained may or may not be under the physical control of CECo.

CECo will assure that those records which are important, in that they have some bearing on the health and safety of the public, are maintained.

5.2.1.1.4 Reactor Coolant Pressure Boundary

Criterion: The reactor coolant pressure boundary shall be designed, fabricated and constructed so as to have an exceedingly low probability of gross rupture or significant uncontrolled leakage throughout its design lifetime.

The RCS, in conjunction with its control and protective provisions, is designed to accommodate the system pressures and temperatures attained under all expected modes of plant operation or anticipated system interactions, and to maintain the stresses within applicable code stress limits. The criteria for vessels and piping within the reactor coolant pressure boundary are discussed in Appendix 5A.

Fabrication of the components which constitute the pressure-retaining boundary of the RCS were carried out in strict accordance with the applicable codes. In addition, there are areas where equipment specifications for RCS components go beyond the applicable codes. Details are given in Section 5.2.3.

The materials of construction of the pressure-retaining boundary of the RCS are protected, by control of coolant chemistry, from corrosion phenomena which might otherwise reduce the system structural integrity during its service lifetime. This is discussed in Chapter 9.

System conditions resulting from anticipated transients or malfunctions are monitored and appropriate action is automatically initiated to maintain the required cooling capability and to limit system conditions such that continued safe operation is possible. This is discussed in Chapter 7.

The system is protected from overpressure by means of pressure-relieving devices as required by Section III of the ASME B&PV Code.

Sections of the system which can be isolated are provided with overpressure-relieving devices discharging to closed systems such that the system code allowable relief pressure within the protected section is not exceeded.

5.2.1.1.5 Monitoring Reactor Coolant Leakage

Criterion: Means shall be provided to detect significant uncontrolled leakage from the reactor coolant pressure boundary.

The above criteria is implemented by positive indications in the Control Room to alert the operator of leakage of coolant from the RCS. Separate systems based on different operating principles provide leakage data from the following individual areas within the Containment:

1. Reactor head to vessel closure joint.
2. Reactor vessel outer surface.
3. Incore instrumentation seal table.

In addition to the above individual areas, the overall containment atmosphere is monitored by means of radioactive air particulate and gas systems, relative humidity monitors, and temperature indications.

5.2.1.1.6 Reactor Coolant Pressure Boundary Capability

Criterion: The reactor coolant pressure boundary shall be capable of accommodating without rupture the static and dynamic loads imposed on any boundary component as a result of an inadvertent and sudden release of energy to the coolant. As a design reference, this sudden release shall be taken as that which would result from a sudden reactivity insertion such as rod ejection (unless prevented by positive mechanical means), rod dropout, or cold water addition.

The reactor coolant boundary is shown to be capable of accommodating, without further rupture, the static and dynamic loads imposed as a result of a sudden reactivity insertion such as a rod ejection which is considered the worst credible case. Details of this analysis are provided in Chapter 15.

The operation of the reactor is such that the severity of an ejection accident is inherently limited. Since control rod clusters are used to control load variations only and core depletion is followed with boron dilution, only the rod cluster control assemblies in the controlling groups are inserted in the core at power. At full power, these rods are only partially inserted. A rod insertion limit monitor is provided as an administrative aid to the operator to assure that this condition is met.

By using flexibility in the selection of control rod groupings and radial location and position as a function of load, the design limits the maximum fuel temperature for the highest worth ejected rod to a value which precludes any resultant damage to the RCS pressure boundary, i.e., gross fuel dispersion in the coolant and possible excessive pressure surges.

The failure of a rod mechanism housing causing a rod cluster to be rapidly ejected from the core is evaluated as a theoretical, though not a credible, accident. While limited fuel damage could result from this hypothetical event, the fission products are confined to the RCS and the Reactor Containment. The environmental consequences of rod ejection are less severe than from the hypothetical loss of coolant, for which public health and safety is shown to be adequately protected. Reference is made to Chapter 15.

5.2.1.1.7 Reactor Coolant Pressure Boundary Rapid Propagation Failure Prevention

Criterion: The reactor coolant pressure boundary shall be designed and operated to reduce to an acceptable level the probability of rapidly propagating type failure. Consideration is given (a) to the provisions for control over service temperature and irradiation effects which may require operational restrictions, (b) to the design and construction of the reactor pressure vessel in accordance with applicable codes, including those which establish requirements for absorption of energy within the elastic strain energy range and for absorption of energy by plastic deformation and (c) to the design and construction of reactor coolant pressure boundary piping and equipment in accordance with applicable codes.

The reactor coolant pressure boundary is designed to reduce, to an acceptable level, the probability of a rapidly propagating type failure.

In the core region of the reactor vessel, it is expected that the notch toughness of the material will change as a result of fast neutron exposure. This change is evidenced as a shift in the nil ductility transition temperature (NDTT), which is factored into the operating procedures in such a manner that full operating pressure is not obtained until the affected vessel material is above the design transition temperature (DTT) and is in the ductile material region. The pressure during startup and shutdown at temperatures below NDTT is maintained below the threshold of concern for safe operation.

The DTT is a minimum of NDTT plus 60°F and dictates the procedures to be followed in the hydrostatic test and in station operations to avoid excessive cold stress. The value of the DTT is increased during the life of the plant. This is required by the expected shift in NDTT and is confirmed by the experimental data obtained from irradiated specimens of reactor vessel materials during the plant lifetime. Further details are given in Section 5.3.1.1.

All pressure-containing components of the RCS are designed, fabricated, inspected, and tested in conformance with the applicable codes. Further details are given in Section 5.2.1.

5.2.1.1.8 Reactor Coolant Pressure Boundary Surveillance

Criterion: Reactor coolant pressure boundary components shall have provisions for inspection, testing, and surveillance of critical areas by appropriate means to assess the structural and leaktight integrity of the boundary components during their service lifetime. For the reactor vessel, a material surveillance program conforming with current applicable codes shall be provided.

The design of the reactor vessel and its arrangement in the system provides for accessibility to the entire internal surfaces of the vessel and certain external zones of the vessel including the nozzle to reactor coolant piping welds and the top and bottom heads.

The reactor arrangement within the Containment provides sufficient space for inspection of the external surfaces of the reactor coolant piping, except for the area of pipe within the primary shielding concrete.

Monitoring of the NDTT properties of the core region plates, forgings, weldments, and associated heat-treated zones are performed in accordance with American Society of Testing Materials (ASTM) E-185-70 (Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels). Samples of reactor vessel plate materials are retained and catalogued in case future engineering development shows the need for further testing.

The material properties surveillance program includes not only the conventional tensile and impact tests, but also fracture mechanics specimens. The fracture mechanics specimens are the wedge opening loading (WOL) type specimens. The observed shifts in NDTT of the core region materials with irradiation will be used to confirm the calculated limits to startup and shutdown transients.

To define permissible operating conditions below DTT, a pressure range is established which is bounded by a lower limit for pump operation and an upper limit which satisfies reactor vessel stress criteria. To allow for thermal stresses during heatup or cooldown of the reactor vessel, an equivalent pressure limit is defined to compensate for thermal stress as a function of rate of change of coolant temperature. Since the normal operating temperature of the reactor vessel is well above the maximum expected DTT, brittle fracture during normal operation is not considered to be a credible mode of failure.

Inservice integrity assurance,is afforded by an operational program as delineated in the Technical Specifications using the capabilities described in Section 5.2.4.1.

5.2.2 Overpressurization Protection

The RCS is protected against overpressure by protective circuits such as the high-pressure trip and by relief and safety valves connected to the top head of the pressurizer. The relief and safety valves discharge into the pressurizer relief tank which condenses and collects the valve effluent. The schematic arrangement of the relief devices is shown in Figure 5.1-1, and the valve design parameters are given in Table 5.4-23. The valves are further discussed in Section 5.4.13.

The safety valves on the pressurizer are sized to prevent system pressure from exceeding the design pressure by more than 10%, in accordance with Section III of the ASME B&PV Code. The capacity of the pressurizer safety valves is determined from considerations of: (1) the Reactor Protection System (RPS) and (2) accident or transient conditions which may potentially cause overpressure.

The combined capacity of the safety valves is equal to or greater than the maximum surge rate resulting from complete loss of load without a direct reactor trip or any other control, except that the safety valves on the secondary plant are assumed to open when the steam pressure reaches the secondary plant safety valve setting.

A report on overpressure protection is contained in Westinghouse Topical Report (WCAP) 7769 and is entitled "Overpressure Protection for Westinghouse PWRs."

5.2.2.1 Reactor Coolant System Pressure Control During Low Temperature Operation

The overall approach to eliminating low temperature overpressure transients incorporates administrative, procedural, and equipment requirements. The Overpressure Mitigating System (OMS) used at Zion Station uses the pressurizer power-operated relief valves (PORVs) to prevent inadvertent operation above the steady-state 10CFR50, Appendix G limits as a result of a mass of heat injection transient. This system is also called the Low Temperature Over-pressure Protection system (LTOP). A manual switch is used to enable and disable the low setpoint of each PORV. An enabling alarm which monitors system pressure, the position of the enabling switch, and the isolation valve upstream of the PORV is provided. The system low setpoint is enabled at a temperature of 320°F during plant cooldown and is disabled at the same temperature during plant heatup. See Section 5.4.13 for a discussion on the normal operation of the PORVs.

5.2.2.1.1 Air Supply to the Power-Operated Relief Valves

The PORVs are spring-loaded-closed, air-required-to-open valves. Air is supplied by the Instrument Air System. To assure operability of the valves upon loss of instrument air, a backup air supply is provided. The backup air supply consists of a seismically qualified passive air accumulator for each PORV. Each tank contains enough air to assure that it will still provide the required number of cycles for 10 minutes.

5.2.2.1.2 Design Criteria

The design basis criteria and the design used to meet these criteria are as follows:

1. Operator Action - "No credit can be taken for operator action for ten minutes after the operator is aware, through an action alarm, that an overpressure transient is in progress."

   The Zion Station OMS, when manually enabled, is designed to automatically perform its function for at least 10 minutes after the operator is aware of the transient through an action alarm.

2. Single Failure - "The system shall be designed to protect the reactor vessel given a single failure in addition to the failure that initiated the overpressure transient."

   The Zion Station OMS provides complete redundancy and meets the single failure criterion. One-of-two pneumatically operated PORVs provides the required relief capacity for the OMS; the second PORV provides redundant relief capacity. Each OMS channel has an air accumulator tank that provides a 10-minute backup air supply to operate the PORV when there is a loss of the primary air supply. Each OMS channel includes sensors, actuating mechanisms, alarms, and valves to prevent an RCS overpressure transient. Complete electrical independence and separation are maintained in both OMS channels.

3. Testability - "The system must be testable on a periodic basis consistent with the frequency that the system is relied upon for low temperature overpressure protection."

   The OMS is designed to allow testing prior to its use. The system will be calibrated during each refueling outage and a functional test will be performed before each use. The functional test excludes stroking the PORVs.

4. Seismic and IEEE 279 Criteria - "The system should meet both Seismic Category I and IEEE 279 criteria. The basic objective is that the system should not be vulnerable to a common failure that would both initiate a pressure transient and disable the overpressure mitigating

system. Such events as loss of instrument air and loss of offsite power must be considered."

The circuitry of each OMS channel is electrically and physically separated from each other. The seismic design of equipment presently installed is maintained for the OMS. The OMS has been designed such that no common mode failure will disable the system.

5. Isolation Valve Alarm - "Provide an alarm that monitors the position of the pressurizer relief valve isolation valves, associated with the low setpoint enabling switch, to assure that the overpressure mitigating system is properly aligned."

   The "Low Temp Overpressurization Protection Not in Service" alarm is annunciated whenever the RCS temperature is below 320°F and both OMS channels are not enabled. The alarm monitors the positions of the selector switches and the PORV isolation valves to ensure that both channels are enabled.

5.2.2.1.3 Procedures

Procedures aid in the prevention of low temperature overpressurization transients resulting from mass (coolant addition) or heat input to the RCS. The procedures for startup (and jogging) of a reactor coolant pump (RCP) require that, at RCS temperatures above 140°F, a steam bubble be established in the pressurizer prior to pump start. Otherwise, the RCS temperature is heated by decay heat to the temperature required for bubble formation or the steam generator shell-side temperature is monitored to assure that it is in equilibrium with the RCS temperature. When starting a reactor coolant pump, when no reactor coolant pumps are running, the temperature in the secondary side of the steam generator in the loop in which the reactor coolant pump is to be started shall be less than 50^0 F higher than the RCS temperature. Also, at least one RCP is operated throughout a normal cooldown to 140°F to ensure that the steam generator follows the RCS temperature.

Both Safety Injection (SI) pumps are de-energized by procedure below 320°F to prevent inadvertent starts. In addition, the discharge valves are closed and power is removed. Above 320°F, the maximum allowable pressure by Appendix G is above the shutoff head of the SI pumps. Thus, it is acceptable to have both SI pumps on line above 320°F. Also, two of the three charging pumps are tagged out of service immediately following initiation of the Residual Heat Removal (RHR) System.

5.2.2.1.4 Administrative Controls and Testing

Technical Specifications contains the limiting conditions for operation and the associated surveillance requirements for ensuring low temperature overpressurization protection.

5.2.2.1.5 Methodology for the LTOP Enable Temperature

The enable temperature setpoint must be calculated for the largest possible plant heatup rate using the equation, ET = RTNDT + 90°F + delta T. In order to remain consistent with existing administrative limits and to minimize impact on existing procedures, a heatup rate of 60°F/hr is utilized. To further simplify administrative changes and minimize unit differences in procedures, a single enabling temperature is developed based on the most conservative (highest) value of RTNDT + 90°F for the two units. From Table A1 of Westinghouse document FDRT-SRPLO-229-92 "Methodology for Calculation Enable Temperature Set Point for Zion Units 1&2" (at 14 EFPY) the most limiting value for RTNDT + 90°F is 296°F. Referring to Table 1 of the above reference and utilizing a metal temperature at the controlling location (1/4 T) of 300°F (as a conservative representation for the 296°F value) a water to metal temperature difference of 15°F is obtained. Add this delta T to the 296°F value for RTNDT + 90 yields an enabling temperature of 311°F. In order to provide an easily retained value, the enabling temperature is rounded up (conservative direction) to 320°F.

5.2.2.1.6 Methodology for LTOP PORV Setpoint

The requirements for installation of this system in PWRs arose as a result of numerous overpressure events that occurred during solid plant operations. Consistent with current and historical methodology, the performance of the LTOP system should be evaluated based on the plant response to two design basis events:

1. The pressure transient resulting from spurious isolation of letdown (RHR) concurrent with charging flow control failed to full flow (a single charging pump supplying flow).

2. The pressure transient resulting from the start of a reactor coolant pump with the steam generator secondary side water at a temperature 50°F higher than the NSSS loop and vessel water.

The setpoint methodology utilizes an LTOP version of the LOFTRAN code to determine the PORV pressure overshoot (the difference between a setpoint pressure and peak RCS pressure) during the assumed transients. This information is then used to determine a maximum setpoint pressure such that the current steady state Appendix G curve (without random pressure instrument uncertainties) is not exceeded. In effect, the setpoint plus the PORV pressure overshoot must not exceed the applicable Appendix G curve.

There are several input parameters to the LOFTRAN code that may change over the life of the plant. When it is anticipated that such changes will occur, the LTOP setpoint analysis should be updated to reflect such changes. The significant parameters/factors are:

1. RCS active volume.
2. PORV stroke times.
3. PORV throttling characteristics (C_v).
4. Pressure sensor signal delay times.
5. Steam Generator Type or heat transfer coefficient.
6. Centrifical Charging Pump curve (Head vs. Flow).
7. Fuel Geometry (Core delta-pressure).
8. RCP or RH pump curve (Head vs. Flow).

Instrument uncertainties are excluded in the LTOP Actuation setpoint determination on the basis that these uncertainty terms are insignificant when compared to the margin terms included in the ASME Section III Appendix G methodology. Specifically, the pressure stress is multiplied by a factor of two (in addition to other conservatisms), resulting in conservative stress intensity values.

Because the operation of RCPs and RH pumps introduces uni-directional error in the non-conservative direction for the wide range pressure transmitters (PT-403 and 405 which sense RCS Hot Leg pressure), error margins are calculated for the operation of these pumps in various combinations. These error margins should then be applied to the calculations (either shifting the Appendix G curve down in pressure by the magnitude of the error, or by adding the error to the pressure overshoot determined in the LOFTRAN analysis) to determine a maximum setpoint for the pump combination considered.

In order to facilitate timely plant heatup and maintain a broad pressure band for plant operation, it may be desirable (and is acceptable) to perform the calculations to support the operation of only one RCP (plus necessary RH pumps) up to a given temperature where the margin between the setpoint plus PORV overshoot, and the applicable Appendix G curve, exceeds the pressure error margin attributable to the operation of two RCPs (plus required RH pumps). This temperature then represents the minimum required temperature to start the second RCP. Once two RCPs are operating, the necessary RCS temperature for drawing a bubble in the pressurizer should be easily attained. This methodology can be utilized to determine a minimum required for any given combination of RCP/RH pumps.

5.2.3 Reactor Coolant Pressure Boundary Materials

5.2.3.1 Material Specifications

Each of the materials used in the RCS is selected for the expected environment and service conditions. The major component materials are listed in Table 5.2-3.

No components of the Nuclear Steam Supply System (NSSS) were designed or fabricated outside of the United States.

Raw material, in the form of rough castings for parts of components, was procured from foreign suppliers, as follows:

Zion 1
Two steam generator channel heads from Japan Steel

Zion 2
One steam generator channel head from Japan Steel
Two steam generator channel heads from Mitsubishi Steel
Three main coolant pump casings from Japan Steel

For these raw materials, the following procedure was followed:

Suppliers' capabilities were evaluated to ensure that they were able to manufacture quality materials.

All specifications used in procurement were identical to the specifications utilized for domestic procurement.

Resident quality assurance coverage was maintained to ensure compliance with specifications. Documentation of suppliers' performance and Westinghouse evaluations is available at Westinghouse offices.

The NDTT of the reactor vessel material opposite the core is established at a Charpy V-notch test value of 30 ft-lb or greater. The material is tested to verify conformity to specified requirements and to determine the actual NDTT value. In addition, this material is 100% volumetrically inspected by ultrasonic test using both straight beam and angle beam methods.

The methods used to measure the initial NDTT of the reactor vessel base plate material are given in Appendix 5B. For further information on this subject, refer to Section 5.3.3.

The remaining material in the reactor vessel, and other RCS components, meets the appropriate design code requirements and specific component function.

5.2.3.2 Compatibility with Reactor Coolant

The phenomena of stress-corrosion cracking and corrosion fatigue are not generally encountered unless a specific combination of conditions is present. The necessary conditions are a susceptible alloy, an aggressive environment, stress, and time.

A characteristic of stress-corrosion is that combinations of alloy and environment which result in cracking are usually quite specific. Environments which have been shown to cause stress-corrosion cracking of stainless steels are free alkalinity in the presence of chlorides, fluorides, and free oxygen. With regard to the former, experience has shown that deposition of chemicals on the surface of tubes can occur in a steam blanketed area within a steam generator. In the presence of this environment under very specific conditions, stress-corrosion cracking can occur in stainless steels having the nominal residual stresses resulting from normal manufacturing procedures. However, the steam generator contains Inconel tubes. Testing to investigate the susceptibility of heat exchanger construction materials to stress-corrosion in caustic and chloride aqueous solutions has indicated that Inconel alloy has excellent resistance to general and pitting-type corrosion in severe operating water conditions. Extensive operating experience with Inconel units has confirmed this conclusion.

All RCS materials which are exposed to the coolant are corrosion-resistant. They consist of stainless steels and Inconel, and they are chosen for specific purposes at various locations within the system for their superior compatibility with the reactor coolant. The chemical composition of the reactor coolant is maintained within the specification given in Table 5.2-4. Reactor coolant chemistry is further discussed in Section 5.1.

The water in the secondary side of the steam generators is held within the chemistry specifications given in Table 5.2-5 to control deposits and corrosion inside the steam generators.

5.2.3.2.1 Compatibility with External Insulation

All external insulation of RCS components is compatible with the component materials. The cylindrical shell exterior, closure flanges, bottom head, and closure head of the reactor vessel are insulated with stainless steel metallic reflective insulation. All other external corrosion-resistant surfaces in the RCS are insulated with stainless steel reflective insulation.

5.2.3.3 Fabrication and Processing of Reactor Coolant Pressure Boundary Materials

5.2.3.3.1 Quality Assurance of Welds and Welders

Table 5.2-6 summarizes the quality assurance program with regard to inspections performed on RCS components. In addition to the inspections shown in Table 5.2-6, the equipment supplier performed tests to confirm the adequacy of material received, and tests were performed by the material manufacturer in producing the basic material.

The inspections of the reactor vessel, pressurizer, and steam generators were governed by ASME Code requirements. The inspection procedures and acceptance standards required on pipe materials and piping fabrication were governed by American Standards Association (ASA) B31.1 and Westinghouse requirements and are equivalent to those performed on ASME-coded vessels.

Procedures for performing the examinations were consistent with those established in the ASME Code Section III and were reviewed by qualified engineers. These procedures have been developed to provide the highest assurance of quality material and fabrication. They consider not only the size of the flaws, but equally as important, how the material is fabricated, the orientation and type of possible flaws, and the areas of most severe service conditions. In addition, the accessible external surfaces of the primary RCS pressure-containing segments receive a 100% surface inspection by magnetic particle or liquid penetrant testing after hydrostatic test (see Table 5.2-6). All reactor vessel plate material was subjected to angle beam, as well as straight beam, ultrasonic testing to give maximum assurance of quality. All reactor vessel forgings received the same inspection. In addition, 100% of the material volume was covered in these tests as an added assurance over the grid basis required in the Code.

Quality Control engineers monitored the supplier's work, witnessing key inspections not only in the supplier's shop, but in the shops of subvendors of the major forgings and plate material. Normal surveillance included verification of records of material, physical and chemical properties, review of radiographs, performance of required tests, and qualification of supplier personnel.

Section III of the ASME Code requires that nozzles carrying significant external loads be attached to the shell by full-penetration welds. This requirement was carried out in the reactor coolant piping, where all auxiliary pipe connections to the reactor coolant loop were made using full-penetration welds.

The RCS components were welded under procedures which required the use of both preheat and postheat. Preheat requirements, not mandatory under Code rules, were performed on all weldments including P1 and P3 materials which are the materials of construction in the reactor vessel, pressurizer, and steam generators. Preheat and postheat of weldments both served a common purpose: the production of tough, ductile metallurgical structures in the completed weldment. Preheating produces tough ductile welds by minimizing the formation of hard zones whereas postheating achieves this by tempering any hard zones which may have formed due to rapid cooling.

Those core structural load-bearing components susceptible to severe sensitization were given special heat treatment at elevated temperatures (1600° to 1800°F) for an extended period of time and cooled slowly to ambient temperatures. This allowed the chromium in the steel to migrate back to the grain boundaries thereby effecting "desensitization" of the steel. The validity of this approach has been confirmed by Strauss tests on representative stainless steel samples subjected to similar heat treatment.

The reactor pressure vessel's bottom head instrument nozzle and the safe-ends of the pressurizers were modified to eliminate severely sensitized stainless steel. The bottom head instrument nozzles on Zion were replaced with Inconel nozzles. The pressurizer safe-ends and welds on the surge, spray, relief, and safety nozzles were removed. The nozzles were manually buttered with Inconel 182 (Sb 295) and stress relieved. Type 316 stainless steel safe-ends were welded back on with Inconel 182 (Sb 295). The new, unsensitized safe-ends restore the original overall dimensions and have original weld preps machined on. The cladding and thermal sleeves were replaced.

Other equipment nozzles of concern (such as reactor vessel coolant nozzles) have weld overlay safe-ends using minimal quantities of stainless steel.

All stainless steel piping systems are fabricated in accordance with the specifications in the following paragraph to minimize the occurrence of sensitized austenitic stainless steel.

All austenitic stainless steel welds contain a controlled amount of ferrite. The supplier or contractor shall verify the ferrite content of 5% to 15% in the as-deposited weld metal by making quantitative measurements on all weld metal pads with the Severin or Magna-gauge. The supplier or contractor may check also the certified chemical analysis of the electrodes to be used against the Schaefler diagram (American Welding Society (AWS) Welding Handbook, Section I, Table 4.17). The Cr, Mo, Si, Cb, Ni, C, and Mn content, when plotted as chrome and nickel equivalents on the Schaefler diagram, shall indicate a ferrite content of 5% to 15%. In making the weld pads, the interpass temperature shall be limited to 300°F maximum. The procedures for control of, and testing the ferrite content of, weld pads shall be submitted to the consulting engineers for approval. One certified copy of all chemical test reports of electrodes and results of quantitative ferrite tests of weld pads were also submitted to the consulting engineers.

Nitrogen was added to enhance the strength of the reactor coolant pipe. The pipe material is seamless, forged, ASTM A376 Type 316. Mechanical properties were obtained for both room and 650°F temperatures. Yield strength values at 650°F were required to meet ASME Section III Code Table N-424. Initially, pipe material of A376 standard chemistry analysis was found to possess borderline mechanical properties at 650°F. To improve these properties, controlled nitrogen addition was developed, which evolved into the ASME Code Case 1423. Westinghouse has applied the Code Case chemical analysis to pipe but has not as yet applied the higher allowable design stresses permitted by the Code Case. Also, based on tests performed on similar material, it is concluded that the nitrogen addition does not adversely affect the corrosion resistance of this material in the pressurized water reactor (PWR) coolant environment.

Since nitrogen was added to stainless steel, the statements concerning nitrogen addition to stainless steel for the purpose of enhancing the strength of the material and steps taken to avoid gas entrapment at high points or nonflowing parts have been answered in topical reports. The first of these reports is referenced in Chapter 4 of the FSAR, and the second report (WCAP-7735) was submitted during July 1971 (see Reference 1).

The material of the pressurizer nozzle safe-ends is 316L type stainless steel.

Selected high points and nonflowing locations within the RCS are provided with vents. Whenever RCS pressure is reduced below the pressure required to maintain hydrogen in solution, the system is vented prior to heatup. Also prior to heatup, hydrazine is added to scavenge oxygen from the system.

5.2.3.3.2 Electroslag Weld Quality Assurance

The 90-degree elbows used in the reactor coolant loop piping are electroslag welded. The following efforts were performed for quality assurance of these components:

1. The electroslag welding procedure employing one-wire technique was qualified in accordance with the requirements of ASME B&PV Code Section IX and Code Case 1355 plus supplementary evaluations as requested by WNES-PWRSD. The following test specimens were removed from a five-inch-thick weldment and successfully tested. They are:

    a. 6 Transverse Tensile Bars - as welded
    b. 6 Transverse Tensile Bars - 2050°F, H_2O Quench
    c. 6 Transverse Tensile Bars - 2050°F, H_2O Quench + 750°F stress relief heat treatment
    d. 6 Transverse Tensile Bars - 2050°F, H_2O Quench, tested at 650°F
    e. 12 Guided Side Bend Test Bars

2. The casting segments were surface-conditioned for 100% radiographic and penetrant inspections. The acceptance standards were ASTM E-186 severity level 2, except no category D or E defectiveness was permitted, and USA Standard (USAS) Code Case N-10, respectively.

3. The edges of the electroslag weld preparations were machined. These surfaces were penetrant-inspected prior to welding. The acceptance standards were USAS Code Case N-10.

4. The completed electroslag weld surfaces were ground flush with the casting surface. Then, the electroslag weld and adjacent base material were 100%, radiographed in accordance with ASME Code Case 1355. Also, the electroslag weld surfaces and adjacent base material were penetrant inspected in accordance with USAS Code Case N-10.

5. Weld metal and base metal chemical and physical analysis were determined and certified.

6. Heat treatment furnace charts were recorded and certified.

Reactor coolant pump casings fabricated by electroslag welding were qualified as follows:

1. The electroslag welding procedure employing two- and three-wire technique was qualified in accordance with the requirements of the ASME B&PV Code Section IX and Code Case 1355 plus supplementary evaluations as requested by WNES-PWRSD. The following test specimens were removed from an 8-inch-thick and from a 12-inch-thick weldment and successfully tested for both the two-wire and the three-wire techniques, respectively. They are:

   a. Two-wire electroslag process - 8-inch-thick weldment.

      - 6 Transverse Tensile Bars - 750°F post weld stress relief
      - 12 Guided Side Bend Test Bars

   b. Three-wire electroslag process - 12-inch-thick weldment

      - 6 Transverse Tensile Bars - 750°F post weld stress relief
      - 17 Guided Side Bend Test Bars
      - 21 Charpy V-Notch Specimens
      - Full section macroexamination of weld and heat-affected zone
      - Numerous microscopic examinations of specimens removed from the weld and heat-affected zone regions.
      - Hardness survey across weld and heat-affected zone.

2. A separate weld test was made using the two-wire electroslag technique to evaluate the effects of a stop and restart of welding by this process. This evaluation was performed to establish proper procedures and techniques as such an occurrence was anticipated during production applications due to equipment malfunction, power outages, etc. The following test specimens were removed from an eight-inch thick weldment in the stop-restart-repaired region and successfully tested. They are:

   - Two Transverse Tensile Bars - as welded
   - Four Guided Side Bend Test Bars
   - Full section macroexamination of weld and heat-affected zone.

3. All of the weld test blocks above were radiographed using a 24-Mev Betatron. The radiographic quality level as defined by ASTM E-94 obtained was between $^1/_2$% to 1%. There were no discontinuities evident in any of the electroslag welds.

   a. The casting segments were surface-conditioned for 100% radiographic and penetrant inspections. The radiographic acceptance standards were ASTM E-186 severity level 2, except no category D or E defectiveness was permitted, for section thickness up to $4^1/_2$ inches and ASTM E-280 severity level 2 for section thicknesses greater than $4^1/_2$ inches. The penetrant acceptance standards were ASME B&PV Code Section III, paragraph N-627.

   b. The edges of the electroslag weld preparations were machined. These surfaces were penetrant-inspected prior to welding. The acceptance standards were ASME B&PV Code Section III, paragraph N-627.

   c. The completed electroslag weld surfaces were ground flush with the casting surface. Then, the electroslag weld and adjacent base material were 100% radiographed in accordance with ASME Code Case 1355. Also, the electroslag weld surfaces and adjacent base material were penetrant-inspected in accordance with ASME B&PV Code Section III, paragraph N-627.

   d. Weld metal and base metal chemical and physical analyses were determined and certified.

   e. Heat treatment furnace charts were recorded and certified.

5.2.3.3.3 In-Process Control of Variables

There are many variables that must be controlled in order to maintain desired quality welds. These, together with an explanation of their relative importance are as follows:

Heat Input vs Output -

The heat input is determined by the product of volts times current and they are measured by voltmeters and ammeters which are considered accurate, as they are calibrated every 30 days. During any specific weld, these meters are constantly monitored by the operators.

The ranges specified are 500 to 620 amperes and 44 to 50 volts. The amperage variation, even though it is less than ASME allows by Code Case 1355, is necessary for several reasons:

1. The thickness of the weld is in most cases the reason for changes.

2. The weld gap variation during the weld cycle will also require changes. For example, the procedure qualifications provide for welding thicknesses from 5 to 11 inches with two wires. The current and voltage are varied to accommodate this range.

3. Also, the weld gap is controlled by spacer blocks. These blocks must be removed as the weld progresses. Each time a spacer block is removed there is the chance of the weld pinching down to as much as 1 inch or opening to perhaps as much as $1^1/_2$ inches. In either case, a change in current may be necessary.

4. The heat output is controlled by the heat sink of the section thickness and metered water flow through the water cooled shoes. The nominal temperature of the discharged water is 100°F.

Weld Gap Configuration -

As previously mentioned, the weld gap configuration is controlled by $1^1/_4$ inch spacer blocks. As these blocks are removed, there is the possibility of gap variation. It has been found that a variation from 1 to $1^3/_4$ inches is not detrimental to weld quality as long as the current is adjusted accordingly.

Flux Chemistry -

The flux used for welding is Arcos BV-1 Vertomax. This is a neutral flux whose chemistry is specified by Arcos Corporation. The molten slag is kept at a nominal depth of $1^3/_4$ inches and may vary in depth by plus or minus

Table 5.2-6 summarizes the quality assurance program for all RCS components. In this table, all of the nondestructive tests and inspections which are required by Westinghouse specifications on RCS components and materials are specified for each component. All tests required by the applicable codes are included in this table. Westinghouse requirements, which are more stringent in some areas than those requirements specified in the applicable codes, are also included. The fabrication and quality control techniques used in the fabrication of the RCS were equivalent to those used for the reactor vessel. Additional details regarding reactor cooling loop piping and pump casing electroslag welds are provided in Section 5.2.3.3.2.

Westinghouse requires as part of its reactor vessel specification that certain special tests, which are not specified by the applicable codes, be performed. These tests are listed in the following subsections.

5.2.4.1.1.1 Ultrasonic Testing

During fabrication, Westinghouse requires that a 100% volumetric ultrasonic test of the reactor vessel plate for shear wave be performed in addition to code requirements. This 100% volumetric ultrasonic test is a severe requirement, but it assures that the plate is of the highest quality.

5.2.4.1.2 Inservice Inspection Capability

The Inservice Inspection Program for ASME Class 1, 2, and 3 components is delineated in UFSAR Section 16.3. The ASME B&PV Code, Section XI, is used to determine inservice inspection requirements to the maximum extent possible as required by 10CFR50.55a(g). Details of exceptions from the Code and alternative examinations to ensure structural integrity are updated as necessary in accordance with 10CFR50.55a(g)(5) and (6).

The initial inspections and tests were performed by manufacturers and suppliers in accordance with the applicable code in effect at the time. These tests and inspections, along with the preoperational test program, will form the basis for reference data.

Provisions have been made in the design and arrangement of the RCS, engineered safety features (ESF) systems, and certain associated auxiliary systems to allow access for inservice inspection, to the degree required by the code in effect at the time of design and construction.

With regard to the RCS Components, the layout of the equipment and support structures is designed to permit access to the areas for examination during a plant shutdown. Access implies ability to visually examine surfaces and perform other required examinations.

The accessibility to the external surfaces is possible by the use of removable sections of metallic reflective thermal insulation, which is being provided for all equipment and piping within the Containment structure which require insulation.

5.2.4.1.2 Inservice Inspection Capability

The Inservice Inspection Program for ASME Class 1, 2, and 3 components is delineated in the Zion Technical Specifications Section 4.3.4. The ASME B&PV Code, Section XI, is used to determine inservice inspection requirements to the maximum extent possible as required by 10CFR50.55a(g). Details of exceptions from the Code and alternative examinations to ensure structural integrity are updated as necessary in accordance with 10CFR50.55a(g)(5) and (6).

The initial inspections and tests were performed by manufacturers and suppliers in accordance with the applicable code in effect at the time. These tests and inspections, along with the preoperational test program, will form the basis for reference data.

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With regard to the RCS Components, the layout of the equipment and support structures is designed to permit access to the areas for examination during a plant shutdown. Access implies ability to visually examine surfaces and perform other required examinations.

The accessibility to the external surfaces is possible by the use of removable sections of metallic reflective thermal insulation, which is being provided for all equipment and piping within the Containment structure which require insulation.

Design of this insulation is such that not only will weld areas be available but also general access to reactor coolant piping surfaces and critical safety system piping surfaces will be available.

All critical vessels, pumps, and valves which are located outside of the Containment structure are easily accessible for visual and volumetric examinations.

Inspection techniques identified in ASME Section XI will be utilized where possible. Where new inspection techniques are developed, which would extend existing capabilities, these techniques will be incorporated in the inspection programs where possible.

In general, the scope of baseline inspections was that required by ASME Section XI 1971 to the Summer 1971 Addenda, to the extent that the design of the plant, state of nondestructive testing technology, and access to areas to be inspected would allow.

The Zion Unit 1 and 2 reactor vessel baseline inspections were performed utilizing a remotely operated tool manufactured by Westinghouse. Manual ultrasonic inspections were performed on Unit 1 vessel nozzle welds, safe-end welds, and safe-ends to reactor coolant pipe welds.

Westinghouse developed the tool which inspects vessels of various sizes and satisfies the reactor vessel requirements of Section XI of the ASME Code. The tool is capable of performing remote, submerged inspections of the circumferential, longitudinal, and nozzle welds. This tool can also inspect the ligaments between the threaded flange holes and safe-end welds.

With regards to Steam Generator Eddy Current tube testing, as of November 5, 1984 Code Case N-401 may be used. This code case describes the use of digitized collection and storage of eddy current test data, rather than using strip chart recording (see Reference 2).

5.2.5 Detection of Leakage Through Reactor Coolant Pressure Boundary

5.2.5.1 General

Reactor coolant identified leakage is conservatively set at 10 gpm maximum. The basis for the 10 gpm identified source leakage limit in Technical Specifications is a summation of potential reactor coolant pump leakage past the Number 3 seal and leakage from various valve 2packings from valves in accessible locations. The 10 gpm limit contains allowances for seal degradation on the reactor coolant pumps. Normal makeup and letdown flows from the RCS are 55 gpm and 75 gpm respectively, so that ratios of the leakage limit to the normal makeup and letdown flows are 0.182 and 0.133 respectively. Based on maximum makeup and letdown flows, the ratios are 0.100 and 0.083, respectively.

Reactor coolant unidentified leakage is conservatively set at one gpm. The basis for the one gpm unidentified leakage in Technical Specifications was adopted because the leak source may increase with time or coolant may adversely affect critical components.

The one gpm value was chosen as being conservatively consistent with detectability, plant availability, and good maintenance practices.

Operational experience from other PWRs shows that normal coolant leakage is about 0.5 gpm, average, with a measured range of about 0.2 to 0.9 gpm over a six-month period.

Leakage sources are not known in detail, but leakage is assumed to originate from valve packing and pump seals inside the Containment. This leakage collects in the containment sump by way of the containment floor drains and fan coolers.

These levels of leakage represent operation of PWRs similar in concept, but not in size, to Zion. Furthermore, the PWRs surveyed did not have reactor coolant loop stop valves. No major degradation of reactor coolant pump

5.2.5 Detection of Leakage Through Reactor Coolant Pressure Boundary

5.2.5.1 General

Reactor coolant identified leakage is conservatively set at 10 gpm maximum. The basis for the 10 gpm identified source leakage limit in Section 3.3.3 of the Technical Specifications is a summation of potential reactor coolant pump leakage past the Number 3 seal and leakage from various valve 2packings from valves in accessible locations. The 10 gpm limit contains allowances for seal degradation on the reactor coolant pumps. Normal makeup and letdown flows from the RCS are 55 gpm and 75 gpm respectively, so that ratios of the leakage limit to the normal makeup and letdown flows are 0.182 and 0.133 respectively. Based on maximum makeup and letdown flows, the ratios are 0.100 and 0.083, respectively.

Reactor coolant unidentified leakage is conservatively set at one gpm. The basis for the one gpm unidentified leakage in Section 3.3.3 of the Technical Specifications was adopted because the leak source may increase with time or coolant may adversely affect critical components.

The one gpm value was chosen as being conservatively consistent with detectability, plant availability, and good maintenance practices.

Operational experience from other PWRs shows that normal coolant leakage is about 0.5 gpm, average, with a measured range of about 0.2 to 0.9 gpm over a six-month period.

Leakage sources are not known in detail, but leakage is assumed to originate from valve packing and pump seals inside the Containment. This leakage collects in the containment sump by way of the containment floor drains and fan coolers.

These levels of leakage represent operation of PWRs similar in concept, but not in size, to Zion. Furthermore, the PWRs surveyed did not have reactor coolant loop stop valves. No major degradation of reactor coolant pump seals had been experienced during the period considered. For these reasons, the leakage rates noted above cannot be considered representative of long-term operation of Zion Station.

Positive indications of leakage of coolant from the RCS to the Containment are provided by the following:

1. Leakage through the head-to-vessel closure joint will result in a flow to the leak-off provided between the double gaskets of the closure joint which will show up as a high temperature in this line.

2. Any leakage will cause an increase in the amount of makeup water required to maintain a normal level in the pressurizer.

3. The most sensitive indication of RCS leakage is the Containment Air Monitoring System. Experience at Dresden has shown that the particulate activity in the atmosphere responds very rapidly to increased leakage. Systems are provided to monitor particulate and gaseous activity from the areas enclosing the RCS components so that any leakage from them will be easily detected.

4. A reactor vessel leakage detection system which samples air from around the reactor vessel and compares the activity to a reference activity level in the Containment. The alarm level is a constant value derived from the ratio of the average to the detected activity levels.

5. Relative humidity sensors are provided at the inlet to three of the reactor containment fan cooler intakes. Low-level signals from the relative humidity sensors are transmitted to relative humidity transmitter/indicators mounted on local panels outside of the Reactor Containment. The transmitters, in turn, transmit electrical signals to indicators on the main control boards. Readout is in percent relative humidity.

   The relative humidity meters indicate potential leakage within the Containment at various points in the Containment. The Containment relative humidity indication is used in conjunction with the other leakage detection systems described within this section and is intended to provide verification of a potential leak within the Containment.

6. Other methods of detecting leakage in the Containment are containment pressure, temperature, and the containment sump water level. Primary-to-secondary system leakage will be detected by the air ejector and steam generator blowdown monitors as well as by chemical analyses of secondary water samples. Pressurizer relief tank (PRT) level and reactor coolant drain tank (RCDT) level can also be used to detect leakage from the RCS.

The pressurizer level, Chemical and Volume Control System, and containment instrumentation have indications and/or alarms in the Control Room to monitor and warn the operator to any deviation from normal conditions and can also be used to assist the operator in determining if a leak is occurring. An alarm in the Control Room is also provided for the reactor vessel head to closure joint leakage temperature instrumentation. Resistance temperature detectors (RTDs) are installed on the discharge pipe downstream of the reactor vessel head vent solenoid valves to provide annunciation indication of valve actuation or leakage past these valves.

The containment radiation monitoring channels alarm in the Control Room when the activity exceeds a preset level. When the containment radiation alarm sounds, the operator can have a technician obtain samples of air through filter paper from various areas of the Containment. Four sample lines are installed so that this can be done. These samples will be counted in a beta counter in the Counting Room. The area having the highest air activity level will indicate the approximate location of the leak so that a search can be made in a limited area.

The air particulate monitor filter paper will be replaced periodically and will be counted in a sensitive beta counter in the Counting Room. An average air activity for the period is calculated and recorded so that minor variations in air activity can be discovered and evaluated.

The isotope response curves for these monitors represent the various sensitivities of each detector assembly in relation to the indicated isotopes. These curves are the result of detailed tests performed by the manufacturer, which allows the operator of the radiation monitoring system to determine the relative activity level of various isotopes within the environment being monitored.

The detector sensitivity is tested, in accordance with manufacturer's test procedures, by calibrating the detector to find the proper operating voltage. A count rate versus voltage curve is made upon initial installation of equipment, utilizing a check source, which serves as a reference for future adjustment. After an extended period of operation, the detector check source is used to compare the operating voltage to the reference count rate obtained earlier. If the count rates differ by more than 15%, the voltage is readjusted. If the change in voltage is greater than 50 Vdc, the detector tube is replaced.

During the initial period of operation, radiation monitoring equipment setpoints are set one decade above the minimum sensitivity of the instrument. The typical particulate instrument response time for a ten-fold increase in activity level is approximately eight hours. This means that if, for example, the activity level was a constant 10^{-11} μCi/cc and was increased as a step change to 10^{-10} μCi/cc, the instrument would require eight hours to reach an equilibrium indicated level of 10^{-10} μCi/cc. The indicated level would be increasing during this eight-hour period. The equivalent response time for gaseous monitors is much faster and is on the order of minutes.

In addition, the installed sample lines can be sampled on a regular basis and analyzed to help define variations in airborne activity found in the daily 24-hour composite from the continuous monitor.

Once a leakage rate is reached which requires action and entry of the Containment, the location of the leak will be determined by sampling and visual means. Past experience has indicated that if smears of suspect areas are made and analyzed for radioactivity, it is possible to locate leaks even after the plant is shut down and depressurized.

The leak detection provisions outlined here can detect small leaks from the reactor coolant and associated systems and warn the plant operator in sufficient time to determine the necessary course of action to be followed to maintain the plant in a safe condition.

A more detailed discussion of the radiation monitoring equipment used for leak detection purposes can be found in Chapter 11.

5.2.5.2 Design Basis

1. Potential leakage sources from primary systems are minimized due to the fact that nearly all of the systems are welded, radiographically examined, hydrostatically tested, and installed and inspected with extreme care. During the preoperational primary system hydrostatic tests, all known sources of abnormal leakage were eliminated, with particular attention being paid to welds, valve stems, reactor coolant pump seals, the reactor vessel head to vessel closure flange, and steam generator tubes. All critical areas of the primary systems can be made accessible for visual inspection during plant shutdown. Visual inspection thus provided a valuable means of leakage detection during the preoperational testing and, as required, for postoperational inspections.

2. Supplemental leakage detection provisions, where necessary, are sufficiently sensitive so that any increase in leakage rates can be detected while the total leakage rate is still below a value consistent with continued safe operation of the plant.

3. Where supplemental leakage detection provisions are considered necessary, the following types of detection methods are employed:

    a. Radiation monitors;

    b. System process instrumentation;

    c. Area sampling systems;

    d. Relative humidity monitors; and

    e. Flow sumps (instrumented).

4. Following a loss-of-coolant accident (LOCA), leakage detection provisions in systems required for postaccident cooling meet the following criteria:

    a. The time required for detection of the maximum credible leakage, isolation of the affected system, and transferring the cooling function to the redundant system shall not result in flooding of pump motors.

b. Leakage from cooling systems carrying radioactive fluids prior to isolation shall not result in radiation dose values exceeding the guide limits of 10CFR100 at the site boundary when considered in conjunction with the calculated doses from the LOCA.

5.2.5.3 Residual Heat Removal System Leakage

Separate equipment rooms are provided for each individual RHR pump and heat exchanger. The pump rooms each have a drain collection box which drains to a common sump equipped with two 100 gpm sump pumps. A single sump pump can more than adequately handle the flow which would result from the largest credible leak, failure of the pump seal. The sump capacity is 2530 gallons and is based on the maximum time required to detect and isolate the leak.

As further insurance against the effect of leakage, the pump motor base is located 38 inches off the floor. In the unlikely event that both sump pumps failed, 496,000 gallons could be accumulated on the floor at El 542′ before flooding of the motors could occur. This exceeds the combined volume of the primary loop and the refueling water storage tank (RWST).

Supplemental radiation monitors are provided in the ventilation discharge from each pump and heat exchanger room as a backup to the sump level alarm. Particulate monitors continuously sample each of the pump rooms and heat exchanger rooms. A passive gas monitor will sequentially sample all the pump and heat exchanger rooms on a continuous basis. These monitors have a sensitivity of 5×10^{6} μCi/cc for radionoble gas and a $10^{-10}$$\mu$Ci/cc for radioparticulates.

The RHR heat exchangers are also located in separate rooms. Each room is provided with a leak detection sump which will handle up to 125 gpm and drains to the auxiliary building floor drain analysis tank. These sumps are designed in such a manner that the minimum detectable leak rate is adjustable between zero and two gpm. The time to detect a leak is given in Table 5.2-7.

Supplemental radiation monitors are provided in the ventilation discharge from each heat exchanger room and are part of the same system used to monitor the pump rooms.

In the unlikely event of leakage in other parts of the system, outside of the separate rooms in the Auxiliary Building, any leakage will be detected by means of process instrumentation in each RHR loop. A separate monitor having particulate and iodine channels is provided for the Class I pipe tunnel. In addition, the Class I pipe tunnel is included in the passive gas monitoring system described above for the pump and heat exchanger rooms.

If the radiation monitor for Class I pipe tunnels exceeds the alarm limit, the dampers in the number two section of the ventilation system will align to circulate the exhaust air through charcoal filters.

Offsite doses to the thyroid and whole body, resulting from leakage in the RHR system (outside of Containment during recirculation), will be determined by the methodology given in the Offsite Dose Calculation Manual (ODCM). Source terms for recirculation leakage and resultant offsite doses are addressed in Chapter 15. Without considering the charcoal filters, a 50 gpm leak for 30 minutes would contribute 3.6 rem to the thyroid dose at the site boundary.

Isolation valves are located in such a manner as to allow complete isolation of one RHR loop while the other loop remains in service. Motor-operated valves are located in the system in such a manner as to allow rapid isolation of any desired section of the system. Operating instructions direct the operator in the proper operation of isolation valves to minimize the time required to isolate any portion of the system. Permanently installed flow and pressure instrumentation can also assist the operator in determining if excessive leakage exists within the system. Other leakage detection provisions for portions of the system inside the Containment are discussed in Sections 5.2.5.1 and 5.2.5.2.

5.2.6 References, Section 5.2

1. WCAP-7735, "Topical Report - Sensitized Stainless Steel in Westinghouse Nuclear Steam Supply System," W.S. Hazelton, July 1971, (WNES Proprietary Class 3).

2. Letter dated 11-5-84 from S. A. Varga of NRC to D. L. Farrar of CECo.

3. WCAP-10529 Rev. 1, "Cold Overpressure Mitigating System", November 1985.

4. Westinghouse Letter CWE-93-181, "Commonwealth Edison Company Zion Units 1&2 Evaluation of COMS Analysis", October 4, 1993.

TABLE 5.2-1 (1 of 2)

REACTOR COOLANT SYSTEM - CODE REQUIREMENTS

Component	Codes	Addenda and Code Cases Unit 1	Unit 2
Reactor Vessel	ASME III* CLASS A	1965 ed thru Summer 1966 Addenda, Code Cases 1332-4, 1335-2, 1338-4, 1358-1, 1359-1	Same as Unit 1
Control Rod Drive Mechanism Housings	ASME III* CLASS A	FL-1965 ed thru Summer 1966 Addenda	FL-1965 ed, No Addenda
		Winter 1969 Addenda	PL-1968 ed thru PL-same as Unit 1
Steam Generators			
Tube Side	ASME III* CLASS A	1965 ed thru Summer 1967 Addenda Code Case N-401	Same as Unit 1
Shell Side***	ASME III* CLASS C		
Reactor Coolant Pump Casing	No Code (Design per ASME III Article 4)	1968 ed	Same as Unit 1
Pressurizer	ASME III* CLASS A	1965 ed thru Summer 1967 Addenda	Same as Unit 1
Pressurizer Relief Tank	ASME III* CLASS C	1968 ed thru Winter 1968 Addenda	Same as Unit 1

TABLE 5.2-1 (2 of 2)

REACTOR COOLANT SYSTEM - CODE REQUIREMENTS (continued)

Component	Codes	Addenda and Code Cases Unit 1	Addenda and Code Cases Unit 2
Pressurizer Safety Valves	ASME III*	1968 ed	Same as Unit 1
Reactor Coolant Main Loop Piping and Fittings	USAS B31.1**	Code Cases N7 and N10	Same as Unit 1
Loop Isolation Valves	ASME III CLASS A	1965 ed thru Winter 1968 Addenda	Same as Unit 1
Other Valves	ANSI B16.5 MSS SP-66		
Reactor Coolant Branch Line Piping	USAS B31.1 ****	Code Case N7	Same as Unit 1

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* ASME Boiler and Pressure Vessel Code, Section III, Nuclear Vessels.

** USAS B31.1 Code for Pressure Piping, 1955 edition.

*** The shell side of the steam generator conforms to the requirements for Class A vessels and is so stamped as permitted under the rules of Section III.

**** USAS B31.1 code for Pressure Piping 1967 Edition.

TABLE 5.2-2 (1 of 2)

DESIGN THERMAL AND LOADING CYCLES

		Design Cycles*
1.	Heatup at 100°F/hr (including loop isolations)	200
	Cooldown at 100°F/hr (including loop isolations) (Pressurizer 200°F/hr)	200
2.	Unit Loading at 5% of full power/min	18,300
	Unit Unloading at 5% of full power/min	18,300
3.	Step Load Increase of 10% of full power	2000
	Step Load Decrease of 10% of full power	2000
4.	50% Step Decrease in Load (with steam dump)	200
5.	Loss of Load (without immediate turbine or reactor trip)	80
6.	Loss of Power (blackout with natural circulation in Reactor Coolant System)	40
7.	Loss of Flow (partial loss of flow one pump only)	80
8.	Reactor Trip From Full Power	400
9.	Turbine Roll Test	10
10.	Hydrostatic Test Conditions	
	a. Primary Side Hydrostatic Test Before Initial Startup at 3107 psig	5
	b. Secondary Side Hydrostatic Test Before Initial Startup	5
11.	Primary Side Leak Test	50
12.	Accident Conditions	
	a. Reactor Coolant Pipe Break	1
	b. Steam Pipe Break	1
	c. Steam Generator Tube Rupture	1

TABLE 5.2-2 (2 of 2)

DESIGN THERMAL AND LOADING CYCLES

13. Steady-state fluctuations - the reactor coolant average temperature, for purpose of design is assumed to increase and decrease a maximum of 6°F in one minute. The temperature changes are assumed to be around the programmed value of T_{avg}, $T_{avg} \pm 3°F$. The corresponding reactor coolant average pressure is assumed to vary accordingly and thus be within 2200 and 2300 psia or 2250 $\pm$ 50 psia.

NOTE

ASME case conditions are not provided for by the code in effect, but the following categorization can be made in the itemized listing above:

Normal condition	-	Items 1 through 4, 13
Upset condition	-	Items 5 through 8
Test condition	-	Item 9 through 11
Faulted condition	-	Items 12a, b, c,

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* 1. Estimated for equipment design purposes (40-year life) and not intended to be an accurate representation of actual transients or to reflect actual operating experience.

2. The associated temperature and pressure transients represents an envelope with margin in the number of cycles. As an example, consider reactor trip for which 400 design cycles are considered. One cycle of this transient would represent any operational occurrence which would result in a reactor trip. Thus the reactor trip represents an envelope design approach to various operational occurrences. The same approach applies to the other design transients listed.

TABLE 5.2-3 (1 of 3)

MATERIALS OF CONSTRUCTION OF THE REACTOR COOLANT SYSTEM COMPONENTS

Component	Section	Materials
Reactor Vessel	Pressure Plate	ASTM A-533 Grade B Class 1
	Pressure Forgings	ASTM A508 Class 2
	Cladding, Stainless	Type 304 or equivalent
	Stainless Weld Rod	Type 308, 309, or Type 312
	O-Ring Head Seals	Inconel - 718
	CRDM Housings	SA-182 Type 304
	Lower Tube	SB-167
	Studs	SA-540 Grade B-23
	Instrumentation Nozzles	Inconel SB 167
	Insulation	Stainless Steel
Steam Generator	Pressure Plate	ASTM A-533 Grade A Class 1
	Pressure Forgings	ASTM A-508 Class 2
	Cladding for Heads, Stainless	Type 304 or equivalent
	Stainless Weld Rod or Type 309	Type 304, Type 308L,
	Cladding for Tube Sheets	Inconel
	Tubes	Inconel - 600
	Channel Head Castings	ASTM A-216 Grade WCC

TABLE 5.2-3 (2 of 3)

MATERIALS OF CONSTRUCTION OF THE REACTOR
COOLANT SYSTEM COMPONENTS

Component	Section	Materials Unit 1	Materials Unit 2
Pressurizer	Shell	SA-533 Grade A (Class 1)	SA-533 Grade A (Class 2)
	Heads	SA-216 Grade WCC	SA-513 Grade A (Class 2)
	Support Skirt	SA-516 Grade 70	SA-516 Grade 70
	Nozzle Weld Ends	SA-182 F316	SA-182 F316
	Inst. Tube Coupling	SA-182 F316	SA-182 F316
	Cladding, Stainless	Type 304 or equivalent	Type 304 or equivalent
	Nozzle Forgings		SA-508 Class 2 Mn-Mo
	Internal Plate	SA-240 Type 304	SA-240 Type 304
	Inst. Tubing	SA-213 Type 304	SA-213 Type 304
	Heater Well Tubing	SA-213 Type 316 Seamless	SA-213 Type 316 Seamless
	Heater Well Adaptor	SA-182 F316	SA-182 F316
Pressurizer Relief Tank	Shell	ASTM A-285 Grade C	
	Heads	ASTM A-285 Grade C	
	Internal Coating Amercoat 55		
Pipe	Pipes	ASTM A-376 Type 316	
	Fittings	ASTM A-351 Grade CF8M	
	Nozzles	ASTM A-182 Grade F316	

TABLE 5.2-3 (3 of 3)

MATERIALS OF CONSTRUCTION OF THE REACTOR
COOLANT SYSTEM COMPONENTS

Component	Section	Materials
Pump	Shaft	ASTM A-182 Grade F347
	Impeller	ASTM A-351 Grade CF8
	Casing	ASTM A-351 Grade CF8
Valves	Pressure Containing Parts	ASTM A-351 Grade CF8M

TABLE 5.2-4

REACTOR COOLANT WATER CHEMISTRY SPECIFICATION

Electrical Conductivity	Determined by the concentration of boric acid and alkali present. Expected range is < 1 to 40 μMhos/cm at 25°C.
Solution pH	Determined by the concentration of boric acid and alkali present. Expected values range between 4.2 (high boric acid concentration) to 10.5 (low boric acid concentration) at 25°C.
Oxygen, ppm, max.	0.10
Chloride, ppm, max	0.15
Fluoride, ppm, max.	0.15
Hydrogen, cc (STP)/kg H_2O	25-35
Total Suspended Solids. ppm, max.	1.0
pH Control Agent (Li^7 OH_{H_2O})	0.3×10^{-4} to 4.6×10^{-4} molal (equivalent to 0.22 to 3.2 ppm Li^7)
Boric Acid as ppm B	Variable from 0 to ~4000

TABLE 5.2-4

REACTOR COOLANT WATER CHEMISTRY SPECIFICATION

Electrical Conductivity	Determined by the concentration of boric acid and alkali present. Expected range is < 1 to 40 μMhos/cm at 25°C.
Solution pH	Determined by the concentration of boric acid and alkali present. Expected values range between 4.2 (high boric acid concentration) to 10.5 (low boric acid concentration) at 25°C.
Oxygen, ppm, max.	0.10
Chloride, ppm, max	0.15
Fluoride, ppm, max.	0.15
Hydrogen, cc (STP)/kg H_2O	25-35
Total Suspended Solids. ppm, max.	1.0
pH Control Agent (Li^7 OH)	0.3×10^{-4} to 3.2×10^{-4} molal (equivalent to 0.22 to 2.2 ppm Li^7)
Boric Acid as ppm B	Variable from 0 to ~4000

TABLE 5.2-5

STEAM GENERATOR WATER (STEAM SIDE) CHEMISTRY SPECIFICATION

	Reactor Power > 40%	Reactor Power ≤ 40%	Hot Standby/ Hot Shutdown	Cold Shutdown
pH	> 9.0	> 9.0	> 9.0	9.8-10.5
Cation Conductivity (μmho/cm)	< 0.8	< 2	< 2	
Sodium (ppb)	< 20	< 100	< 100	< 1000
Chloride (ppb)	< 20	< 100	< 100	< 1000
Sulfate (ppb)	< 20	< 100	< 100	< 1000
Dissolved oxygen (ppb)			< 5	
Hydrazine (ppm)				75 - 200

TABLE 5.2-6 (1 of 3)

REACTOR COOLANT SYSTEM
QUALITY ASSURANCE PROGRAM
(DURING CONSTRUCTION)

Component	RT*	UT*	PT*	MT*	ET*
1. Steam Generator					
1.1 Tube Sheet					
1.1.1 Forging		yes		yes	
1.1.2 Cladding		yes(+)	yes(++)		
1.2 Channel Head					
1.2.1 Casting	yes			yes	
1.2.2 Cladding			yes		
1.3 Secondary Shell & Head					
1.3.1 Plates		yes			
1.4 Tubes		yes			yes
1.5 Nozzles (forgings)		yes		yes	
1.6 Weldments					
1.6.1 Shell, longitudinal	yes			yes	
1.6.2 Shell, circumferential	yes			yes	
1.6.3 Cladding (Channel Head-Tube Sheet joint cladding restoration)			yes		
1.6.4 Steam and Feedwater Nozzle to shell	yes			yes	
1.6.5 Support brackets				yes	
1.6.6 Tube to tube sheet			yes		
1.6.7 Instrument connections (primary and secondary)				yes	
1.6.8 Temporary attachments after removal				yes	
1.6.9 After hydrostatic test (all welds and complete channel head - where accessible)				yes	
1.6.10 Nozzle safe ends (weld deposit)	yes		yes		
2. Pressurizer					
2.1 Heads					
2.1.1 Casting	yes			yes	
2.1.2 Cladding			yes		
2.2 Shell					
2.2.1 Plates		yes		yes	
2.2.2 Cladding			yes		
2.3 Heaters					
2.3.1 Tubing(++++)		yes	yes		
2 3.2 Centering of element	yes				
2.4 Nozzle		yes	yes		

TABLE 5.2-6 (2 of 3)

REACTOR COOLANT SYSTEM
QUALITY ASSURANCE PROGRAM
(DURING CONSTRUCTION)

	Component	RT*	UT*	PT*	MT*	ET*
	2.5 Weldments					
	2.5.1 Shell, longitudInal	yes			yes	
	2.5.2 Shell, circumferential	yes			yes	
	2.5.3 Cladding			yes		
	2.5.4 Nozzle Safe End (forging)	yes		yes		
	2.5.5 Instrument Connections			yes		
	2.5.6 Support Skirt				yes	
	2.5.7 Temporary Attachments after removal				yes	
	2.5.8 All welds and cast heads after hydrostatic test				yes	
	2.6 Final Assembly					
	2.6.1 All accessible surfaces after hydrostatic test				yes	
3.	Piping					
	3.1 Fittings and Pipe (Castings)	yes		yes		
	3.2 Fittings and Pipe (Forgings)		yes	yes		
	3.3 Weldments					
	3.3.1 Circumferential	yes		yes		
	3.3.2 Nozzle to runpipe (No RT for nozzles less than 4 inches)	yes		yes		
	3.3.3 Instrument connections			yes		
4.	Pumps					
	4.1 Castings	yes		yes		
	4.2 Forgings					
	4.2.1 Main Shaft		yes	yes		
	4.2.2 Main Studs		yes	yes		
	4.2.3 Flywheel (Rolled Plate)		yes			
	4.3 Weldments					
	4.3.1 Circumferential	yes		yes		
	4.3.2 Instrument connections		yes			
5.	Reactor vessel					
	5.1 Forgings					
	5.1.1 Flanges		yes		yes	
	5.1.2 Studs		yes		yes	
	5.1.3 Head Adapters		yes	yes		
	5.1.4 Head Adapter Tube		yes	yes		
	5.1.5 Instrumentation Tube	yes	yes			

TABLE 5.2-6 (3 of 3)

REACTOR COOLANT SYSTEM
QUALITY ASSURANCE PROGRAM
(DURING CONSTRUCTION)

Component	RT*	UT*	PT*	MT*	ET*
5.1.6 Main Nozzles		yes		yes	
5.1.7 Nozzle Safe Ends (If forging is employed)		yes	yes		
5.2 Plates		yes		yes	
5.3 Weldments					
5.3.1 Main Steam	yes			yes	
5.3.2 CRD Head Adapter Connection			yes		
5.3.3 Instrumentation tube connection			yes		
5.3.4 Main nozzles	yes			yes	
5.3.5 Cladding		yes(+++)		yes	
5.3.6 Nozzle safe-ends (If forging)	yes		yes		
5.3.7 Nozzle safe-ends (If weld deposit)	yes		yes		
5.3.8 Head adaptor forging to head adaptor tube	yes		yes		
5.3.9 All welds after hydrotest				yes	
6. Valves					
6.1 Castings	yes		yes		
6.2 Forgings (No UT for valves two inches and smaller)		yes	yes		

* RT - Radiographic
UT - Ultrasonic
PT - Dye Penetrant
MT - Magnetic Particle
ET - Eddy Current

(+) Flat Surfaces Only
(++) Weld Deposit Areas Only
(+++) UT of Clad Bond-to-Base Metal
(++++) Or a UT and ET

TABLE 5.2-7

TIME TO DETECT A LEAK IN THE RHR HEAT EXCHANGER ROOMS

Alarm Type (seconds)	Leak Rate (gpm)
345	2
69	10
13.8	50

5.3 REACTOR VESSELS

5.3.1 Reactor Vessel Materials

The reactor vessel is cylindrical with a welded hemispherical bottom head and a removable, flanged, and gasketed hemispherical upper head. The vessel contains the core, core support structures, control rods, thermal shield, and other parts directly associated with the core.

The reactor vessel head is bolted to the reactor vessel by closure studs. During refueling operations, when the head is removed, the reactor vessel closure studs are held in a carrier basket placed on the operating floor above the refueling cavity. The studs are protected by a permanent phosphate coating and cleaned prior to reinstallation. The studs are also visually examined for corrosion during each refueling outage and are subjected to the examinations required by ASME Section XI at appropriate intervals. These measures ensure that the reactor vessel studs are not subject to unmonitored corrosive deterioration or deterioration by other mechanisms.

The reactor vessel closure head contains head adaptors, which are tubular members, attached by partial penetration welds to the underside of the closure head. The upper end of these adaptors contain acme threads for the assembly of control rod drive mechanisms or instrumentation adaptors. The seal arrangement at the upper end of these adaptors consists of a welded flexible canopy seal. The vessel has inlet and outlet nozzles located in a horizontal plane just below the vessel flange but above the top of the core. Coolant enters the inlet nozzles,flows down the core barrel and vessel wall annulus, turns at the bottom, and flows up through the core to the outlet nozzles.

The bottom head of the vessel contains penetration nozzles for connection and entry of the nuclear incore detection instrumentation. Each tube is attached to the inside of the bottom head by a partial penetration weld.

The reactor vessel is designed to provide the smallest and most economical volume required to contain the reactor core, control rods, and the necessary supporting and flow-directing internals. Inlet and outlet nozzles are spaced around the vessel. Outlet nozzles are located on opposite sides of the vessel to facilitate optimum layout of the Reactor Coolant System (RCS) equipment. The inlet nozzles are tapered from the coolant loop-vessel interfaces to the vessel inside wall to reduce loop pressure drop.

The reactor vessel flange and head are sealed by two hollow metallic O-rings. Seal leakage is detected by means of two leak-off connections; one between the inner and outer ring, and one outside the outer O-ring. Piping and associated valving are provided to direct any leakage to the reactor coolant drain tank. Leakage will be indicated by a high-temperature alarm from a detector in the leakoff line.

Ring forgings have been used in the following areas of the reactor vessel:

1. Closure Head Flange

2. Vessel Flange

3. Nozzle Shell Course (Upper and Lower Belt)

4. Eight Primary Nozzles

5. Transition Ring

The pressure or strength bearing stainless steel components or parts in the reactor vessel and associated primary loop components that may become furnace sensitized* during the fabrication sequence have been minimized. The areas having sensitization include:

1. Reactor Vessel

   a. Primary nozzle safe-ends (weld metal buttering)
   b. Gasket monitor tubes

2. Steam Generators

   a. Primary nozzle safe-ends (weld metal buttering)

The cylindrical portion of the reactor vessel below the refueling seal ledge is permanently insulated with a metallic reflective-type insulation supported from the reactor coolant nozzles. This insulation consists of inner and outer sheets of stainless steel spaced 3 inches apart with multilayers of stainless steel as the insulating agent. Removable panels of the metallic reflective-type insulation described above are provided for the reactor vessel head and closure region. These panels are supported on the refueling seal ledge and vent shroud support ring. The rest of the closure head is insulated with removable panels of at least three inches of the reflective insulation described. The bottom head is also insulated with reflective insulation, but it is not removable.

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* The term "furnace sensitized" is interpreted as wrought austenitic stainless steel (>0.02C) components which have been post weld heat treated; the temperature and minimum times are consistent with ASME III requirements. A detailed discussion of this subject is contained in Reference 1.

A schematic of the reactor vessel is shown in Figure 5.3-1. The total number of control rod drive mechanism (CRDM) housings given in Figure 5.3-1 includes those penetrations which have been cut and capped. The materials of construction are given in Table 5.2-3 and the design parameters are given on Table 5.3-1. Reactor vessel toughness data for Unit 1 and Unit 2 is given in Tables 5.3-7 and 5.3-8. A description of the reactor vessel internals is given in Chapter 4.

The reactor vessel material is heat-treated specifically to obtain good notch-ductility, which ensures a low nil ductility transition temperature (NDTT). This assures that the finished vessel can be initially hydrostatically tested and operated as near to room temperature as possible without restrictions. The stress limits established for the reactor vessel are dependent upon the temperatures at which the stresses are applied. As a result of fast neutron irradiation in the region of the core, the material properties will change, including an increase in the NDTT. A value of NDTT of +10°F for Unit 1 and +35° for Unit 2 in this region has been established during fabrication.

The techniques used to measure and predict the integrated fast neutron (E > 1 Mev) exposure of the reactor vessel are identical to those described for the irradiation samples. Since the neutron spectrum at the sample can be applied with confidence to the adjacent section of reactor vessel, the vessel exposure will be obtained from the measured sample exposure by appropriate application of the calculated azimuthal neutron flux variation.

The details of the neutron flux analysis can be found in Westinghouse Topical Report (WCAP)-10962, "Zion Units 1&2 Reactor Vessel Fluence and Rt_{pts} Evaluation." Fast neutrons will also be emitted from the Gamma-Metrics fission detectors used for source and intermediate range NIS, but are a negligible contributor to the reactor vessel NDTT effects.

To evaluate the NDTT shift of welds, heat affected zones, and base material for the vessel, test coupons of these material types have been included in the Reactor Vessel Surveillance Program (RVSP) described in Section 5.3.1.1. The current analysis of NDTT shift as a function of EFPH can be found in Westinghouse Topical Report (WCAP)-13406 "Heatup and Cooldown Limit Curves for Normal Operation for Zion Units 1 and 2", July 1992.

5.3.1.1 Material Surveillance

In the surveillance programs, the evaluation of the radiation damage is based on pre-irradiation testing of Charpy V-notch and tensile specimens and post-irradiation testing of Charpy V-notch, tensile, and wedge opening loading (WOL) fracture mechanics test specimens. These programs are directed toward evaluation of the effect of radiation on the fracture toughness of reactor vessel steels based on the transition temperature approach and the fracture mechanics approach and are in accordance with American Society of Testing Materials (ASTM)-E-185-70, "Recommended

Practice for Surveillance Tests for Nuclear Reactor Vessels." The surveillance program does not include thermal control specimens. These specimens are not required since the surveillance specimens will be exposed to the combined neutron irradiation and temperature effects, and the test results will provide the maximum transition temperature shift. Thermal control specimens, as considered in ASTM-E-185-70, would not provide any additional information on which the operational limits for the reactor vessel are set.

The Zion Station RVSP uses eight specimen capsules which exceeds the minimum number recommended by ASTM-E-185-82 and Reg. Guide 1.99 Rev. 2. The capsules are located about three inches from the vessel wall directly opposite the center portion of the core. Sketches of an elevation and plan view showing the location and dimensional spacing of the capsules with relation to the core, thermal shield, and vessel and weld seams is shown in Figure 5.3-3 and 5.3-4, respectively. Figure 5.3-5 shows the weld attachment detail of the specimen guide, which contains the capsule, to the thermal shield. The capsules can be removed when the vessel head is removed and can be replaced when the internals are removed. The capsules contain reactor vessel steel specimens from the shell plates located in the core region of the reactor and associated weld metal and heat-affected zone metal. (As part of the surveillance program, a report of the residual elements in weight percent to the nearest 0.01% will be made for surveillance material base metals and as deposited weld metal.) In addition, 32 correlation monitors made from fully documented specimens of SA-533 Grade B class 1 material obtained through Subcommittee II of ASTM Committee E10, Radioisotopes and Radiation Effects, are inserted in the capsules. The eight capsules contain approximately 32 tensile specimens, 352 Charpy V-notch specimens (which include weld metal and heat-affected zone material), and 32 WOL specimens. Dosimeters including Ni, Cu, Fe, Co-Al, Cd shielded Co-Al, Cd shielded Np-237, and Cd shielded U-238 are placed in filler blocks drilled to contain the dosimeters. The dosimeters permit evaluation of the flux seen by the specimens and vessel wall. In addition, thermal monitors made of low melting alloys are included to monitor the temperature of the specimens. The specimens are enclosed in a tight fitting stainless steel sheath to prevent corrosion and to ensure good thermal conductivity. The complete capsule is helium leak tested. Vessel material sufficient for at least two capsules will be kept in storage should the need arise for additional replacement test capsules in the program.

The RVSP meets the intent of the NRC fracture toughness requirements for nuclear power reactors 10CFR50, Appendix H. The specimens included in the program are outlined in Table 5.3-5.

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The fast neutron exposure of the specimens occurs at a faster rate than that experienced by the adjacent vessel wall because the specimens are located between the core and the vessel. Since these specimens experience accelerated exposure and are actual samples from the materials used in the vessel, the NDTT measurements are representative of the vessel at a later time in life. Data from WOL fracture toughness samples are expected to provide additional information for use in determining allowable stresses for irradiated material.

The calculated maximum fast neutron exposure (nvt) at the vessel wall is 2.5×10^{19} nvt > 1 Mev. The reactor vessel surveillance capsules are located at 4 degrees and 40 degrees as shown in Figure 5.3-4. The relative exposures of the capsules and the adjacent vessel wall and the vessel maximum are listed below:

Capsules at	Lead Adjacent Vessel Wall by Multiplying Factor of:	Lead Vessel Maximum by a Multiplying Factor of:
4 degrees	2.6	0.6
40 degrees	2.7	2.6

Correlations between the calculations and the measurements on the irradiated samples in the capsules, assuming the same neutron spectrum at the samples and the vessel inner wall, are described in Appendix 5B and have indicated agreement. The calculation of the integrated flux at the vessel wall is conservative by up to 20%.

The anticipated degree to which the specimens will perturb the fast neutron flux and energy distribution will be considered in the evaluation of the surveillance specimen data. Verification and possible readjustment of the calculated wall exposure will be made by use of data on all capsules withdrawn.

Specimen capsules to be used in the Reactor Vessel Material Surveillance Program will be withdrawn during the refueling period either immediately preceding or following the effective full power years (EFPY) of unit life as shown in Table 5.3-6.

The Babcock & Wilcox (B&W) Owners Group Materials Committee Reactor Vessel Working Group maintains a Master Integrated Reactor Vessel Material Surveillance Program (MIRVP) for those reactor vessels fabricated by B&W that contain automatic submerged arc welds fabricated with Mn-Mo-Ni weld wire and Linde 80 flux (see Reference 2). The Zion reactor vessels, being of that manufacture, are components of the MIRVP. In accordance with the cooperative character of the MIRVP, the Zion Station RVSP is included in the MIRVP and information obtained from the plant-specific RVSPs of the other MIRVP members is interpreted for the benefit of Zion Station.

5.3.2 Pressure-Temperature Limits

5.3.2.1 Design Pressure

The RCS design and operating pressures, the safety, power-operated relief, and pressurizer spray valves setpoints, and the protection system setpoint pressures are listed in Table 5.1-2. The selected design margin includes operating transient pressure changes from core thermal lag, coolant transport times and pressure drops, instrumentation and control response characteristics, and system relief valve characteristics. Table 5.3-2 gives the design pressure drop of the RCS components.

The RCS serves as a barrier preventing radionuclides contained in the reactor coolant from reaching the atmosphere. In the event of a fuel cladding failure, the RCS is the primary barrier against the uncontrolled release of fission products. By establishing a system pressure limit, the continued integrity of the RCS is assured. Thus, the safety limit of 2735 psig (110% of design pressure) has been established. This represents the maximum transient pressure allowable in the RCS under the ASME Code, Section III.

5.3.2.2 Design Temperature

The design temperature for each component is selected to be above the maximum coolant temperature in that component under all normal and anticipated transient load conditions. The design and operating temperatures of the respective system components are listed in Tables 5.3-1, 5.4-1, 5.4-2, 5.4-13, 5.4-22 and 5.4-23.

5.3.2.3 Maximum Heating and Cooling Rates

The RCS operating cycles used for design purposes are given in Table 5.2-2 and described in Section 5.3.3.3. The normal system heating and cooling rate is 50°F/hr. Sufficient electrical heaters are installed in the pressurizer to permit a heatup rate of 55°F/hr, starting with a minimum water level. This rate takes into account the small continuous spray flow provided to maintain the pressurizer liquid homogeneous with the coolant. The fastest cooldown rates which result from the hypothetical case of a break of a main steamline are discussed in Chapter 15. Operating limits for the RCS with respect to heatup and cooldown rates are defined in the Technical Specifications and Westinghouse Topical Report (WCAP)-13406 "Heatup and Cooldown Limit Curves for Normal Operation for Zion Units 1 and 2."

The stress level of material in the reactor vessel, or in other RCS components, is a combination of stresses caused by internal pressures and by thermal gradients. The latter are significant as they may result from a rate of change of reactor coolant temperature. Operating restrictions are imposed to limit the combined stresses based upon the proximity to design transition temperature (DTT), as described in Section 5.3.3.2.1. The DTT is defined as the initial NDTT plus the increase in NDTT due to irradiation experienced plus 60°F. Curves are incorporated in the plant Technical Specifications which define the operating limits. To establish the latter, an adjustment is made for the maximum expected NDTT shift which the reactor vessel material will experience because of the fast neutron dose it will receive. The predicted shift will be verified by the surveillance program testing. The limits for initial operation are used to define operational limitation and these curves are periodically updated to reflect irradiation exposure of the vessel and the results of the surveillance program.

5.3.3 Reactor Vessel Integrity

The reactor vessel is the only component of the RCS which is exposed to a significant level of neutron irradiation and it is therefore the only component which is subject to material radiation damage effects.

The NDTT shift of the vessel material and welds during service due to radiation damage effects is monitored by the RVSP. Details are given in Section 5.3.1.1.

Reactor vessel design is based on the transition temperature method of evaluating the possibility of brittle fracture of the vessel material as a result of operation.

As part of the design control on materials, Charpy V-notch toughness tests are run on all ferritic material used in fabricating pressure parts of the reactor vessel, steam generator, and pressurizer to provide assurance for hydrotesting and operation in the ductile region at all times. In addition, drop-weight tests and Charpy V-notch transition temperature evaluations are performed on the reactor vessel materials.

As an assurance of system integrity, all components in the system are hydrotested at 3107 psig prior to initial operation.

The safety of the reactor vessel and all other RCS pressure containing components and piping is dependent on several major factors including design and stress analysis, material selection and fabrication, quality control and operations control.

5.3.3.1 Reactor Internals

The Indian Point II reactor was the prototype for the Westinghouse four-loop plant internals verification program. All subsequent four-loop plants, including Zion, are essentially identical in design. Past experience with other reactors indicates that plants of similar designs behave in a similar manner. For these reasons a comprehensive instrumentation program was conducted on the Indian Point Plant to confirm the behavior of the reactor components. The main objectives of this test were to increase confidence in the adequacy of the internals by determining stress or deflection levels at key locations and to obtain data that could be used to develop improved analytical tools for prediction of internals vibration. The final report was published as WCAP-7879 entitled "Four-Loop PWR Internals Assurance and Test Program." Additionally, a test on the primary coolant loop to determine natural frequencies, mode shapes, damping and vibration during pump operation was conducted and the results were published in WCAP-7920 entitled "Primary Loop Vibration Test Program."

5.3.3.2 Reactor Vessel

The following reactor pressure vessel components are analyzed in detail through systematic analytical procedures.

1. Control Rod Housings
2. Closure Head Flange and Shell
3. Main Closure Studs

4. Inlet Nozzle (and Vessel Support)
5. Outlet Nozzle (and Vessel Support)
6. Vessel Wall Transition
7. Core-Barrel Support Pads
8. Bottom Head to Shell Juncture
9. Bottom Head Instrument Penetrations, etc.

5.3.3.2.1 Method of Analysis

Item (1). An interaction analysis is performed on the CRDM housing. The flange is assumed to be a ring and the tube a long cylinder. The different values of Young's Modulus and coefficients of thermal expansion of the tubes are taken into account in the analysis. The local flexibility is considered at appropriate locations. The closure head is treated as a perforated spherical shell with modified elastic constants. The effects of redundants on the closure head are assumed to be local only. Using the mechanical and thermal stresses from this analysis, a fatigue evaluation is made for the J weld. Seismic loadings are considered in the stress analysis as primary loadings and are included in the fatigue analysis.

Item (2). The closure head, closure head flange, vessel flange, vessel shell and closure studs are all evaluated in the same analysis. An analytical model is developed by dividing the actual structure into different elements such as sphere, ring, long cylinder and cantilever beam, etc. An interaction analysis is performed to determine the stresses due to mechanical, thermal and seismic loads. These stresses are evaluated in light of the strength and fatigue requirements of the ASME Boiler and Pressure Vessel (B&PV) Code Section III.

Item (3). A similar analysis is performed for the vessel flange to vessel shell juncture and the main closure studs.

Item (4). For the analysis of nozzle and nozzle-to-shell juncture, the loads considered are internal pressure, operating transients, thermally induced and seismic pipe reactions, static weight of vessel, earthquake loading and expansion and contraction, etc. A combination of methods is used to evaluate the stresses due to mechanical and thermal loads and external loads resulting from seismic pipe reactions, earthquake and pipe break, etc.

For fatigue evaluation, peak stresses resulting from external loads and thermal transients are determined by concentrating the stresses as calculated by the above described methods. Combining these stresses enables the fatigue evaluation to be performed.

Item (5). Method of analysis for outlet nozzle and vessel supports is the same as described above.

Item (6). Vessel wall transition is analyzed by means of a standard interaction analysis. The thermal stresses are determined by the skin stress method, where it is assumed that the inside surface of the vessel is at the same temperature as the reactor coolant, and the mean temperature of the shell remains at the steady state temperature. This method is considered conservative.

Item (7). Thermal, mechanical, and pressure stresses are calculated at various locations on the core barrel support pad and at the vessel wall. Mechanical stresses are calculated by the flexure formula for bending stress in a beam; pressure stresses are taken from the analysis of the vessel to bottom head juncture; and thermal stresses are determined by the conservative method of skin stresses. The stresses due to the cyclic loads are multiplied by a stress concentration factor where applicable and used in the fatigue evaluation.

Item (8). The standard interaction analysis and skin stress methods are employed to evaluate the stresses due to mechanical and thermal stresses respectively. The fatigue evaluation is made on a cumulative basis where superposition of all transients is taken into consideration.

Item (9). An interaction analysis is performed by dividing the actual structure into an analytical model composed of different structural elements. The effects of the redundants on the bottom head are assumed to be local only. It is also assumed that for any condition where there is interference between the tube and the head, no bending at the weld can exist. Using the mechanical and thermal stresses from this analysis a fatigue evaluation is made for the J weld. The location and geometry of the areas of discontinuity and/or stress concentration are shown in Figures 5.3-6, 5.3-7, and 5.3-8.

A summary of the estimated primary plus secondary stress intensity for components of the reactor vessel and the estimated cumulative fatigue usage factors for the components of the reactor vessel is given in Tables 5.3-3 and 5.3-4, respectively.

The cycles specified for the fatigue analysis are the results of an evaluation of the expected plant operation coupled with experience from nuclear power plants now in service, such as Yankee-Rowe.

The conservatism of the design fatigue curves used in the fatigue analysis has been demonstrated by the Pressure Vessel Research Committee (PVRC) in a series of cyclic pressurization tests of model vessels fabricated to the Code. The results of the PVRC tests showed that no crack initiation was detected at any stress level below the code allowable fatigue curve and that no crack progressed through a vessel wall in less than three times the allowable number of cycles. Similarly, fatigue tests have been performed on irradiated pressure vessel steels with comparable results (see Reference 3).

The vessel design pressure is 2485 psig while the normal operating pressure will be 2235 psig. The resulting operating membrane stress is therefore amply below the code allowable membrane stress to account for operating pressure transients.

The stress allowed in the vessel in relation to operation below NDTT and DTT to preclude the possibility of brittle failure are:

1. At DTT; a maximum stress of 20% yield

2. From DTT to DTT minus 200°F; a maximum stress decreasing from 20% to 10% yield

3. Below DTT minus 200°F; a maximum stress of 10% yield

These limits are based on a conservative interpretation of the Fracture Analysis Diagram developed at the Naval Research Laboratory (References 1, 4 and 5) after many years of research and are confirmed by extensive correlations with service failures. There have been no known service failures under conditions permitted by these limits. The Fracture Analysis Diagram is the most widely known and generally accepted criterion for brittle fracture prevention and includes linear elastic fracture mechanics concepts. These limits established by the Fracture Analysis Diagram have been correlated with linear elastic fracture mechanics insofar as possible, (see Reference 6) and are conservative in providing protection against brittle fractures. The stress limits are maintained by prescribing operating procedures which rely upon administrative pressure and temperature control during heatup and cooldown as described in Reference 7.

After hydrotesting Unit No. 1 reactor vessel, an examination of the pressure boundary welds were made using ultrasonic testing as described in Section XI of the ASME B&PV Code. As a result of this examination, the circumferential Weld WR-16, between the MK-6 transition forging and the MK-5 lower head, twenty three defects were indicated. Of these, twelve were evaluated as being nonfusion and were removed. A detailed description of the inspection, indications, and corrective actions taken are given in Appendix 5C. The final test results show that the repaired weld meets the requirements of ASME B&PV Code Section III and all other applicable documents.

The report of Appendix 5C includes a historical review of the reactor vessel fabrication sequence, the nondestructive test inspections specified and performed, the acceptance standards adhered to, and the controlled welding variables. The transition ring, lower head, instrument tubes, and weld WR-16 were stress relieved for a total of 38 hours and 16 minutes. The original coupons were heat treated for 50 hours. Therefore, the additional stress relief time should not affect the metal properties. The shell courses in the core region were not subjected to any additional stress relief time since the thermal insulation barrier was located below the core support pads. Note that the only stainless steel part exposed to the additional stress relief temperatures for 12 hours and 17 minutes was the stainless steel cladding.

The code acceptable indications of slag did not change before or after the localized stress relief or after the final hydrotest. Based on ASME Code Section XI, these indications are smaller than those which have to be reported and accelerated inservice inspections are not required.

The actual shift in NDTT will be established periodically during plant operation by testing of vessel material samples, which are irradiated cumulatively by securing them near the inside wall of the vessel in the core area. To compensate for any increase in the NDTT caused by irradiation, the limits given in the plant operating manual on the pressure-temperature relationship are periodically changed to stay within the stress limits, which are stated above, during heatup and cooldown.

The vessel closure contains 54 seven-inch studs. The stud material is ASTM A-540 as modified by code case 1335-2 which has a minimum yield strength of 104,400 psi at design temperature. The membrane stress in the studs when they are at the steady state operational condition is less than half this value. This means that about half of the 54 studs have the capability of withstanding the hydrostatic end load on vessel head without the membrane stress exceeding yield strength of the stud material at design temperature.

The emphasis on conservative operation in setting up the pressure-temperature relationship is placed on heatup and cooldown because the normal operating temperature always exceeds even the highest anticipated DTT during the life of the plant. The emphasis on conservatism is required because long term irradiation of the vessel raises the DTT and thereby limits the heatup or cooldown rates. The conservatism in the limits stated above are:

1. Use of a stress concentration factor of four on assumed flaws in calculating the stresses.

2. Use of nominal yield of material instead of actual yield.

3. Neglecting the increase in yield strength resulting from radiation effects.

The factor of four is not an actual stress concentration factor such as described in Article 4 of Section III but is a margin of conservatism based on the Fracture Analysis Diagram in ASTM E-208 as well as the stress limits maintained by the prescribed operating procedures which rely upon administrative pressure and temperature control during heatup and cooldown (see Reference 7). At the DTT, the stresses are 20% of the yield strength versus a prescribed upper limit of 80% of the yield strength; therefore, at this point there is a margin of four (80%/20%).

Since the Fracture Analysis Diagram is based on a plot of nominal stress versus temperature, and different size flaws (cracks) are assumed, the use of actual stress concentration factors does not apply.

As part of the Plant Operator Training Program, supervisory and operating personnel are instructed in reactor vessel design, fabrication and testing, as well as present and future precautions necessary for pressure testing and operating modes. The need for record keeping is stressed. Such records are helpful for future summation of time at power level and temperature which tend to influence the irradiated properties of the material in the core region. These items are incorporated in the operating instructions. Analysis of system incidents are discussed in Chapter 15.

5.3.3.3 <u>Operating Conditions</u>

The RCS and its components are designed to accommodate 10% of full power step changes in plant load and 5% of full power per minute ramp changes over the range from 15% of full power up to and including, but not exceeding, 100% of full power without reactor trip. The RCS can accept a complete loss of load from full power with reactor trip.

In addition, the steam dump system coupled with rod insertion makes it possible to accept a 50% load rejection from full power without reactor trip.

All components in the RCS are designed to withstand the effects of cyclic loads due to reactor system temperature and pressure changes. These cyclic loads are introduced by normal power changes, reactor trip, and startup and shutdown operations. The number of thermal and loading cycles used for design purposes and their bases are given in Table 5.2-2. During unit startup and shutdown, the rates of temperature and pressure changes are limited as indicated in Section 5.3.3.2.1.

To provide the necessary high degree of integrity for the equipment in the RCS, the transient conditions selected for equipment fatigue evaluation are based on a conservative estimate of the magnitude and frequency of the temperature and pressure transients resulting from normal operation, normal and abnormal load transients, and accident conditions. To a large extent, the specific transient operating conditions to be considered for equipment fatigue analyses are based upon engineering judgment and experience. Representative transients are chosen which prudently should be considered to occur during plant operation and which are sufficiently severe or frequent to be of possible significance to component cyclic behavior.

It is difficult to discuss, in absolute terms, the transients that the plant will actually experience during the 40-year operating life. However, each transient condition is discussed below in order to make clear the nature and basis for the various transients.

5.3.3.3.1 Heatup and Cooldown

The normal heatup or cooldown cases are conservatively represented by a continuous operation performed at a uniform temperature rate of 100°F per hour (the design rate).

For these cases, the heatup occurs from ambient to the no-load temperature and pressure condition and the cooldown represents the reverse situation. In actual practice, the rate of temperature change of 100°F per hour will not be attained because of other limitations such as:

1. Material NDTT considerations which establish maximum permissible temperature rates of change, as a function of plant pressure and temperature, below the design rate of 100°F per hour.

2. Slower heatup rates when using pumping energy only.

3. Interruptions in the heatup and cooldown cycles due to such factors as drawing a pressurizer steam bubble, rod withdrawal, sampling, water chemistry and gas adjustments.

The number of such complete heatup and cooldown operations is specified at 200 times each. This corresponds to five such occurrences per year for the 40-year plant design life. For the ideal plant, only one heatup and one cooldown would occur per 100% full power year, i.e., the period between refueling.

In practice, experience to date indicates that during the first year or so of operation additional unscheduled plant cooldowns may be necessary for plant maintenance; the frequency of maintenance shutdowns decreases as the plant matures.

5.3.3.3.2 Unit Loading and Unloading

The unit loading and unloading cases are conservatively represented by a continuous and uniform ramp power change of 5% per minute between 15% load and full load. This load swing is the maximum possible consistent with operation with automatic reactor control. The reactor coolant temperature will vary with load as prescribed by the temperature control system. The number of each operation is specified at 18,300 times, or 1 time per day, with approximately 40% margin for plants with 40-year design life. Figure 5.3-9 represents what may be regarded as a typical load follow program for a power plant.

5.3.3.3.3 Step Increase and Decrease of 10%

The ± 10% step change in load demand is a control transient assumed to be a change in turbine control valve opening caused by disturbances in the electrical network into which the plant output is tied. The Reactor Control System is designed to restore plant equilibrium without reactor trip following a ± 10% step change in turbine load demand initiated from nuclear plant equilibrium conditions in the range between 15% and 100% full load, the power range for automatic reactor control. In effect, during load change conditions, the Reactor Control System attempts to match turbine and reactor outputs in such a manner that peak reactor coolant temperature is minimized and reactor coolant temperature is restored to its programmed setpoint at a sufficiently slow rate to prevent excessive pressurizer pressure decrease.

Following a step load decrease in turbine load, the secondary side steam pressure and temperature initially increase since the decrease in nuclear power lags behind the step decrease in turbine load. During the same increment of time, the RCS average temperature and pressurizer pressure also initially increase. Because of the power mismatch between the turbine and reactor and the increase in reactor coolant temperature, the control system automatically inserts the control rods to reduce core power. With load decrease, the reactor coolant temperature will ultimately be reduced from its peak value to a value below its initial equilibrium value at the inception of the transient. The reactor coolant average temperature setpoint change is made as a function of turbine-generator load which is determined by first stage turbine pressure measurement. The pressurizer pressure will also decrease from its peak pressure value and follow the reactor coolant decreasing temperature trend. At some point during the decreasing pressure transient, the saturated water in the pressurizer begins to flash thereby reducing the rate of pressure decrease. Subsequently, the pressurizer heaters come on to restore the plant pressure to its normal value.

Following a step load increase in turbine load, the reverse situation occurs, i.e., the secondary side steam pressure and temperature initially decrease and the reactor coolant average temperature and pressure initially decrease. The control system automatically withdraws the control rods to increase core power. The decreasing pressure transient is reversed by actuation of the pressurizer heaters and eventually the system pressure is restored to its normal value. The reactor coolant average temperature will be raised to a value above its initial equilibrium value at the beginning of the transient.

The number of each operation is specified at 2000 times, or 50 per year, for the 40-year plant design life.

5.3.3.3.4 <u>50% Step Decrease In Load</u>

This transient applies to a 50% step decrease in turbine load. The resultant rapid increase in reactor coolant average temperature and secondary side steam pressure and temperature will automatically initiate a secondary side steam dump that will prevent a reactor shutdown or lifting of steam generator safety valves. If a steam dump system was not provided to cope with this transient, there would be such a strong mismatch between what the turbine is asking for and what the reactor is furnishing that a reactor trip would occur and the steam generator safety valves would lift.

The number of occurrences of this transient is specified at 200 times, or 5 per year, for the 40-year plant design life. Reference to the Yankee-Rowe record indicates that this basis is adequately conservative.

5.3.3.3.5 <u>Loss of Load</u>

This transient applies to a step decrease in turbine load from full power to no load without immediately initiating a reactor trip. This represents the most severe transient on the RCS. In this case, the reactor and turbine eventually trip as a consequence of a low-low steam generator level trip initiated by the Reactor Protection System (RPS). Since redundant means of tripping the reactor upon turbine trip are provided as part of the RPS (high pressurizer level), this situation is not credible and is of value only from the standpoint of fatigue analysis.

The number of occurrences of this transient is specified at 80 times, or 2 per year, for the 40-year plant design life.

5.3.3.3.6 <u>Loss of Power</u>

This transient applies to a blackout situation involving the loss of outside electrical power to the station resulting in a reactor and turbine trip on low reactor coolant flow. Under these circumstances, the reactor coolant pumps (RCPs) are de-energized and, following the coastdown of the RCPs, natural circulation builds up in the system to some equilibrium value. This condition permits removal of core residual heat through the steam generators which at this time are receiving feedwater from the Auxiliary Feedwater (AFW) System operating from diesel generator power. Steam is removed for reactor cooldown through atmospheric relief valves provided for this purpose.

The number of occurrences of this transient is specified at 40 times, or 1 per year, for the 40-year plant design life.

5.3.3.3.7 Loss of Flow

This transient applies to a partial loss-of-flow accident from full power in which a RCP is tripped out of service as a result of a loss of power to that pump. The consequences of such an accident above approximately 60% power are a reactor and turbine trip on low reactor coolant flow, followed by automatic opening of the steam-relief system and flow reversal in the affected loop. The flow reversal results in reactor coolant, at cold-leg temperature, being passed through the steam generator and cooled still further. This cooler water then passes through the hot leg piping and enters the reactor vessel outlet nozzles. The net result of the flow reversal is a sizeable reduction in the hot leg coolant temperature of the affected loop. Between 10% and 60% power, loss of two loops of flow results in the same consequences as for the loss of flow in one loop above 60% power.

The number of occurrences of this transient is specified at 80 times, or 2 per year, for the 40-year plant design life.

5.3.3.3.8 Reactor Trip From Full Power

A reactor trip from full power may occur for a variety of reasons resulting in temperature and pressure transients in the RCS and in the secondary side of the steam generator. This is the result of continued heat transfer from the reactor coolant in the steam generator. The transient continues until the reactor coolant and steam generator secondary side temperatures are in equilibrium at zero power conditions. A continued supply of feedwater and controlled relief of secondary steam remove the core residual heat and prevent the steam generator safety valves from lifting. The reactor coolant temperature and pressure undergo a rapid decrease from full power values as the RPS causes the control rods to move into the core.

The number of occurrences of this transient is specified at 400 times, or 10 per year, for the 40-year plant design life.

5.3.3.3.9 Turbine Roll Test

This transient is imposed on the plant during the hot functional test period for turbine cycle checkout. RCP power will be used to heat the reactor coolant to operating temperature and the steam generated will be used to perform a turbine roll test. However, the plant cooldown during this test will exceed the 100°F per hour maximum rate specified in Section 5.3.3.3.1 above.

The number of such test cycles is specified at 10 times and are to be performed at the beginning of plant operating life prior to irradiation of the reactor vessel.

5.3.3.3.10 Hydrostatic Test Conditions

The pressure tests covered by this section include both shop and field hydrostatic tests which occur as a result of component or system testing.

The pressure tests are outlined below:

1. Primary Side Hydrostatic Test at 3107 psig Before Initial Startup

   This hydro test is performed at a water temperature which is compatible with reactor vessel material DTT requirements and at a maximum test pressure of 3107 psig. In this test, the primary side of the steam generator will be pressurized to 3107 psig, coincident with the secondary side pressure of 0 psig. The RCS is designed for five cycles of this hydro test.

2. Secondary Side Hydrostatic Test Before Initial Startup

   The secondary side of the steam generator is pressurized to 1356 psig with a minimum water temperature of 70°F, coincident with the primary side at 0 psig.

   The steam generator may experience five cycles of this test.

5.3.3.3.11 Primary Side Leak Test

Each time the primary system is opened, a leak test will be performed. During this test the primary system pressure is raised to 2500 psia for design purposes and the system temperature is brought above the DTT, while the system is checked for leaks.

In actual practice, the primary system will be pressurized to below 2500 psia to prevent the pressurizer safety valves from lifting during the leak test.

During this leak test, the secondary side of the steam generator must be pressurized so that the pressure differential across the tube sheet does not exceed 1600 psi. This is accomplished by closing off the steam lines.

For design purposes, it is assumed that the primary side will experience 50 cycles of this test during the 40-year design life of the plant.

5.3.3.3.12 Pressurizer Surge and Spray Line Connections

The surge and spray nozzle connections at the pressurizer vessel are subject to cyclic temperature changes resulting from the transient conditions described previously. The various transients are characterized by variations in reactor coolant temperature which result in water surges into or out of the pressurizer. The surges manifest themselves as changes in system pressure which, depending upon whether an increase or decrease in pressure occurs, result in introducing spray water into the pressurizer to reduce pressure or actuating the pressurizer heaters to increase pressure to the equilibrium value. To illustrate a load change cycle as it affects the pressurizer, consider a design step increase in load. The pressurizer initially experiences an outsurge with a drop in system pressure which actuates the pressurizer heaters to restore system pressure. As the Reactor Control System reacts, the reactor coolant temperature is increased which causes an insurge into the pressurizer and raises system pressure. As pressure is increased, the heaters go off, and at some pressure setpoint, the spray valves open to limit the pressure rise and restore system pressure.

Thus, the pressurizer surge nozzle is subjected to a temperature increase on the outsurge, followed by a temperature decrease on the insurge during this load transient. The pressurizer spray nozzle is subjected to a temperature decrease when the spray valve opens to admit reactor coolant cold leg water into the pressurizer. The pressurizer experiences a reverse situation during a load decrease transient, i.e., an insurge followed by an outsurge. It is assumed that the spray valve opens to admit spray water into the pressurizer once at the design flowrate for each design step change in plant load. Design thermal and loading cycles for the spray nozzle for different transients are given in Table 5.2-2.

During plant cooldown, spray water is introduced into the pressurizer to cool down the pressurizer and to remove gas from the reactor coolant. The maximum pressurizer cooldown rate is specified at 200°F per hour which is twice the rate specified for the other RCS components.

5.3.3.3.13 Accident Conditions

The effect of the accident loading is evaluated in combination with normal loads to demonstrate the adequacy to meet the stated plant safety criteria.

The design criteria used to examine the effects on system restraints of pipe rupture have considered both longitudinal and circumferential pipe breaks at any location within the reactor coolant boundary, as well as the main steam and feedwater systems. The longitudinal thrust system was defined as having the same effects as the circumferential rupture. The maximum unsupported lengths in a pipe run have been calculated to determine the spacing between restraints required to prevent pipe whipping regardless of the break location.

A brief description of each accident transient to be considered follows. In each case, one occurrence is evaluated.

1. Reactor Coolant Pipe Break

This accident involves the rupture of an RCS pipe resulting in a loss of primary coolant. In confirmatory dynamic analyses of the RCS, three circumferential ruptures were postulated. One rupture was assumed in each leg of the loop. It is conservatively assumed that the system pressure and temperature are reduced rapidly and the Emergency Core Cooling System (ECCS) is initiated to introduce 70°F water into the RCS. The safety injection actuation signal will also result in a turbine and reactor trip. Because of the rapid blowdown of coolant from the system and the comparatively large heat capacity of the metal sections of the components, it is likely that the metal is still at no-load temperature conditions when the 70°F safety injection water is introduced into the system.

2. Steamline Break

For component evaluation, the following conservative conditions were considered:

a. The reactor is initially in a hot shutdown condition assuming all rods in, except the most reactive rod, which is assumed to be stuck in its fully withdrawn position.

b. A steamline break occurs inside the containment upstream of the flow limiter, resulting in a reactor and turbine trip.

c. Subsequent to the break, it is assumed for fatigue analysis purposes that there is no return to power and the reactor coolant temperature cools down to 212°F.

d. The ECCS pumps restore the reactor coolant pressure to 2500 psia.

The above conditions result in the most severe temperature and pressure variations which the component will encounter during a steam-break accident.

3. Steam Generator Tube Rupture

This accident postulates the double-ended rupture of a steam generator tube resulting in a decrease in pressurizer level and reactor coolant pressure. Reactor trip will occur due to a safety injection signal on low pressurizer pressure. When the accident occurs, some of the reactor coolant blows down into the affected steam generator causing the level to rise. If the level rises sufficiently, a high-level alarm will occur, and the feedwater regulating valve will close. Approximately 30 minutes after the rupture, the primary system pressure is reduced to below the secondary safety valve settings

(~1100 psia). At this time, the planned procedure for recovery from this accident calls for isolation of the steam line leading from the affected steam generator. Therefore, this accident will result in a transient which is no more severe than that associated with a reactor trip. For this reason, it requires no special treatment insofar as fatigue evaluation is concerned.

5.3.3.3.14 Service Life

The service life of the RCS pressure-containing components depends upon the end-of-life material radiation damage, unit operational thermal cycles, design and manufacturing quality standards, environmental protection, maintenance standards, and adherence to established operating procedures.

5.3.4 References, Section 5.3

1. Pellini, W.S. and Loss, F.J., "Integration of Metallurgical and Fracture Mechanics Concepts of Transition Temperature Factors Relating to Fracture-Safe Design for Structural Steels," Welding Research Council Bulletin No. 141 (1969).

2. Babcock & Wilcox Nuclear Power Division, "Master Integrated Reactor Vessel Surveillance Program," BAW-1543, Rev. 3, September 1989.

3. Fatigue Properties of Irradiated Pressure Vessels by Gibbon et. al., ASTM STP 426, pages 408 to 437.

4. Pellini, W.S. and Puzak, P.P., "Fracture Analysis Diagram Procedures for the Fracture Safe Engineering Design of Steel Structures." NRL Report 5920, Mar. 15, 1963, Welding Research Council Bulletin No 88 (1963).

5. Pellini, W.S. and Puzak, P.P., "Practical Considerations in Applying Laboratory Fracture Test Criteria to the Fracture - Safe Design of Pressure Vessels," NRL Report 6630, Nov. 5, 1963.

6. Mager, T.R. and Hazelton, W.S.; Evaluation of Linear Elastic Fracture Mechanics of Radiation Damage to Pressure Vessel Steels, Vienna I AEA-Meeting, June 2-9, 1969.

7. Porse, L. Reactor Vessel Design Considering Radiation Effects, ASME 6 WA 100.

TABLE 5.3-1 (1 of 2)

REACTOR VESSEL DESIGN DATA

Design/Operating Pressure, psig	2485/2235
Hydrostatic Test Pressure, psig	3107
Design Temperature, °F	650
Overall Height of Vessel and Closure Head, ft-in. (Bottom Head O.D. to top of Control Rod Mechanism Adapter)	43-9 23/32
Thickness of Insulation, min., in.	3
Number of Reactor Closure Head Studs	54
Diameter of Reactor Closure Head Studs, in.	7
ID of Flange, in.	172-9/16
OD of Flange, in.	205
ID at Shell, in.	173
Inlet Nozzle ID, in.	27-1/2
Outlet Nozzle ID, in.	29
Clad Thickness, min., in.	5/32"
Lower Head Thickness, min., in. (base metal)	5-3/8"
Vessel Belt-Line Thickness, min., in. (base metal)	8.441
Closure Head Thickness, in.	6-1/2"
Reactor Coolant Inlet Temperature, °F	530.2
Reactor Coolant Outlet Temperature, °F	594.2
Reactor Coolant Flow, lb/hr	135.0×10^6
Total Water Volume Below Core, ft^3	1050
Water Volume in Active Core Region, ft^3	665

TABLE 5.3-1 (2 of 2)

REACTOR VESSEL DESIGN DATA

Total Water Volume to Top of Core, ft^3	2164
Total Reactor Vessel Water Volume to Coolant Piping Nozzles Centerline, ft^3	2959
Total Reactor Vessel Water Volume, (with core and internals in place), ft^3	4945
Total Reactor Coolant System Volume, ft^3 (ambient)	12,710

TABLE 5.3-2

REACTOR COOLANT SYSTEM DESIGN PRESSURE DROP

	Pressure Drop, psi (estimated)
Across Pump Discharge Leg (including valve)	3.8
Across Vessel, Including Nozzles	52.0
Across Hot Leg (including valve)	3.0
Across Steam Generator	30.1
Across Pump Suction Leg	2.6
Total Pressure Drop	91.5

TABLE 5.3-3

SUMMARY OF ESTIMATED PRIMARY PLUS SECONDARY STRESS INTENSITY FOR COMPONENTS OF THE REACTOR VESSEL

Area	Stress Intensity (psi)	Allowable Stress (psi) (at Operating Temperature)
Control Rod Housing	37,100	69,900
Head Flange	50,500	80,000
Vessel Flange	45,400	80,000
Primary Nozzles	55,000	80,000
Stud Bolts	95,000	110,200
Vessel Support	55,000	80,000
Core Support Pad	19,000	69,900
Bottom Head to Shell	28,600	80,000
Bottom Instrumentation	59,340	69,900

TABLE 5.3-4

SUMMARY OF ESTIMATED CUMULATIVE FATIGUE USAGE FACTORS FOR COMPONENTS OF THE REACTOR VESSEL

Item	Usage Factor*[a]
Control Rod Housing	54
Head Flange	0.015
Vessel Flange	0.015
Stud Bolts	0.032
Primary Nozzles	0.042
Vessel Support	0.042
Core Support Pad (lateral)	0.0
Bot. Head to Shell	0.22
Bot. Instrumentation	0.03

* covers all transients

[a] as defined in Section III of the ASME Boiler and Pressure Vessel Code, Nuclear Vessels.

TABLE 5.3-5

REACTOR VESSEL SURVEILLANCE PROGRAM SPECIMENS

Material	FOUR CAPSULES CONTAIN[1]: No. of Charpys	No. of Tensiles	No. of WOLs	TWO CAPSULES CONTAIN[1]: No. of Charpys	No. of Tensiles	No. of WOLs	TWO CAPSULES CONTAIN[1]: No. of Charpys	No. of Tensiles	No. of WOLs
High NDT Plate (RW Direction)	10	--	--	10	2	4	10	2	--
High NDT Plate (WR Direction)	10	2	4	10	--	--	10	--	--
Weld Metal	8	2	--	8	2	--	8	2	4
Heat-Affected Zone Metal	8	--	--	8	--	--	8	--	--
ASTM Reference	8	--	--	8	--	--	8	--	--

(1) The following dosimeters and thermal monitors are included in each capsule:

Dosimeters

Iron
Nickel
Copper
Cobalt-Aluminum (0.15% Co)
Cobalt-Aluminum (Cadmium shielded)
U238 (Cadmium shielded)
Np 237 (Cadmium shielded)

Thermal Monitors

97.5% Pb, 2.5% Ag (≈ 579°F Melting Point)
97.5% Pb, 1.75% Ag, 0.75% Sn (≈ 590°F Melting Point)

TABLE 5.3-6

REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM
SPECIMEN CAPSULE WITHDRAWAL SCHEDULE

UNIT 1

CAPSULE DESIGNATION	CAPSULE REMOVAL TIME (EFPY)
T	REMOVED (1.16)
U	REMOVED (3.52)
X	REMOVED (5.17)
Y	8.5
Z	32
W,S,V	STANDBY

UNIT 2

CAPSULE DESIGNATION	CAPSULE REMOVAL TIME (EFPY)
U	REMOVED (1.27)
T	REMOVED (3.56)
Y	8.5
X	13
W,S,V,Z	STANDBY

TABLE 5.3-7

UNIT 1 REACTOR VESSEL TOUGHNESS DATA

COMPONENT	HEAT NO.	MATERIAL TYPE	Cu (%)	NI (%)	P (%)	T_{NDT} (°F)	50 FT-LB/35 MIL TEMP[a] (°F)	RT_{NDT} (°F)	TRANS[a] USE (FT-LB)
Closure Head Dome	B9094-2	A5338, CL.1	.14	.55	.012	20	90	30	77
Closure Head Segment	C5086-1	" "	.09	.54	.014	10	32	10	103
Closure Head Segment	B8793-3	" "	.09	.52	.012	10	53	10	96
Closure Head Flange	123W323	A508, CL.2	-	.69	.010	55[a]	26	55	96
Vessel Flange	123V236	" "	.06	.68	.004	7[a]	-2	7	131
Inlet Nozzle	ZT3600-1	" "	.12	.68	.009	60[a]	27	60	79
Inlet Nozzle	ZT3600-2	" "	.11	.67	.009	60[a]	41	60	82
Inlet Nozzle	ZT3592-1	" "	.10	.66	.011	60[a]	103	60	77
Inlet Nozzle	ZT3592-2	" "	.11	.67	.010	60[a]	51	60	62
Outlet Nozzle	ZT3592-3	" "	.11	.68	.010	60[a]	60	60	86
Outlet Nozzle	ZT3592-4	" "	.11	.68	.009	46[a]	16	46	85
Outlet Nozzle	ZT3600-3	" "	.10	.67	.011	60[a]	52	60	82
Outlet Nozzle	ZT3600-4	" "	.11	.68	.011	60[a]	46	60	>63
Upper Nozzle Shell	123V426	" "	.06	.75	.005	10	43	10	115
Lower Nozzle Shell	ZV3300	" "	.06	.83	.008	20	72	20	87
Inter. Shell	C3795-2	A5338, CL.1	.12	.49	.010	-10	70	10	85
Inter. Shell	B7835-1	" "	.12	.49	.010	-20	65 (Actual)	5	115 (Actual)
Lower Shell	B7823-1	" "	.13	.48	.013	-20	56 (Actual)	-4	115.5 (Actual)
Lower Shell	C3799-2	" "	.15	.50	.010	-20	80 (Actual)	20	116 (Actual)
Bottom Head Trans. Ring	ZV3779	A508, CL.2	.09	.71	.010	10	60	10	92
Bottom Head Dome	B7777-1	A5338, CL.1	-	.62	.015	-30	33	-27	84
Inter. to Lower Shell Girth Weld Seam	WF70[b]	SAW	.32	.56	.017	0[a]	--	0	--
Inter. Shell Long. Weld Seam	WF4[c]	SAW	.29	.55	.013	0[a]	--	0	--
Inter. Shell Long. Weld Seam	WF8[d]	SAW	.29	.55	.013	0[a]	--	0	--
Lower Shell Long. Weld Seam	WF8[d]	SAW	.29	.55	.013	0[a]	--	0	--
Nozzle to Inter. Shell	WF154[e]	SAW	.31	.59	.013	0[a]	--	0	--
Girth Weld Seam	SA1769[f]	SAW	.26	.60	.019	0[a]	--	0	--

(a) Estimated using Methods of U.S.NRC NUREG-0800, Branch Technical Position MTEB 5-2, July, 1981
(b) Weld Wire Heat No. 72105 and Linde 80 Flux Lot No. 8669
(c) Weld Wire Heat No. 8T1762 and Linde 80 Flux Lot No. 8597
(d) Weld Wire Heat No. 8T1762 and Linde 80 Flux Lot No. 8632
(e) Weld Wire Heat No. 406L44 and Linde 80 Flux Lot No. 8720
(f) Weld Wire Heat No. 71249 and Linde 80 Flux Lot No. 8738

TABLE 5.3-8

UNIT 2 REACTOR VESSEL TOUGHNESS DATA

COMPONENT	HEAT NO.	MATERIAL TYPE	Cu (%)	NI (%)	P (%)	T_{NDT} (°F)	50 FT-LB/35 MIL TEMP[a] (°F)	RT_{NDT} (°F)	TRANS[a] USE (FT-LB)
Closure Head Dome	B9094-1	A5338, CL.1	.14	.55	.012	-20	71	11	72
Closure Head Segment	C4787-1A	" "	.13	.62	.008	0	30	0	88
Closure Head Segment	C5086-2	" "	.09	.54	.014	30	45	30	88
Closure Head Flange	124W609	A508, CL.2	.08	.70	.010	12[a]	-13	12	105
Vessel Flange	2V-965	" "	.12	.74	.010	60[a]	33	60	79
Inlet Nozzle	ZT4007-2	" "	.11	.70	.009	48[a]	32	48	>78
Inlet Nozzle	ZT3885-1	" "	.11	.58	.012	60[a]	43	60	82
Inlet Nozzle	ZT3885	" "	.11	.56	.011	43[a]	31	43	78
Inlet Nozzle	ZT3885	" "	.11	.56	.012	60[a]	48	60	>84
Outlet Nozzle	ZV3930	" "	.12	.66	.010	58[a]	20	58	93
Outlet Nozzle	ZV3930	" "	.11	.65	.011	48[a]	15	48	>80
Outlet Nozzle	ZV3930	" "	.12	.67	.011	55[a]	28	55	84
Outlet Nozzle	ZT3885-4	" "	.11	.57	.013	60[a]	41	60	>61
Upper Nozzle Shell	ZD3940	A508, CL.2	.07	.62	.008	10	65	10	106
Lower Nozzle Shell	ZV3855	" "	.09	.66	.008	10	70	10	>80
Lower Shell	B8029-1	A5338, CL.1	.12	.51	.010	-10	82	22	81
Lower Shell	C4007-1	" "	.12	.53	.010	10	82 (Actual)	22	94 (Actual)
Inter. Shell	B8006-1	" "	.12	.54	.010	10	68	10	89
Inter. Shell	B8040-1	" "	.14	.52	.008	-10	62	2	92
Bottom Head Trans. Ring	3V-433	A508, CL.2	.09	.76	.010	0	43	0	87
Bottom Head Dome	C4007-2	A5338, CL.1	.12	.53	.010	-20	60	0	72
Inter. to Lower Shell Girth Weld Seam	SA1769[b]	SAW	.26	.60	.019	0[a]	--	0	--
Lower Shell Long. Weld Seam	WF29[c]	SAW	.23	.63	.019	0[a]	--	0	--
Inter. Shell Long. Weld Seam	WF70[d]	SAW	.32	.56	.017	0[a]	--	0	--
Nozzle to Inter. Shell Girth Weld Seam	WF200[e]	SAW	.24	.63	.010	0[a]	--	0	--

(a) Estimated using Methods of U.S.NRC NUREG-0800, Branch Technical Position MTEB 5-2, July, 1981
(b) Weld Wire Heat No. 71249 and Linde 80 Flux Lot No. 8738
(c) Weld Wire Heat No. 72102 and Linde 80 Flux Lot No. 8650
(d) Weld Wire Heat No. 72105 and Linde 80 Flux Lot No. 8669
(e) Weld Wire Heat No. 821T44 and Linde 80 Flux Lot No. 8773

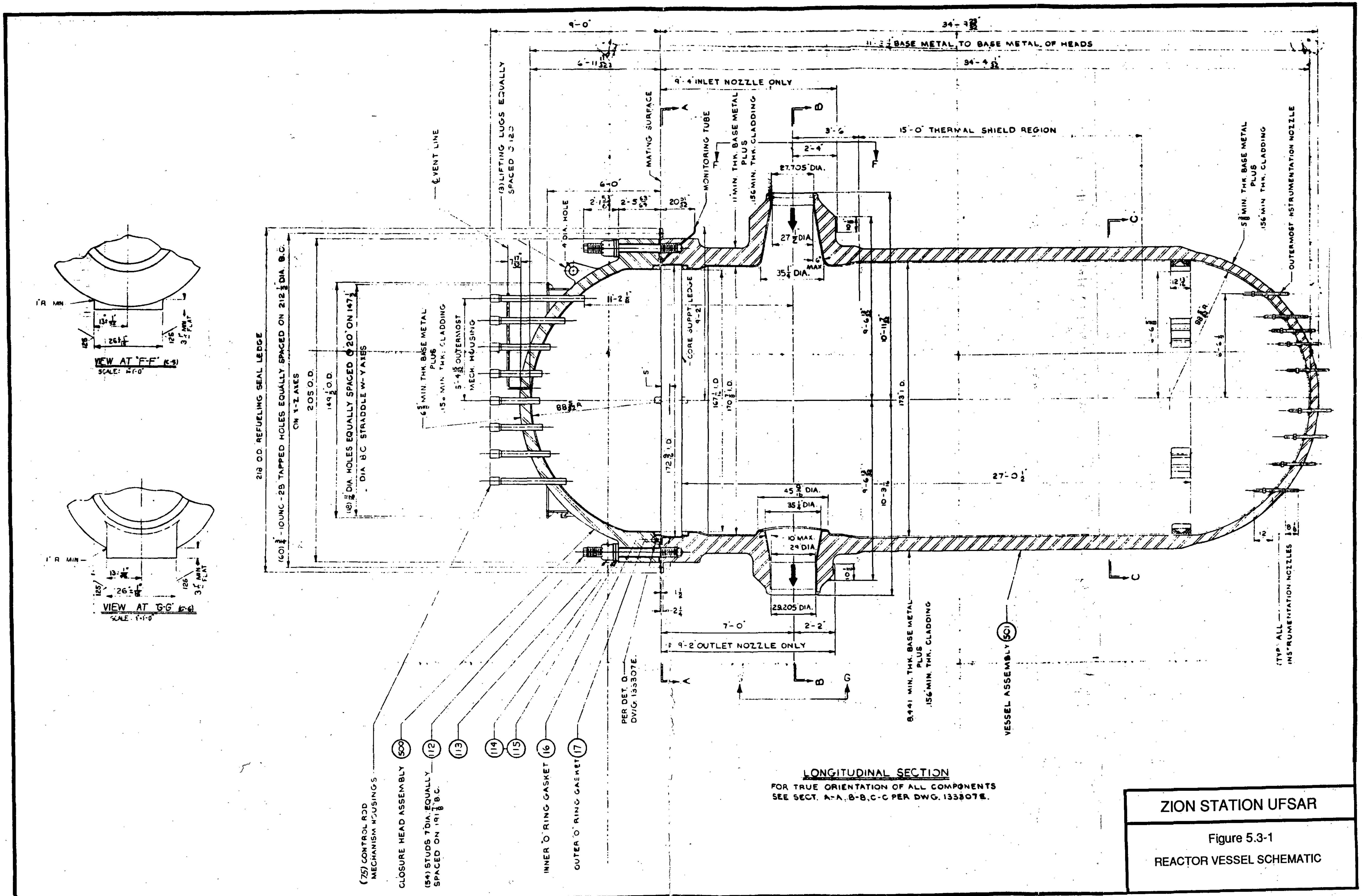

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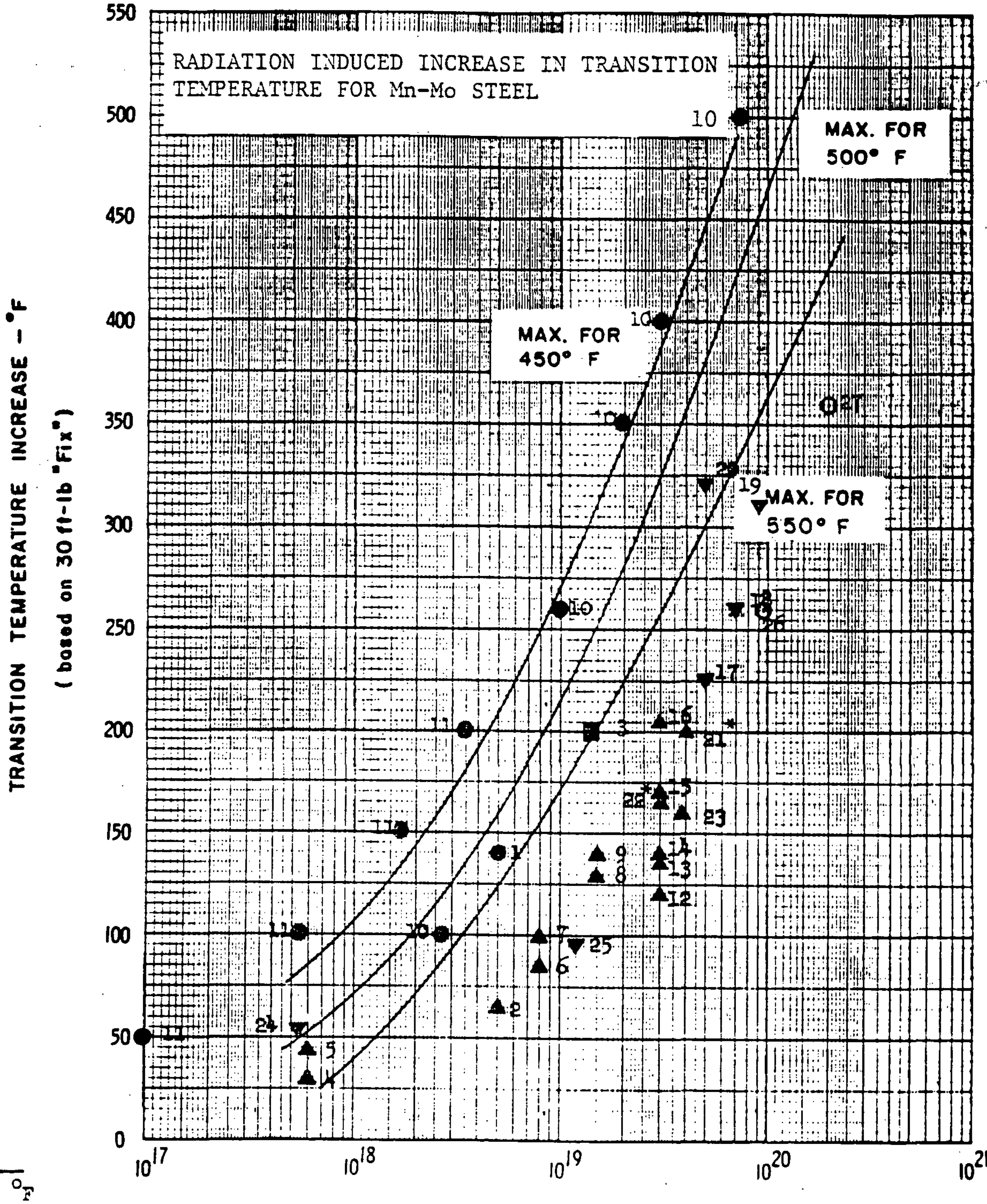
Figure 5.3-1

REACTOR VESSEL SCHEMATIC



Code

Temp. °F
● 450
■ 490
▲ 550
▼ 475 to 540
⊙ 600

Numbers 1 through 27 (see attached sheets)

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Figure 5.3-2(1 of 4)

RADIATION INDUCED INCREASE IN TRANSITION TEMPERATURE FOR Mn-Mo STEEL

REFERENCES FOR FIGURE 5.3-2 (2 of 4)

RADIATION INDUCED INCREASE IN TRANSITION TEMPERATURE FOR Mn-Mo STEEL

References	Material	Temp, °F	Neutron Exposure, n/cm^2 (> 1 Mev)	Change in NDTT, °F
1. NRL Report 6160 Page 12	SA302B	450	5×10^{18}	140
2. NRL Report 6160 Page 12	SA302B	550	5×10^{18}	65
3. NRL Report 6160 Page 13	SA302B	490	1.4×10^{19}	200
4. ASTM-STP 341 Page 226	SA302B	550	$6 \times 106,^{17}$	30**
5. ASTM-STP 341 Page 226	SA302B	550	6×10^{17}	45
6. ASTM-STP 341 Page 226	SA302B	550	8×10^{18}	85**
7. ASTM-STP 341 Page 226	SA302B	550	8×10^{18}	100
8. ASTM-STP 341 Page 226	SA302B	550	1.5×10^{19}	130**
9. ASTM-STP 341 Page 226	SA302B	550	1.5×10^{19}	140
10. NRL Report 6160 Page 6	All Steels	<450	Various	Various
11. Nuclear Science & Engineering 19:18-38 (1964)	SA302B	<450	Various	Various
12. Quarterly Report of Progress, "Irradiation Effects on Reactor Structural Materials" 11-1-64/1-31-64	SA302B	550	3×10^{19}	120

**Transverse Specimens

REFERENCES FOR FIGURE 5.3-2 (3 of 4)

References	Material	Neutron Temp, °F	Change in Exposure, n/cm^2 (> 1 Mev)	NDTT, °F
13. Quarterly Report of Progress, "Irradiation Effects on Reactor Structural Materials" 11-1-64/1-31-64	SA302B	550	3×10^{19}	135
14. Quarterly Report of Progress, "Irradiation Effects on Reactor Structural Materials" 11-1-64/1-31-64	SA302B	550	3×10^{19}	140
15. Quarterly Report of Progress, "Irradiation Effects on Reactor Structural Materials" 11-1-64/1-31-64	SA302B	550	3×10^{19}	170
16. Quarterly Report of Progress, "Irradiation Effects on Reactor Structural Materials" 11-1-64/1-31-64	SA302B	550	3×10^{19}	205
17. NRL Report 6179 Page 9	SA302B	475-540	5×10^{19}	225
18. NRL Report 6179 Page 9	SA302B	475-540	7×10^{19}	260
19. NRL Report 6179 Page 9	SA302B	475-540	9×10^{19}	310
20. NRL Report 6179 Page 9	SA302B	475-540	5×10^{19}	320
21. NRL Report 6160 Page 15	SA302B	540*	4×10^{19}	200

* Plotted as a 550°F data point

REFERENCES FOR FIGURE 5.3-2 (4 of 4)

References	Material	Temp, °F	Neutron Exposure, n/cm^2 (> 1 Mev)	Change in NDTT, °F
22. NRL Report 6160 Page 15	SA302B	540*	3×10^{19}	165
23. Private Communication with NRL	SA302B	550	3.8×10^{18}	160
24. Progress Report No. 1, "Irradiation Tests on Reactor Pressure Vessel Steels in Br-3 Reactor Facilities" August, 1965	SA302B	≃ 525	5.4×10^{18}	54
25. IBID	SA302B	≃ 525	1.2×10^{19}	96
26. Progress Report No. 1, "Irradiation Tests on Reactor Pressure Vessel Steels in Br-3 Reactor Facilities August, 1965"	SA302B	≃ 600	9.5×10^{19}	260
27. IBID	SA302B	≃ 600	2×10^{20}	360

* Plotted as a 550°F data point

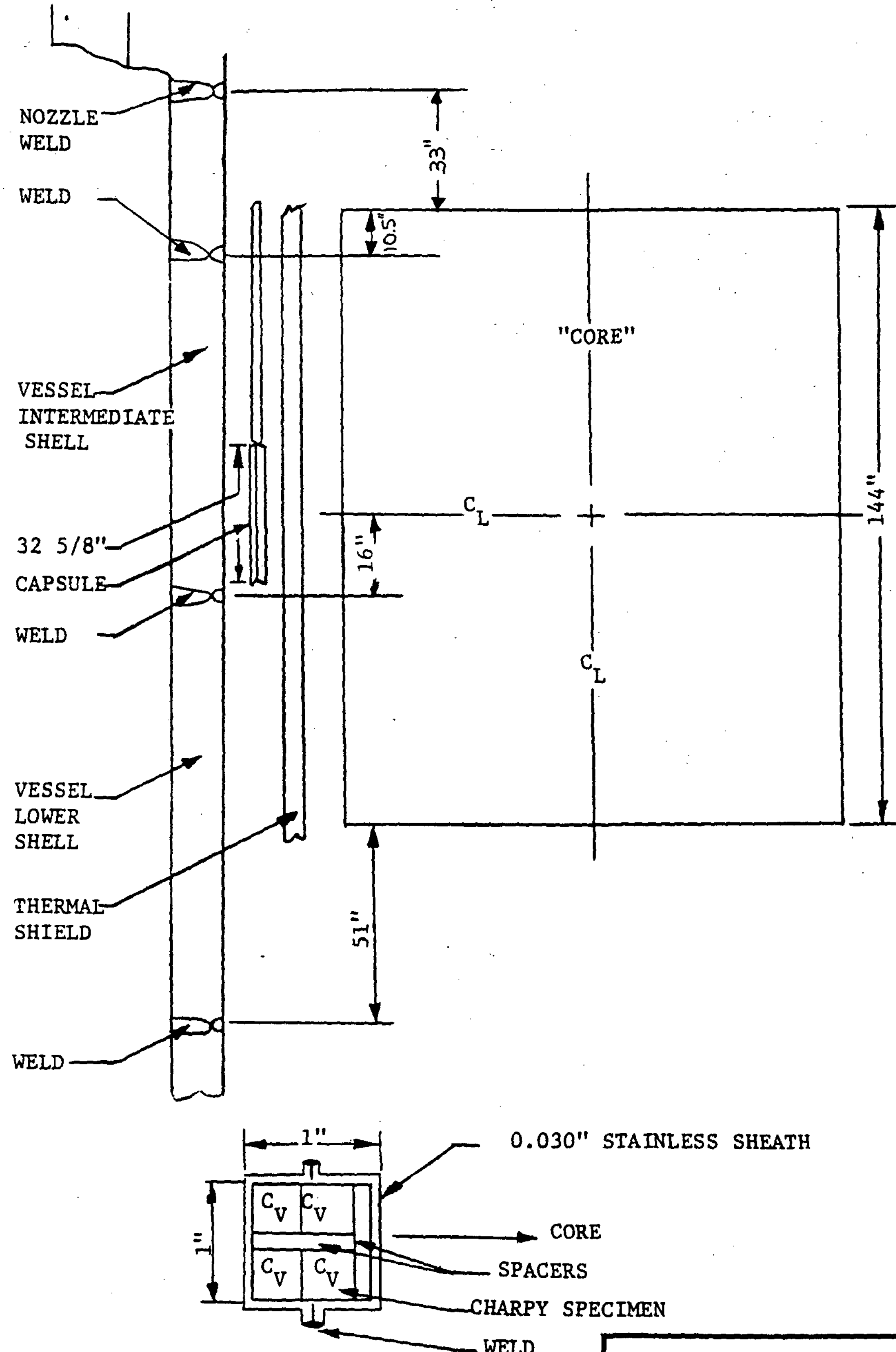

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Figure 5.3-3

SURVEILLANCE CAPSULE
ELEVATION VIEW

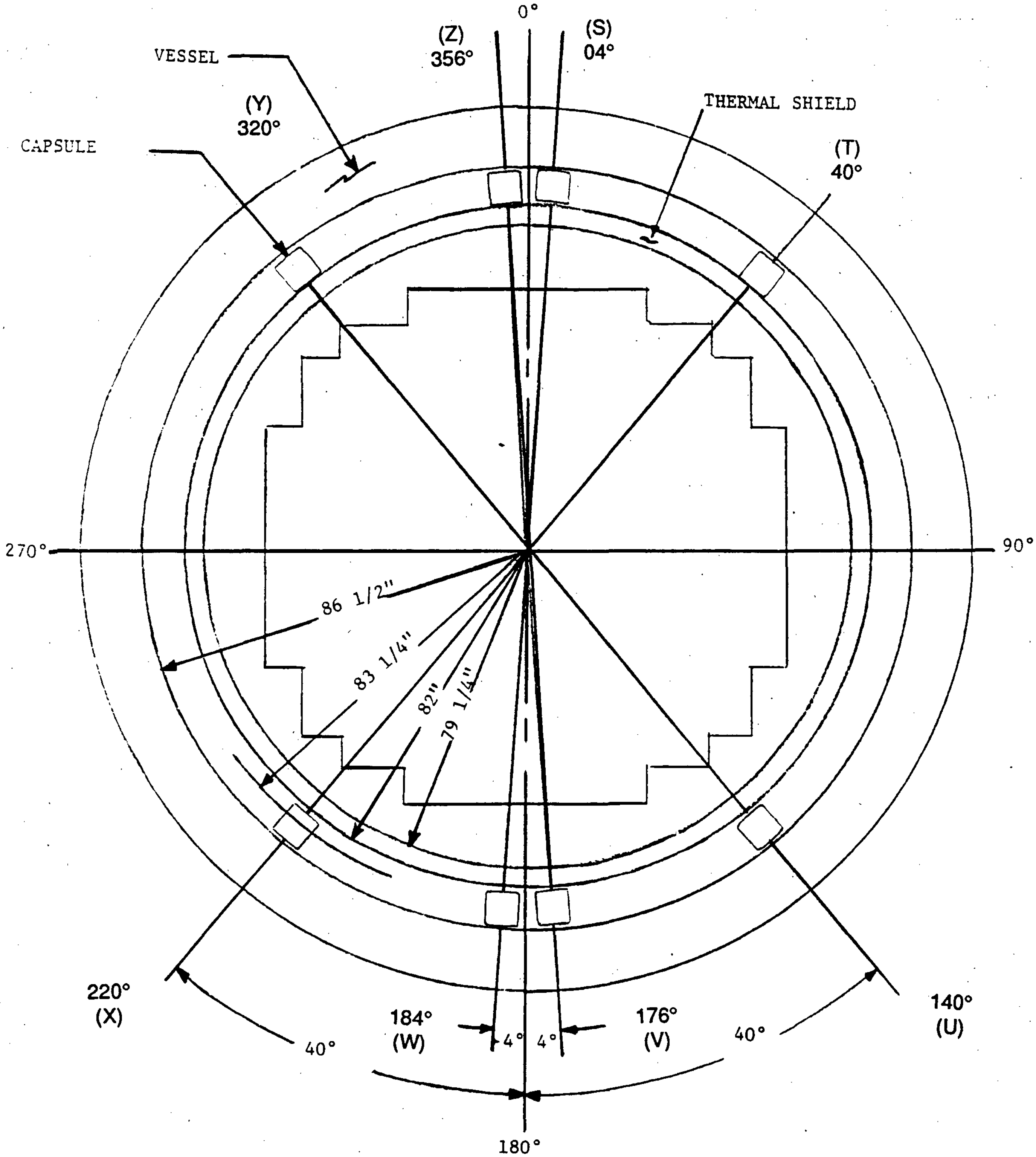

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Figure 5.3-4

SURVEILLANCE CAPSULE
PLAN VIEW

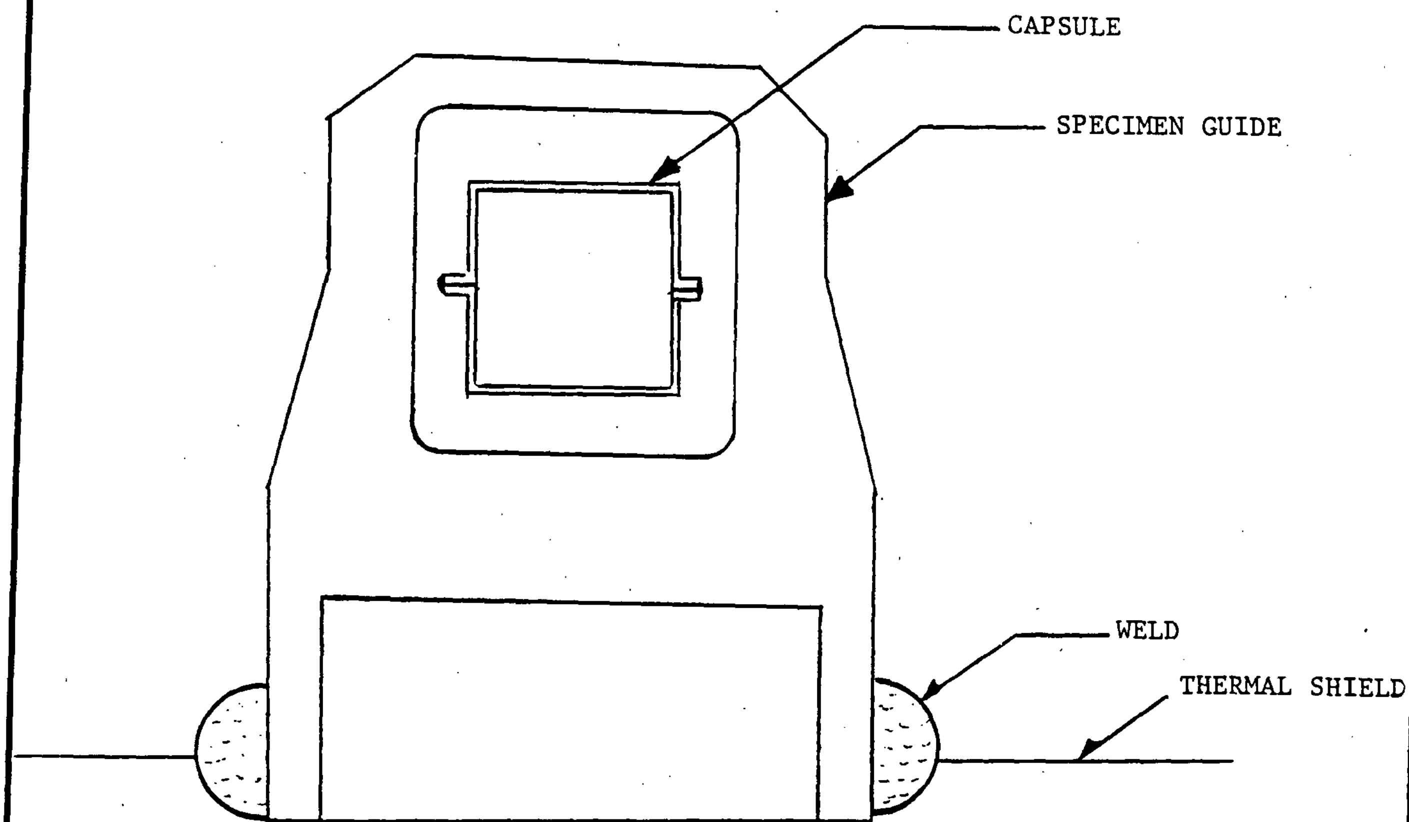

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Figure 5.3-5

SPECIMEN GUIDE TO THERMAL SHIELD ATTACHMENT

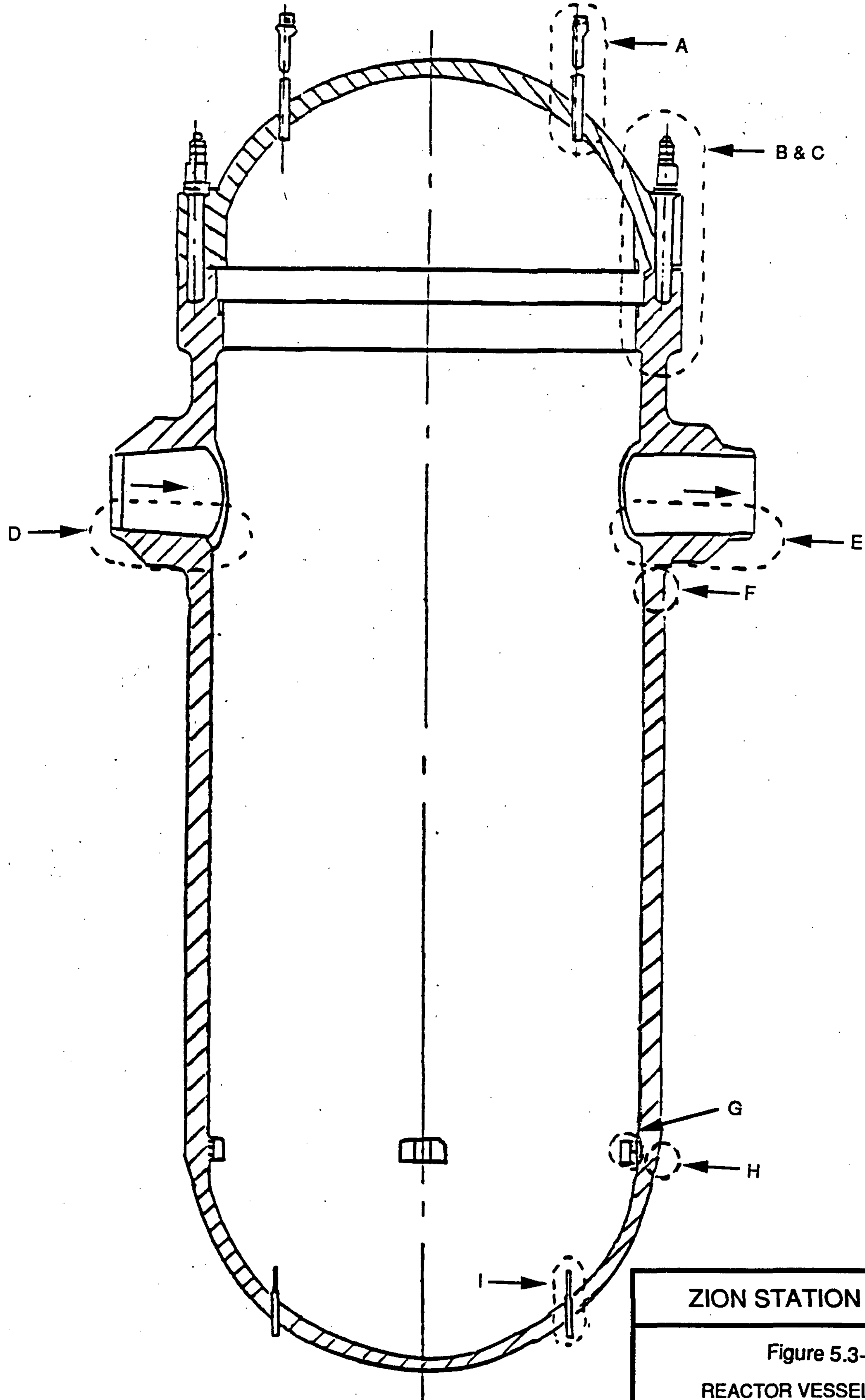

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Figure 5.3-6

REACTOR VESSEL STRESS ANALYSIS: AREAS EXAMINED

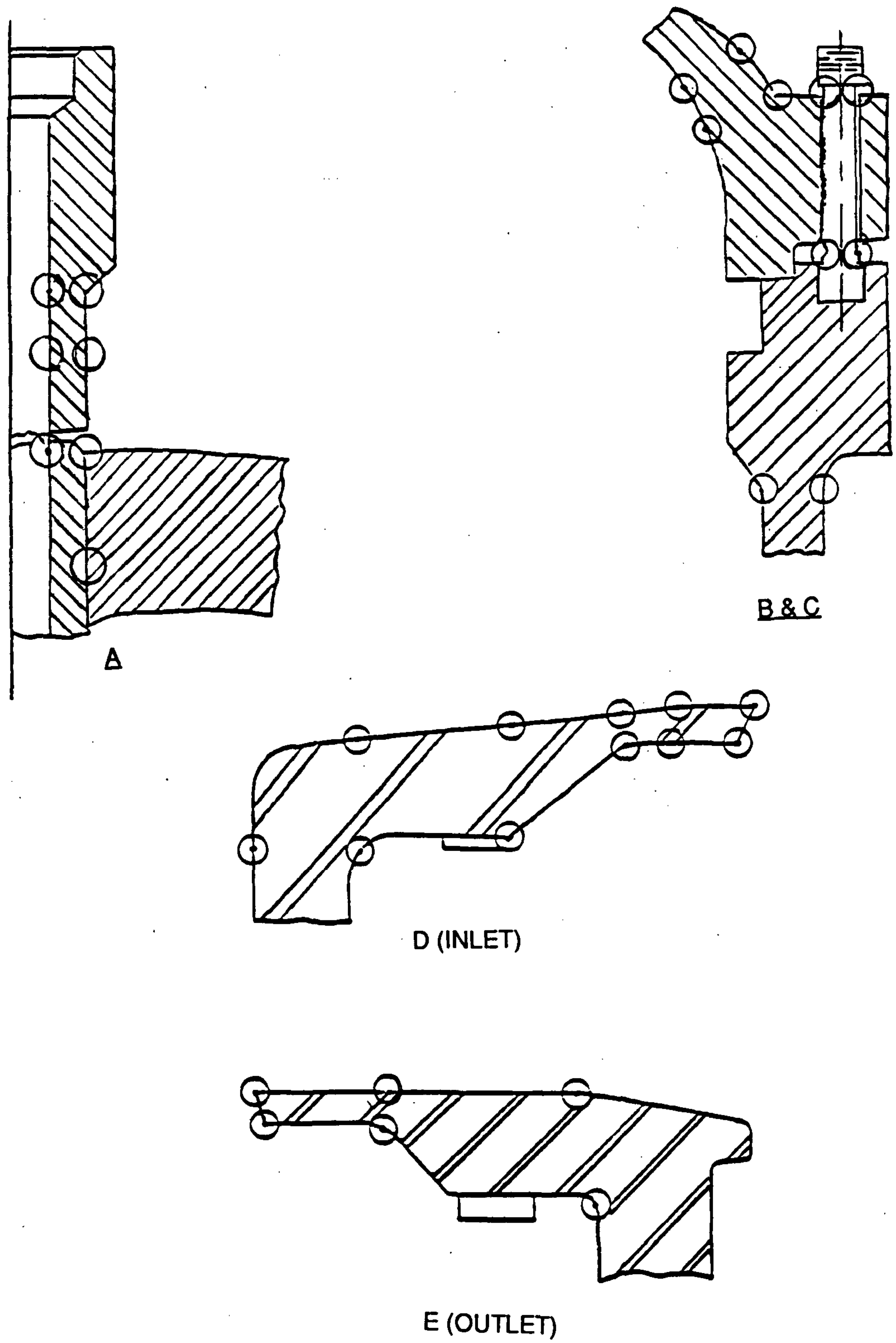

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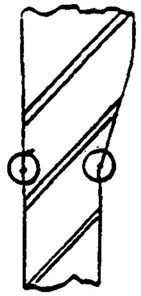
Figure 5.3-7

REACTOR VESSEL STRESS ANALYSIS: DETAILS-UPPER



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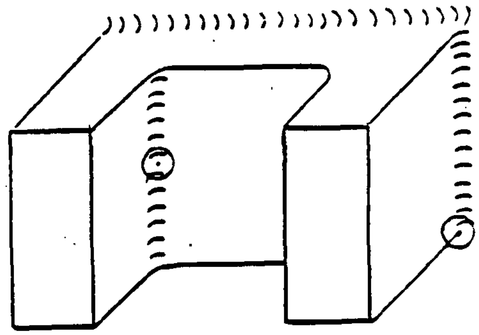

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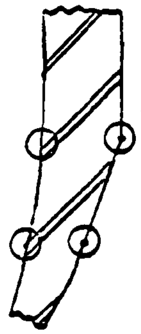
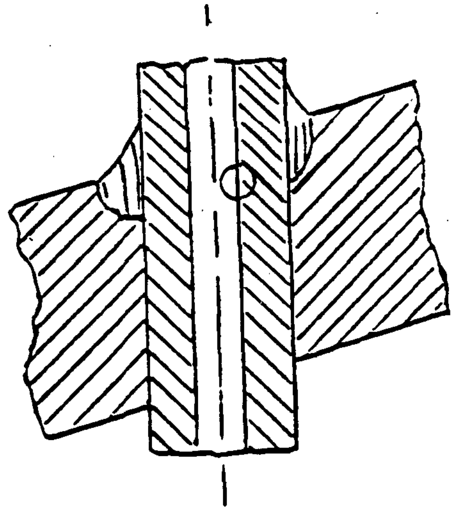


C



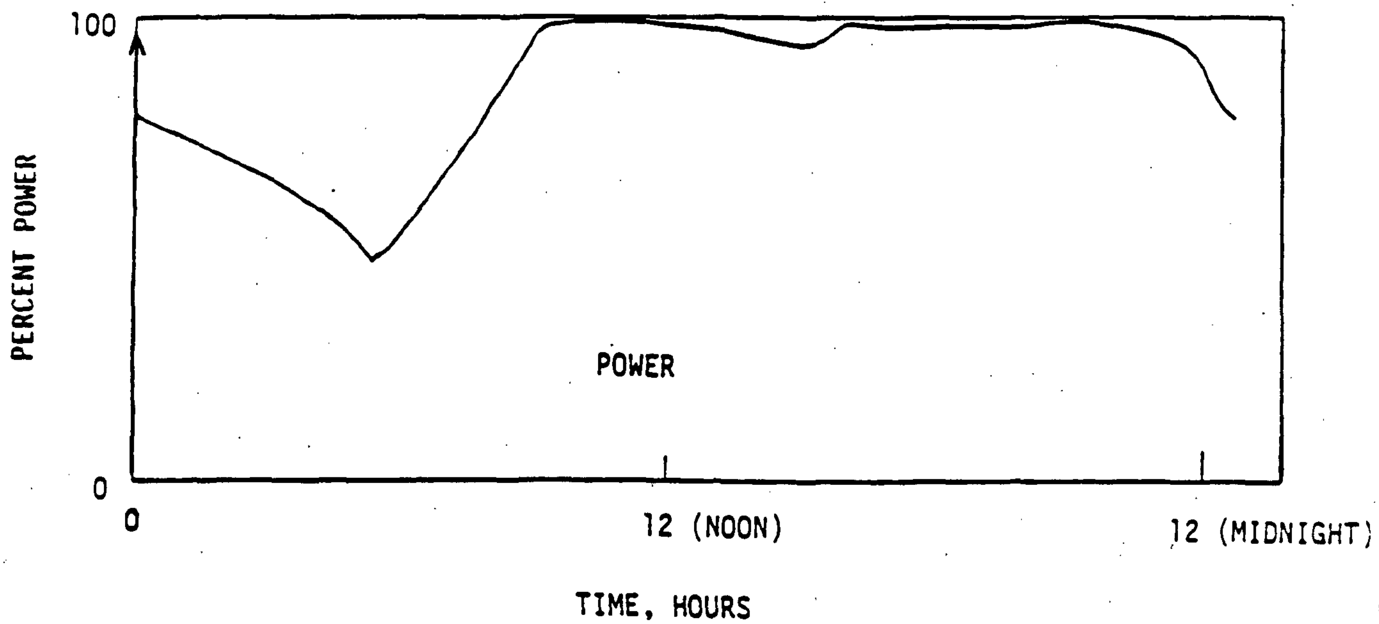
H

NOTE:
THE POINTS CIRCLED IN THE SKETCHES REPRESENT THE GENERAL LOCATION AND GEOMETRY OF THE AREAS OF DISCONTINUITY AND/OR STRESS CONCENTRATION.

ZION STATION UFSAR

Figure 5.3-8

REACTOR VESSEL STRESS ANALYSIS: DETAILS-LOWER



ZION STATION UFSAR

Figure 5.3-9

TYPICAL LOAD FOLLOW PROGRAM

5.4 COMPONENT AND SUBSYSTEM DESIGN

The principal systems which are interconnected with the Reactor Coolant System (RCS) are the Steam and Feedwater Systems, Chemical Volume and Control System (CVCS) and the Safety Injection (SI) and Residual Heat Removal (RHR) Systems. The RCS is dependent upon the steam generators, and the Steam, Feedwater, and Condensate Systems for decay heat removal from normal operating conditions to a reactor coolant temperature of approximately 350°F. The layout of the system ensures the natural circulation capability to permit plant cooldown following a loss of all reactor coolant pumps (RCPs).

The flow diagrams of the Steam, Feedwater, and Condensate Systems are shown in Figures 10.1-1 through 10.1-6. In the event that the condensers are not available to receive the steam generated by residual heat, the water stored in the Feedwater System may be pumped into the steam generators and the resultant steam is vented to the atmosphere. The Steam, Feedwater, and Condensate Systems are described in Chapter 10. In the event that the main feedwater pumps are inoperable, the Auxiliary Feedwater (AFW) System will supply water to the steam generators. The AFW System is described in Section 6.8.

The SI System and the Emergency Core Cooling System (ECCS) portion of the RHR System are described in Chapter 6. The CVCS is described in Chapter 9.

5.4.1 Reactor Coolant Pumps

5.4.1.1 Design Description

Each reactor coolant loop contains a vertical, single-stage, mixed-flow pump which employs a controlled leakage seal assembly. A view of a controlled leakage pump is shown in Figure 5.4-1 and the principal design parameters for the pumps are listed in Table 5.4-1. The RCP estimated performance and net positive suction head (NPSH) characteristics are shown in Figure 5.4-2.

Reactor coolant is drawn up through the primary pump impeller, discharged through passages in the diffuser and exits through a discharge nozzle in the side of the casing. The rotor-impeller can be removed from the casing for maintenance or inspection without removing the casing from the piping. All parts of the pumps in contact with the reactor coolant are austenitic stainless steel or equivalent corrosion resistant materials.

The pump employs a controlled leakage seal assembly to restrict leakage along the pump shaft, a second seal which directs the controlled leakage out of the pump, and a third seal which minimizes the leakage of water out of the pump. The second and third seals drain to the reactor coolant drain tank.

A portion of the high pressure water flow from the charging pumps is injected into the RCP between the impeller and the controlled leakage seal. Part of the flow enters the RCS through a labyrinth seal in the lower pump shaft to serve as a buffer to keep reactor coolant from entering the upper portion of the pump. The remainder of the injection water flows along the drive shaft, through the controlled leakage seal, and finally out of the pump. A very small amount which leaks through the second seal is also collected and removed from the pump. Flow measuring devices are installed in the leakoff lines of the #1 and #2 seals.

An extensive test program has been conducted for several years to develop the controlled leakage shaft seal for pressurized water reactor (PWR) applications. Long term tests have been conducted on less than full-scale prototype seals as well as on full-size seals. San Onofre, Connecticut Yankee, and others have demonstrated satisfactory performance of the controlled leakage shaft seal pump.

The squirrel cage induction motor driving the pump is air cooled and has oil lubricated thrust and radial bearings. A water lubricated bearing provides radial support for the pump shaft.

Component cooling water is supplied to the motor bearing oil coolers and the thermal barrier cooling coil. The thermal barrier cooling coil insures the cooling of seal water in the event of loss of injection water.

5.4.1.2 Bearings

The RCP motor bearings are of conventional design. The radial bearings are the segmented pad type, and the thrust bearings are tilting pad Kingsbury bearings. All are oil lubricated; the lower radial bearing and the thrust bearings are submerged in oil, and the upper radial bearing is oil fed from an impeller integral with the thrust runner. Low oil levels would signal an alarm in the Control Room and require shutting down of the pump. Each motor bearing contains embedded temperature detectors, so initiation of failure, separate from loss of oil, would be indicated and alarmed in the Control Room as a high bearing temperature. This, again, would require pump shutdown. Even if these indications were ignored, and the bearing proceeded to failure, the low melting point babbitt metal on the pad surfaces would ensure that no sudden seizure of the bearing would occur. In this event, the motor would continue to drive, as it has sufficient reserve capacity to operate, even under such conditions. However, it would demand excessive currents and, at some stage, would be shut down because of high current demand.

The design requirements of the bearings are primarily aimed at ensuring a long life with negligible wear, so as to give accurate alignment and smooth operation over long periods of time. Therefore, the surface bearing stresses are held at a very low value, and even under the most severe seismic transients or other accidents, they do not begin to approach loads which cannot be adequately carried for short periods of time.

Because there are no established criteria for short-time stress-related failures in such bearings, it is not possible to make a meaningful quantification of such parameters as margins to failure, safety factors, etc. A qualitative analysis of the bearing design, embodying such considerations, gives assurance of the adequacy of the bearing to operate without failure.

5.4.1.3 Locked Rotor

It may be hypothesized that the pump impeller might severely rub on a stationary member and then seize. Analysis has shown that under such conditions, assuming instantaneous seizure of the impeller, the pump shaft would fail in torsion just below the coupling to the motors. This would constitute a loss-of-coolant flow in the one loop, the effect of which is analyzed in Chapter 15. Following the seizure, the motor would continue to run without any overspeed, and the flywheel would maintain its integrity, as it would still be supported on a shaft with two bearings.

There are no other credible sources of shaft seizure other than impeller rubs. Any seizure of the pump bearing would be precluded by shearing of the graphitar in the bearing. Any seizure in the seals would result in a shearing of the antirotation pin in the seal ring. The motor has adequate power to continue pump operation even after the above occurrences. Indications of pump malfunction in these conditions would initially be high temperature signals from the bearing water temperature detector and excessive No. 1 seal leakoff indications, respectively. Following these signals, pump vibration levels would be checked and found to be excessive, and the pump would be shut down for investigation.

5.4.1.4 Critical Speed

The shafts are Type 347 stainless steel (American Society of Testing Materials (ASTM) A-182 Grade F). This grade has columbium added to prevent sensitization during thermal treatments and was specifically chosen for this application to give assurance that the material will not become sensitized during the stress-relieving treatment required for dimensional stability.

As is generally the case with machines of this size, the shaft dimensions are predicated on avoidance of shaft critical speed conditions, rather than actual levels of stress.

There are many machines as large as the RCPs, and larger, that are designed to run at speeds in excess of first shaft critical. However, it is considered desirable for a superior product to operate below first critical speed, and the RCPs are designed in accordance with this philosophy. This results in the shaft design which, even under the severest postulated transient, gives very low values of actual stress. While it would be possible to present quantitative data of imposed operational stress relative to maximum tolerable levels, if the mode of postulated failure were clearly defined, such figures would have little significance in a meaningful assessment of the adequacy of the shaft to maintain its integrity under operational transients. However, a qualitative assessment of such factors gives assurance of the conservative stress levels experienced during these transients.

So in each of these cases, the functional requirements of the component control its dimensions. If these are met, the stress-related failure cases are more than adequately satisfied.

It is thus considered to be out of the bounds of reasonable credibility that any bearing or shaft failure could occur that would endanger the integrity of the pump flywheel. There would not be added safety value gained by the installation of RCP overspeed protection devices.

5.4.1.5 Missiles

Precautionary measures, taken to preclude missile formation from RCP components, assure that the pumps will not produce missiles under any anticipated accident condition.

Each component of the RCP motors has been analyzed for missile generation. Any fragments of the motor rotor would be contained by the heavy stator. The same conclusion applies to the pump impeller, because the small fragments that might be ejected would be contained by the heavy casing.

The most adverse operating condition of the flywheels is visualized to be the loss-of-load situation. The conservative design operation conditions preclude missile production by the pump flywheels.

5.4.1.6 Seismic Considerations

The design specifications for the RCPs include as a design condition the stresses generated by the ground acceleration due to a Design Basis Earthquake (DBE). Besides examining the externally produced loads from the nozzles and support lugs, an analysis is made of the effect of gyroscopic reaction on the flywheel and bearings and in the shaft, due to rotational movements of the pump about a horizontal axis, during the maximum seismic disturbance.

The pump would continue to run, unaffected by such conditions. In no case does any bearing stress in the pump or motor exceed or even approach a value which the bearing could not carry.

5.4.1.7 Pressure Retaining Capability

All the pressure bearing parts of the RCP are analyzed in accordance with Article 4 of the ASME Boiler and Pressure Vessel (B&PV) Code, Section III. This includes the casing, the main flange, and the main flange bolts. The analysis includes pressure, thermal, and cyclic stresses, and these are compared with the allowable stresses in the Code. No emergency or faulted condition categories were defined or recognized by the design codes identified in Table 5.2-1.

Mathematical methods of the parts are prepared and used in the analysis which proceeds in two phases.

1. In the first phase, the design is checked against the design criteria of the ASME B&PV Code with pressure stress calculations. Thermal effects are included implicitly with the experience factors. By this procedure, the shells are profiled to attain optimum metal distribution with stress levels adequate to meet the more limiting requirements of the second phase.

2. In the second phase, the interactivity forces needed to maintain geometric capability between the various components are determined at design pressure and temperature and are applied to the components along with the external loads to determine the final stress state of the components. These are compared with the Code allowable values.

There are no other sections of the Code which are specified as areas of compliance, but where Code methods, allowable stresses, fabrication methods, etc., are applicable to a particular component, they are used to give a rigorous analysis and conservative design.

5.4.1.8 Flywheels

A flywheel on the shaft above the motor provides additional inertia to extend flow coastdown. Each pump contains a ratchet mechanism to prevent reverse rotation. The RCP flywheel is shown in Figure 5.4-3.

The flywheel blanks are fabricated from rolled, vacuum degassed, ASTM A-533 Grade B Class 1 steel plates. Flywheel blanks are flame-cut from the plate, with allowance for exclusion for flame-affected metal. A minimum of three Charpy tests are made from each plate parallel (RW, longitudinal) and normal (WR, transverse) to the rolling direction; they determine that each blank satisfies the design requirements. A nil ductility transition temperature (NDTT) less than +10°F is specified. Westinghouse has a great deal of experience and data in determining fracture toughness of A-533 Grade B Class 1 steel utilizing fracture mechanics specimens as well as Charpy-V specimens. Fracture mechanics specimens up to 12 inches in thickness have been tested to characterize A-533 Grade B material. From

Westinghouse's experience and those of others found in the literature, an empirical relationship can be established for Charpy-V data and fracture toughness data (K_{Ic}).

These design-fabrication techniques yield flywheels with primary stress at operating speed (shown in Figure 5.4-4) less than 50% of the minimum specified material yield strength at room temperature (100° to 150°F). Bursting speed of the flywheels has been calculated on the basis of Griffith-Irwin's results (see Reference 1 and 2), to be 3900 rpm, more than three times the operating speed.

A fracture mechanics evaluation was made on the RCP flywheel. This evaluation considered the following assumptions:

1. Maximum tangential stress at an assumed overspeed of 125%;

2. A crack through the thickness of the flywheel at the bore; and

3. 400 cycles of startup operation in 40 years.

Using critical stress intensity factors and crack growth data attained on flywheel material, the critical crack size for failure was greater than 17 inches radially and the crack growth data was 0.030 to 0.060 inches per 1000 cycles.

The finished flywheels are subjected to 100% volumetric ultrasonic inspection using Section III Class A code. The finished machined bores are also subjected to magnetic particle or liquid penetrant examination. No preoperational overspeed tests on the flywheel were performed.

The flywheels were visually examined during the first refueling and will be reexamined at the end of each 10-year interval. The outside surface of all flywheels were examined by ultrasonic techniques during the second refueling and were reexamined in the third period of the first ten year interval. Upon disassembly for maintenance, or within each of the followup 10 year intervals, all flywheels shall have a surface examination performed on the bore and key way areas. Additionally, a visual exam shall be performed on 100% of the flywheel surface, including the pawls. Where new inspection techniques are developed which would extend existing capabilities, these inspection techniques will, where possible, be incorporated into the Inservice Inspection Program. Acceptance criteria shall be in accordance with ASME III Class A.

5.4.1.9 Tests and Inspections

The RCPs are inspected in accordance with Zion Station's Inservice Inspection Program.

5.4.2 Steam Generators

The steam generators are vertical shell and U-tube heat exchangers with integral moisture separating equipment. The reactor coolant flows through the inverted U-tubes, entering and leaving through the nozzles located in the hemispherical bottom head of the steam generator. Antivibration bars (AVB) are installed on the tube bundles to help prevent flow induced vibrations. The head is divided into inlet and outlet chambers by a vertical partition plate extending from the head to the tube sheet. Manways are provided for access to both sides of the divided head. Feedwater enters the steam generator through a feedring, which has been provided with J-tubes to reduce the potential for water hammer. The feedring is located just above the U-tubes and the water is directed down through the annulus between the tube wrapper and the shell, and then upward through the tube bundle. Steam is generated on the shell side and flows upward through the moisture separators to the outlet nozzle at the top of the vessel.

The units are primarily low-alloy steel. The heat transfer tubes are Inconel, the primary side of the tube sheets are clad with Inconel, and the interior surfaces of the reactor coolant channel heads and nozzles are clad with austenitic stainless steel. A steam generator of this type is shown in Figure 5.4-5 and design data are given in Table 5.4-2.

The steam generators were constructed with weld deposited cladding in the center lane where the partition plates attach to the tube sheet. Previous experiences with clad detachment were in plants with partition plates attached to deta (explosive deposited) cladding. No clad detachment on steam generators with weld deposited cladding has occurred.

The steam generators are designed to produce the steam flow required at full power operation. The internal moisture-separating equipment is designed to insure that the moisture carryover will not exceed 0.25% by weight under the following conditions:

1. Steady-state operation up to 105% of full-load steam flow, with water at the normal operating level;

2. Loading or unloading at a rate of 5% of full power steam flow per minute in the range from 15% to 105% of full load steam flow; or

3. A step-load change of 10% of full power in the range from 15% to 105% full load steam flow.

5.4.2.1 Blowdown and Seismic Loads

Calculations confirm that the steam generator tube sheet will withstand the loading (which is quasi-static rather than a shock loading) by loss of reactor coolant. The maximum primary membrane stress plus primary bending stress in the tube sheet under these conditions is 23,853 psi. This is well below ASME Section III yield strength of 41,112 psi at 660°F.

The rupture of primary or secondary piping has been assumed to impose a maximum pressure differential of 2485 psi across the tubes and tube sheet from the primary side; a secondary piping rupture assumes that a maximum pressure differential of 1100 psi across the tubes and tube sheet from the secondary side will be imposed. Under these conditions there is no rupture of the primary to secondary boundary, including the tubes and tube sheet. This criterion prevents any violation of the containment boundary.

To meet this criterion, it has been established that under the postulated accident conditions, where a primary to secondary side differential pressure of 2250 psia exists, the primary membrane stresses in the tube sheet ligaments, averaged across the ligament and through the tube sheet thickness, does not exceed 90% of the material yield stress at the operating temperature.

A complete tube sheet analysis was performed to verify the structural integrity of the primary-secondary boundary under blowdown plus seismic conditions.

Also, the primary membrane stress plus primary bending stress in the tube sheet ligaments, averaged across the ligament width at the tube sheet surface location giving maximum stress, does not exceed 135% of the material yield stress at the operating temperature. This criterion is felt to be applicable to abnormal operating circumstances in that it is consistent with the ASME B&PV Code, Section III, Paragraph N-712 for hydrotest criteria. The stresses and stress factors in the actual design tube sheet, obtained by using the above stress criteria, are given in Table 5.4-3.

The tubes have been designed to the requirements (including stress limitations) of Section III for normal operation, assuming 2485 psig as the normal operating pressure differential. Hence, the secondary pressure loss accident condition imposes no extraordinary stress on the tubes beyond that normally expected and considered in Section III requirements.

No significant corrosion of the Inconel tubing is expected during the lifetime of the unit. The corrosion rates reported by Berry and Fink (see Reference 3) show a "worst case" rate of 15.9 mg/dm^2 in the 2000-hour test under steam generator operating conditions. Conversion of this rate to a 40-year plant life gives a corrosion loss of 1.3 x 10^{-3} inch which is insignificant compared to the nominal tube wall thickness of 0.050 inch.

In the case of a primary pressure loss accident, the secondary-to-primary pressure differential can reach 1100 psig. This pressure differential is less than the primary-to-secondary design pressure differential (1520 psi) for normal operating conditions. Hence, no stresses in excess of those covered in Section III for normal operation are experienced on the tube sheet for this accident case. ASME Section VIII design curves for iron-chromium- nickel steel cylinders under external pressure indicate a collapse pressure of 2310 psi for tubes having the minimum factor of safety of 2.4 against collapse. Collapse tests of $^{7}/_{8}$-inch outside diameter straight tubes with 0.050-inch thick walls at room temperature indicate actual tube strengths are significantly higher than specification, and a collapse pressure of 6000 psi was recorded for the straight tube. The Code charts indicate a collapse pressure of 2740 psi for this tube. The difference is attributed to the fact that the yield strength of the tube tested was 44,000 psi. The Code charts are based on a yield strength of approximately 29,000 psi at room temperature.

Consideration has been given to the superimposed effects of secondary side pressure loss and the DBE loading. The fluid dynamic forces on the internal components affecting the primary-to-secondary boundary (tubes) have been considered as well. For this condition, the criterion is that no rupture of the primary-to-secondary boundary (tubes and tube sheet) occurs.

For the case of the tube sheet, the DBE loading will contribute an equivalent static pressure loading over the tube sheet of less than 10 psi (for vertical shock). Such an increase is small when compared to the pressure differentials (up to 2485 psig) for which the tube sheet is designed. Under horizontal shock loading of the Design Basis Earthquake, the stresses are less than those for 1.0g gravity loading experienced in a horizontal position, which the design can readily accept.

The fluid dynamic forces on the internals under secondary side steam break accident conditions indicate, in the most severe case, that the tubes are adequate to constrain the motion of the baffle plates with some plastic deformation, while boundary integrity is maintained.

The ratio of the allowable stresses on various components (based on an allowable membrane stress of 0.9 of the nominal yield stress of the material) to the computed stresses for a primary-to-secondary pressure differential of 2485 psig are summarized in Table 5.4-4.

The evaluation of Westinghouse steam generator tube sheets is performed in accordance with the ASME B&PV Code for Nuclear Vessels, Section III, Article 4 - Design. The design criteria considered steady state, transient, and emergency operations specified in the Equipment Specification. Due to the complex nature of the tube-tubesheet-shell-head structure, the analysis of the tubesheet required the application of results of related research programs (such as the design data on perforated plates resulting from Piping and Valve Review Committee (PVRC) programs) and the utilization of then current techniques in computer analysis, the application of which was verified by comparison of analytical and experimental results for related equipment.

The introductory paragraph, I-900, of the ASME B&PV Code, Section III states that consideration may be given to the stiffening effect of tubes in perforations and staying action of the tubes (if applicable), effect of stiffening on the plate stress levels, etc. Further, the stress analysis methods in Appendix I of Section III are described as accepted techniques for obtaining solutions to problems for which these procedures are applicable. Use of other valid analytical or experimental techniques are allowed and are required where necessary.

The Nuclear Pressure Vessel Code Article I-9 provides the techniques for analysis of perforated plates, but the stress intensity levels for perforated plates are given for triangular perforation arrays. Westinghouse tube sheets contain square hole arrays. Hence, Westinghouse utilizes its own data and that obtained from PVRC research in square array perforation patterns for development of similar charts for stress intensity factors and elastic constants. The resulting stress intensity levels and fatigue stress ranges are evaluated according to the stress limitation of the Code.

The Westinghouse analysis of the steam generator tubesheet is included as part of the Stress Report requirement for Class A Nuclear Pressure Vessels. The evaluation is based on the stress and fatigue limitations outlined in Article 4 of Section III. The stress analysis techniques utilized include all factors considered appropriate for conservative determination of the stress levels utilized in evaluation of the tubesheet complex. The analysis of the tubesheet complex includes the effect of all appurtenances attached to the perforated region of the tubesheet and involves the heat conduction and stress analysis of the tubesheet, channel head, and secondary shell structure for particular steady design conditions for which Code stress limitations were to be satisfied. Also included were discrete points during transient operation for which the temperature/pressure conditions must be known to evaluate maximum and minimum stress for fatigue life usage. In addition, limit analyses were performed to determine tubesheet capability to sustain emergency operating conditions for which elastic analysis does not suffice. The analytic techniques utilized were computerized and significant stress problems were verified experimentally to justify the techniques when possible.

The analytic treatment of the tube-tubesheet complex includes determination of elastic equivalent plate stress within the perforated region from an interaction analysis utilizing effective elastic constants appropriate to the nature of the perforation array. For the perforated region of the tubesheet, the flexural rigidity is based on studies of behavior of plates with square hole arrays utilizing techniques such as those reported by O'Donnell (see Reference 4), Mahoney (see Reference 5), Lemcoe (see Reference 6), and others. Similarly, stress intensity factors are determined for square hole arrays using the combined equivalent plate interaction forces and moments applied to results of photo-elastic tests of model coupons of such arrays, as well as verification using computer analysis techniques such as "Point Matching" or "Collocation". The stress analysis considers stress due to symmetric temperature and pressure distribution, as well as asymmetric temperature distribution due to temperature drop across the tubesheet divider lane.

The fatigue analysis of the complex was performed at potentially critical regions in the complex, such as the junction between tubesheet and channel head or secondary shell, as well as at many locations throughout the perforated region of the tubesheet. For the holes for which fatigue evaluation is done, several points around the hole periphery were considered to assure that the maximum stress excursion was considered. The fatigue evaluation was computerized to include stress maxima-minima excursions considered on the intra-transient basis.

The evaluation of the tube-to-tubesheet juncture of Westinghouse PWR System steam generators is based on a stress analysis of the interaction between tube and tubesheet hole for the significant thermal and pressure transients that are applied to the steam generator in its predicted histogram of cyclic operation. The evaluation is based on the numerical limits specified in the ASME B&PV Code, Section III.

In the analysis of the interaction system, the tube hole behavior is a function of the behavior of the entire tubesheet complex with attached head and shell. Hence, the output of the tubesheet analysis, giving equivalent plate stresses in the perforated region, was utilized in determining the free boundary displacements of the perforation to which the tube is attached.

Analysis of the juncture for the fillet-type weld utilized in the Westinghouse steam generator design was made with consideration of the effect of the rolled-in joint in the weld region, and that the tube flexure, relative to the perforation, is not inhibited by the rolled-in effect.

The major concern in fatigue evaluation of the tube weld is the fatigue strength reduction factor to be assigned to the weld root notch. For this reason, Westinghouse conducted low-cycle fatigue tests of tube material samples to determine the fatigue strength reduction factor and applied it to the analytic interaction analysis results in accordance with the accepted techniques in the Nuclear Pressure Vessel Code for Experimental Stress Analysis. The fatigue strength reduction factor determined therefrom is not different from that reported in the well-known paper on the subject by O'Donnell and Purdy (see Reference 7). An actual tubesheet joint contained in a tubesheet was successfully tested experimentally under thermal transient conditions much more severe than that achieved in anticipated power plant operation.

A wide range of computational tools were utilized in these solutions including finite element, heat conduction and thin shell computer solutions. In addition, analysis techniques have been verified by photo-elastic model tests and strain gaging of prototype models of an actual steam generator tubesheet.

To evaluate the ultimate safety of the structural complex, a computer program for determining a lower-bound pressure limit for the complex based on elastic-plastic analysis was developed and applied to the structure. This was verified by a strain gage on a steel model of the complex which was tested to failure.

In all cases evaluated, the Westinghouse steam generator tubesheet complex meets the stress limitations and fatigue criteria specified in Article 4 of the Code as well as emergency condition limitations specified in the Equipment Specifications or otherwise anticipated.

In this way the tube-tubesheet integrity of a Westinghouse steam generator was demonstrated under the most adverse conceivable conditions resulting from a major breach in either the primary or secondary system piping.

Tabulations of significant results of the tubesheet complex are in Tables 5.4-5 through 5.4-12 and Figures 5.4-6 through 5.4-8. Figure 5.4-9 denotes the primary-secondary boundary components shell locations.

5.4.2.2 Secondary Side Flow Instabilities

Steam generator water hammer has occurred as a result of the rapid condensation of steam in the feedwater line and the consequent acceleration of a slug of water, which upon impact with the piping system, causes undue stresses in the piping and its support system. The potential for steam generator water hammer is eliminated if the feedwater system is maintained full of water.

J-shaped discharge tubes have been installed on the top of each steam generator feedring to provide for top discharge of water rather than bottom discharge. During periods of feedring uncovery, this arrangement increases the time for complete drainage of the feedring and associated horizontal piping, from less than one minute to about 30 minutes, over the original bottom hole discharge design. Also, the J-tube design permits feedwater flow rates as low as 10 gpm to keep the feedring and feedwater piping full of water until feedring recovery occurs. The J-tube arrangement, in conjunction with prompt automatic initiation of auxiliary feedwater flow, eliminates the potential for water hammer.

The safety and technical evaluations for steam generator water hammer at Zion Station are presented in Appendices 10B and 10C, respectively.

5.4.2.3 Steam Generator Inservice Inspection

The steam generators are inspected in accordance with the Technical Specifications. The ASME Code for Inservice Inspection of Nuclear Power Plant Components, Section XI, is used as a guideline for inservice inspection requirements to the maximum extent possible.

Steam generator tubes are inspected in accordance with the Technical Specifications. Steam generator tubes may be repaired by installing sleeves and then returned to service. If a tube exceeds the allowed degradation limits as specified in Technical Specifications, it is plugged and removed from service.

5.4.3 Reactor Coolant Piping

The reactor coolant piping and fittings which make up the loops are austenitic stainless steel. All smaller piping which comprises part of the RCS boundary, such as the pressurizer surge line, spray and relief line, loop drains, and connecting lines to other systems, are also austenitic stainless steel. All joints and connections are welded, except for the pressurizer relief and the pressurizer code safety valves, where flanged joints are used. Thermal sleeves are installed at points in the system where high thermal stresses could develop due to rapid changes in fluid temperature during normal operational transients. These points include:

1. Charging connections from the CVCS;

2. Return lines from the RHR Loop (also part of the ECCS);

3. Both ends of the pressurizer surge line; and

4. Pressurizer spray line connection to the pressurizer.

Thermal sleeves are not provided for the remaining injection connections of the ECCS since these connections are not in normal use.

All piping systems have been designed and supported to preclude excessive vibration under startup and operating conditions, as delineated in Paragraph 116(a) of American Standards Association (ASA) B31.1 This is accomplished by means of variable spring hangers, rigid supports, constant support hangers, pipe anchors, guides, and snubbers.

During the course of the Preoperational Test Program, all piping systems were visually checked for excessive vibration under such transient conditions as are imposed by routine starting and stopping of pumps and opening and closing of valves. A separate "Vibration Operational Test Program" was not required by B31.1.

In addition to visually checking all systems for excessive vibration during the Preoperational Test Program, specific attention was directed at evaluating possible vibration problems during the performance of the transients listed in Table 5.4-26.

No specific "go-no go" criteria has been established for determining the acceptance of piping systems or components in terms of vibration requirements. Systems and components were physically examined (visually) for the following types of deficiencies which are indicative of a possible vibration problem:

1. Cracks in the grouting of equipment foundations;

2. Leaking gaskets in piping systems and pump seals;
3. Leaks from flanged connections in piping systems;
4. Metal to metal contact indications on piping system restraints; or
5. Water hammer "noises" during transient operations.

If the above types of indications were observed, further investigation was performed to establish and correct any adverse conditions.

All piping connections from auxiliary systems are made above the horizontal centerline of the reactor coolant piping, with the exception of:

1. RHR pump suction, which is 45 degrees down from the horizontal centerline. This enables the water level in the RCS to be lower in the reactor coolant pipe while continuing to operate the RHR System should this be required for maintenance;

2. Loop drain lines and the connection for level measurement of water in the RCS during refueling and maintenance operation; and

3. The differential pressure taps for flow measurement are downstream of the steam generators on the 90° elbow.

Penetrations into the coolant-flow path are limited to the following:

1. The spray line inlet connections extend into the cold-leg piping in the form of a scoop so that the velocity head of the reactor coolant loop flow adds to the spray driving force;

2. The Reactor Coolant Sample System taps are inserted into the main stream to obtain a representative sample of the reactor coolant;

3. The resistance temperature detector (RTD) hot leg bypass connections are scoops which extend into the reactor coolant to collect a representative temperature sample for the RTD manifold; and

4. The wide range temperature detectors are located in RTD wells that extend into the reactor coolant pipes.

Principal design data for the reactor coolant piping are given in Table 5.4-13.

Following is a basic description of the procedures that were originally used to perform a dynamic analysis of the Reactor Coolant Loop Piping System.

The analysis of the Reactor Coolant Loop/Supports System was based on an integrated analytical model which included the effects of the supports and the supported equipment.

A three-dimensional, multimass, elastic-dynamic model was constructed to represent the Reactor Coolant Loop/Supports System. The seismic floor spectrum at the internal concrete-to-support interface, obtained from an elastic-dynamic model of the reactor containment internal structure, was used as input to the piping analysis.

The dynamic analysis employed displacement method, lumped parameter, stiffness matrix formulations and assumed that all components behave in a linearly elastic manner. The proprietary computer code WESTDYN was used in this analysis. Table 5.4-14 describes the loading conditions and corresponding stress limits employed in all Seismic Class I piping, including RCS piping.

The stresses resulting from the different loading conditions were combined in a conservative manner and are compared to the allowable values as noted on Tables 5.4-15 through 5.4-18.

The Faulted Conditions or "Limiting Faults" are events which are not expected to occur, but are postulated because their consequences may impair public health and safety. The DBE is an example of such a postulated event due to its extremely low probability of occurrence. Protection of public health and safety during and after the DBE is assured by designing the critical structures and equipment so that the plant can be shutdown and kept in a safe shutdown condition.

The following describes the loading conditions and resulting stresses for the reactor coolant loop analysis. The completed analysis shows that the reactor coolant loop piping will experience stresses below the USA Standard (USAS)-B31.1.0-1967 allowables using these loading conditions. A complete description of the seismic analysis performed on piping is presented in Section 3.7.3.

5.4.3.1 Normal Operating Loads

System design operating parameters were used as the basis for the analysis of equipment, coolant piping and equipment support structures for normal operating loads. The analysis was performed using a static model to predict deformation and stresses in the system under normal operating conditions. The analysis, with respect to the piping and vessels, was in accordance with the provisions of USAS B31.1 and ASME Section III. Results of the analysis gave six generalized force components, three bending moments and three forces. These moments and forces were resolved into stresses in the pipe in accordance with the applicable codes. Stresses in the structural supports were determined by the material and section properties assuming linear elastic small deformation theory.

5.4.3.2 Seismic Loads

Analysis for seismic loads was based on a dynamic model. The appropriate floor spectral accelerations were used as input forcing functions to the detailed dynamic model. The loads developed from the dynamic model were incorporated into a detailed support model to determine the support member stresses. The seismic analyses of Seismic Class I piping, including the RCS piping, is discussed in detail in Section 3.7.3.

5.4.3.3 Blowdown Loads

Analysis of blowdown loads resulting from a loss-of-coolant accident was based on the time-history response of simultaneously applied blowdown forcing functions on a single broken loop dynamic model. The forcing functions are defined at points in the system loop where changes in cross section or direction of flow occur such that differential loads are generated during the blowdown transient. The loads developed from the dynamic model were incorporated into a detailed support model to determine the equipment support member stresses.

5.4.3.4 Combined Blowdown and Seismic Loads

The stresses in components resulting from normal loads and the worst case blowdown analysis were combined with the worst case seismic analysis to determine the maximum stress for the combined loading case and is discussed in Appendix 5A. This is considered a very conservative method since it is highly improbable that both maxima will occur at the same instant. These stresses were combined to determine that the Reactor Coolant Loop/Supports System will not lose its intended functions under this highly improbable situation. In the combination of loading, the DBE has been treated as part of the loading for the emergency and the faulted conditions.

5.4.4 Main Steamline Flow Restrictions

Each steamline is routed from its steam generator to one of the two feedwater-steamline tunnels by the shortest possible route. A flow restrictor is located in each line to limit the maximum flow and the resulting thrust forces created by a steamline break. The restrictor possesses a 16-inch diameter minimum section and is designed for minimum unrecovered pressure loss. It is located inside the Containment, as close to the steam generator as possible. This action minimizes the length of steam piping exposed to, and restrained against, the higher thrust loads.

The overall length of each flow limiter is approximately 67 inches. They are located in the main steamlines from each generator, 20 feet from the main steam outlet nozzle. The flow limiter is sized to limit the cooldown rate of the RCS so an adequate shutdown reactivity margin will be maintained after a trip at end of core life, following a steamline break upstream of the main steam isolation valves. The criteria used to design and locate steamline restraints downstream of the flow limiters was the same as that used upstream of the limiter; namely, any credible main steamline rupture within the Containment will not result in a loss of containment integrity. A detailed analysis of a steamline break inside Containment is presented in Section 15.1.5. An analysis of a steamline/feedline break outside Containment is presented in Appendix 3A. The effects of other high-energy line breaks occurring outside of the Containment are presented in Appendix 3B.

5.4.5 Main Steamline Isolation System (BWRs Only)

This section is not applicable to Zion Station.

5.4.6 Reactor Core Isolation Cooling System (BWRs Only)

This section is not applicable to Zion Station.

5.4.7 Residual Heat Removal System

The RHR System is designed to remove residual and sensible heat from the core and reduce the temperature of the RCS during the second phase of plant cooldown. During the first phase of cooldown, the temperature of the RCS is reduced by transferring heat from the RCS to the Steam and Power Conversion System (Chapter 10).

In addition, portions of the system are utilized as parts of the ECCS and the Containment Spray (CS) System. These functions and the associated analyses are discussed in this chapter. Refer to Figure 5.4-10 for a schematic presentation of the RHR System.

The RHR System provides sufficient capability in the emergency operational mode to accommodate any single active failure and still function in a manner to avoid risk to the health and safety of the public. Refer to Chapter 6 and Chapter 15 for a discussion of the operability and capability of the RHR System in an emergency core cooling role.

The system design precludes any significant reduction in the overall design reactor shutdown margin when cooling water is introduced into the core for decay heat removal or during the emergency core cooling recirculation mode of operation.

System components whose design pressure and temperature are less than the RCS design limits are provided with redundant isolation means and overpressure protective devices.

All system active components which are relied upon to perform the system functions are redundant, and the system design includes provision for hydrostatic testing of system components to applicable Code test pressures.

5.4.7.1 Residual Heat Removal System Description

The RHR System (shown in Figure 5.4-10) consists of two RHR heat exchangers, two RHR pumps, and associated piping, valves, and instrumentation. The instrumentation is discussed in Chapter 7.

During plant cooldown, coolant flows from the RCS to the RHR pumps, through the tube side of the RHR heat exchangers, and back to the RCS. The inlet line to the RHR System loop begins at the hot leg of reactor coolant loop A, and the return line connects to all four cold legs or can be routed to loops A and D hot legs. The RHR heat exchangers and pumps are used to cool and circulate the water during the latter phase of Emergency Core Cooling and CS System operation. These duties are defined in Chapter 6. The heat loads are transferred by the RHR heat exchangers to the component cooling water.

During plant cooldown, the cooldown rate of the reactor coolant is controlled by regulating the flow through the tube side of the RHR heat exchangers. A bypass line and a remotely-operated control valve around each RHR heat exchanger are used to maintain a constant flow through the RHR System.

The RHR System is isolated from the RCS by two remotely-operated valves on the suction side of the RHR pumps. These are the isolation valves from the RCS to the RHR pumps, MOV-RH8701 and MOV-RH8702. Two check valves in series and a remotely-operated valve provide isolation from the RCS in the discharge lines of the RHR pumps.

When RCS pressure exceeds the RHR System design pressure of 600 psig, an interlock between the RCS wide range pressure channels and the RHR inlet valves (MOV-RH8701 and MOV-RH8702) automatically closes these valves if they are open. A second interlock with a lower setpoint prevents these valves from opening if the 425 psig setpoint is exceeded.

Another set of remotely-operated valves (RWST to RHR pump suction valves MOV-RH8700A and MOV-RH8700B) are interlocked to prevent direct transfer of RCS water from the RHR pump hot leg suction to the Containment recirculation sump (via valves MOV-SI8811A and MOV-SI8811B) during normal conditions or from the Containment to the RWST (via valves MOV-SI8812A and MOV-SI8812B) during accident conditions. This interlock permits MOV-RH8700A (or MOV-RH8700B) to be open only if MOV-CS0049, MOV-CS0050, MOV-SI8804A, and MOV-SI8811A (or MOV-CS0049, MOV-CS0050, MOV-SI8804B, and MOV-SI8811B) are closed. Refer to Figure 5.4-10.

5.4.7.1.1 Codes and Classifications

All piping and components of the RHR System are designed to the applicable codes and standards listed in Table 5.4-19. Since the loop contains reactor coolant when it is in operation, austenitic stainless steel piping is employed.

5.4.7.1.2 Equipment and Component Description

The RHR component design data are listed in Table 5.4-20. A description of nondestructive tests applied to Seismic Class I components is included in Chapter 6. For further information on the nondestructive tests, refer to Table 6.3-2.

5.4.7.1.2.1 Residual Heat Exchangers

Two (Seismic Class I) RHR heat exchangers are installed in the system. Each exchanger is designed to remove one-half of the residual heat load. The design is based on heat load and temperature differences between tube side and shell side existing approximately 20 hours after reactor trip, when the temperature difference between the reactor coolant and the component cooling water is small.

The RHR heat exchangers are of the shell and U-tube type. Reactor coolant circulates through the tubes, while component cooling water circulates through the shell. The tubes are welded to the tube sheet to prevent leakage of reactor coolant.

The tubes and other surfaces in contact with reactor coolant are austenitic stainless steel, while the shell is carbon steel.

5.4.7.1.2.2 Residual Heat Removal Pumps

The two (Seismic Class I) RHR pumps are vertical, centrifugal units with mechanical seals to prevent reactor coolant leakage to the atmosphere. All pump parts in contact with reactor coolant are austenitic stainless steel or equivalent corrosion-resistant material.

A low-pressure alarm in the suction line of the RHR pumps annunciates in the Control Room. This alerts operators to take corrective action to prevent damage to the RHR pumps.

5.4.7.1.2.3 Residual Heat Removal System Valves

The valves used in the RHR System are constructed of austenitic stainless steel or equivalent corrosion-resistant material.

Manual isolation valves are provided to isolate equipment for maintenance. Throttle valves are provided for remote manual control of residual heat exchanger tube side flow, and for remote manual control of bypass flow. Check valves prevent reverse flow through the RHR pumps.

Subsequent to the original design, check valves (RH0257 and RH0258) were installed in RHR pump discharge piping to hydraulically decouple pump minimum flow recirculation paths. This configuration eliminates the potential for one pump to deadhead the other while running simultaneously on recirculation and thus prevent potential damage due to cavitation.

Isolation of the RHR System is achieved with two remotely-operated series stop valves in the pipe from the RCS to the RHR pump suction, and by two check valves in series plus a remotely-operated stop valve in each line from the RHR pump discharge to the RCS. Overpressure in the RHR System is relieved through a relief valve to the pressurizer relief tank in the RCS.

Valves that perform a modulating function are equipped with two sets of packing and an intermediate leakoff connection that discharges to the Waste Disposal System.

Manually-operated valves have backseats to facilitate repacking and to limit the stem leakage when the valves are open. Leakoff connections are provided where required by valve size and fluid conditions.

5.4.7.1.2.4 Piping

All RHR piping is austenitic stainless steel. All piping joints and connections are welded except where flanged connections are required to facilitate maintenance.

5.4.7.2 Design Evaluation

5.4.7.2.1 Availability and Reliability

For RCS cooldown, the unit is provided with two RHR pumps and two RHR heat exchangers. If one of the two pumps, or one of the two heat exchangers, or one pump and one heat exchanger is not operable, safe cooldown of the plant is not compromised; however, the time for cooldown is extended.

5.4.7.2.2 Incident Control

The RHR System is connected to the reactor coolant loop A hot leg on the suction side and to each of the reactor coolant piping cold legs on the discharge side. The discharge side is also connected to two of the reactor coolant piping hot legs. On the suction side, the connection is through two electric motor-operated gate valves in series which are interlocked with RCS pressure. On the discharge side, the connection is made through an electric motor-operated valve and two check valves in series.

Should a large tube side to shell side leak develop in a RHR heat exchanger, the water level in a component cooling surge tank would rise, and the operator would be alerted by a high-water alarm.

If the leaking RHR heat exchanger could not be isolated from the Component Cooling Water System before the inflow completely filled the surge tank, the overflow vent line would discharge the excess water to the drain header.

Since the RHR System is required for long-term, postaccident removal of decay heat from the reactor core and Containment, independent piping systems are provided for the redundant active components so that excessive leakage resulting from the deterioration of, or failure in, some passive element in the system can be identified and isolated without complete system loss of function.

Massive failure of piping is not considered credible because long term operation of the system occurs only at low pressures and temperatures, and the system is protected from environmental conditions by the Seismic Class I structures.

Special precautions have been taken to assure that, in the event of a pipe rupture, the RHR System will be able to function during the cooldown phase of the plant shutdown. The two redundant RHR pump discharge lines have been routed separately and have been located behind barriers in the Containment to prevent damage from other lines which might whip in the event of their rupture. Alternate methods of cooldown, such as the loop and drain line and the letdown line, have been routed separately from the RHR pump suction line to prevent their damage in the event that the suction line should rupture. Check valves on the RHR pump discharge lines and the normally-closed motor-operated gate valve on the RHR pump suction line have been located as close as possible to the reactor coolant loop connections, thereby shortening the length of pipe containing pressurized water and preventing pipe whip from occurring. All three RHR lines have been anchored to the missile barrier wall to prevent reactor coolant pipe rupture forces from being transferred to the Containment through the RHR branches.

5.4.7.2.3 Malfunction Analysis

A failure analysis of RHR pumps, heat exchangers, and valves is presented in Table 5.4-21.

5.4.7.3 Tests and Inspections

The RHR pump flow instrumentation is calibrated during each refueling operation. Periodic visual inspections and preventive maintenance are conducted during plant operation. Refer to Chapter 6 and the Technical Specifications.

5.4.8 Reactor Water Cleanup System (BWRs Only)

This section is not applicable to Zion Station.

5.4.9 Main Steamline and Feedwater Piping

Main steamline and feedwater piping does not form part of the reactor coolant pressure boundary on PWRs. Therefore, this section is not applicable to the Zion Station.

5.4.10 Pressurizer

5.4.10.1 Design Description

5.4.10.1.1 Pressurizer Vessel

The pressurizer provides a point in the RCS where liquid and vapor can be maintained in equilibrium, under saturated conditions, for control purposes.

The pressurizer is a vertical, cylindrical vessel with hemispherical top and bottom heads constructed of low-alloy steel, with austenitic stainless steel cladding on all surfaces exposed to the reactor coolant. Electrical heaters are installed through the bottom head of the vessel, while the spray nozzle and the relief and safety valve connections are located in the top head of the vessel. The heaters are removable for maintenance or replacement.

A missile barrier has been installed around that portion of the pressurizer that extends above the loop compartment. This barrier will prevent potential missiles from reaching the Containment liner, engineered safeguard pipes, or essential equipment which is located outside the reactor compartments and will protect the pressurizer from potential missiles.

The pressurizer is designed to accommodate positive and negative surges caused by load transients. The surge line, which is attached to the bottom of the pressurizer, connects the pressurizer to the hot leg of a reactor coolant loop. During an insurge, the spray system, which is fed from two cold legs, condenses steam in the vessel to prevent the pressurizer pressure from reaching the setpoint of the PORVs. The spray valves on the pressurizer are modulating, air-operated, ball-type, control valves. In addition, the spray valves can be operated manually by a manual/auto controller in the Control Room. A small continuous spray flow is provided through a manual bypass valve around the power-operated spray valves to minimize boron concentration differences between pressurizer liquid and reactor coolant and to prevent excessive cooling of the spray piping.

During an outsurge, flashing of water to steam and generating of steam by automatic actuation of the heaters keep the pressure above the minimum allowable limit. Heaters are also energized on high water level during insurges to heat the subcooled surge water entering the pressurizer from the reactor coolant loop. A screen at the surge line nozzle and baffles in the lower section of the pressurizer prevent cold insurge water from flowing directly to the steam/water interface and assist mixing.

The volume of the pressurizer is equal to, or greater than, the minimum required volume of steam, water, or total of the two which satisfies all of the following:

1. The combined saturated water volume and steam expansion volume is sufficient to provide the desired pressure response to system volume changes;

2. The water volume is sufficient to prevent the heaters from being uncovered during a step-load increase of 10% of full power;

3. The steam volume is large enough to accommodate the surge resulting from the design 50% step load reduction of full-load with reactor control and steam dump, without the water level reaching the high-level reactor trip point;

4. The steam volume is large enough to prevent water relief through the safety valves following a loss of load with the high water level initiating a reactor trip;

5. The pressurizer will not empty following reactor trip and loss of load; and

6. The safety injection signal will not be activated during normal reactor trip and turbine trip.

The general configuration of the pressurizer is shown in Figure 5.4-11 and the design data are given in Table 5.4-22.

5.4.10.1.2 <u>Pressurizer Spray</u>

Two separate, automatically controlled spray valves with remote manual overrides are used to initiate pressurizer spray. Pressurizer spray valve design employs a small hole through the valve plug assembly which permits a small continuous flow through both spray lines to reduce thermal stresses and thermal shock when the spray valves open and to help maintain uniform water chemistry and temperature in the pressurizer. Temperature sensors with low alarms are provided in each spray line to alert the operator to insufficient bypass flow. The layout of the common spray line piping to the pressurizer forms a water seal which prevents steam buildup back to the control valves. The spray rate is selected to prevent the pressurizer pressure from reaching the operating setpoint of the PORVs during a step reduction in power level of 10% full load.

The pressurizer spray lines and valves are large enough to provide adequate spray using the differential pressure between the surge line connection in the hot leg and the spray line connection in the cold leg as the driving force. The spray line inlet connections extend into the cold leg piping in the form of a scoop so that the velocity head of the reactor coolant loop flow adds to the spray driving force. The spray valves and spray line connections are arranged so that the spray will operate with one reactor coolant loop isolated. The line may also be used to assist in minimizing the boron concentration difference between the reactor coolant loops and the pressurizer.

A flow path from the CVCS to the pressurizer spray line is also provided. This additional facility provides auxiliary spray to the vapor space of the pressurizer during cooldown if the RCPs are not operating. The thermal sleeve on the pressurizer spray connection is designed to withstand the thermal stresses resulting from the introduction of cold spray water. The spray lines have been restrained to prevent pipe whip.

Principal design parameters of the pressurizer spray valves are given in Table 5.4-23.

5.4.10.1.3 <u>Pressurizer Surge Line</u>

The surge line is sized to limit the pressure drop during the maximum anticipated surge to less than the difference between the maximum allowable pressure in the reactor vessel and the loops (at the point of highest pressure) and the pressure in the pressurizer at the maximum allowable accumulation with the code safety valves discharging.

The surge line and the thermal sleeves at each end are designed to withstand the thermal stresses resulting from volume surges of relatively hotter or colder water which may occur during operation. The surge line has also been restrained to prevent pipe whip.

5.4.10.2 <u>Design Evaluation</u>

The Pressurizer was analyzed for fatigue conditions in accordance with Section III of the ASME B&PV Code using the thermal and pressure transient conditions listed elsewhere in this chapter.

The vessel loading conditions are as follows:

1. The pressurizer vessel, nozzles, and vessel supports are designed to resist the following normal operational loadings:

    a. Weight of water based on the vessel filled with cold water, including insulation; and

    b. Normal loadings exerted by connecting piping.

2. The pressurizer vessel, nozzles, and vessel supports are designed to resist the following seismic loadings:

   a. For the Operating Basis Earthquake (OBE), the pressurizer vessel is designed to resist earthquake loadings simultaneously in the horizontal and vertical directions and to transmit such loadings through the vessel supports to the foundation. The OBE mechanical loadings in combination with the normal operational loads is considered an upset condition. The components of loadings exerted by the external piping due to the OBE are included in this evaluation.

   b. For the DBE, pressurizer vessel integrity is not impaired so as to prevent a safe and orderly shutdown of the reactor plant when the DBE loadings, both horizontal and vertical and acting simultaneously, are imposed on the vessel. These loadings, and the centers of gravity involved, are determined on the basis of the vessel at normal operating pressure, temperature, and water level.

      The DBE is considered a faulted condition with the following exceptions:

      - The combination of all primary stress intensities in the vessel support skirt shall be within the support skirt material yield strength specified in Section III of the ASME B&PV Code.

      - The stress intensity limits of the vessel associated with this earthquake condition, in combination with the normal operational loads, shall be as follows:

        $$P_m \leq 1.2\ S_m$$
        $$P_L + P_b \leq 1.8\ S_m$$

        where:

P_m	=	primary general membrane stress
P_L	=	primary local membrane stress
P_b	=	primary flexural stress
S_m	=	allowable stress from ANSI B31.7.0 - 1969, Nuclear Piping Code

      The components of loadings exerted by the external piping due to the DBE are included in this evaluation.

3. The pressurizer vessel, nozzles and vessel supports are designed to resist the pipe break loadings in combination with the normal operational loads. The moment and forces are considered as acting in

combination with each force separately. The pipe break accident is considered to be a faulted condition except that the stress intensity limits are specified under the DBE condition.

4. The pressurizer vessel, nozzles and vessel supports are analyzed for the combination of normal operating loads, the DBE loads and the pipe break loads. The resulting stress intensities do not exceed the stress intensity limits of Paragraph N17.11 (faulted conditions) in Section III of the Code with the following exception. The combination of all primary stress intensities in the vessel supports are within the support material yield strength specified in the above Code. If necessary, higher stress intensity values are adopted in the vessel supports where plastic instability analyses of the support and supported component system are performed in accordance with paragraph N417.11 of ASME Code Section III.

5.4.11 Pressure Relief Discharge System

5.4.11.1 Discharge Piping

The discharge piping (from the code safety and air-operated relief valves to the relief tank) is sized to prevent back-pressure at the code safety valves from exceeding 20% of the setpoint pressure at full flow. The pressurizer code safety and PORV discharge lines are also stainless steel.

The design of the pressurizer relief and safety piping takes into account the worst case condition arising from multiple discharge reactions acting at each piping discharge elbow simultaneously. The combined longitudinal stress, including the flexural and torsional stresses for the upset, emergency, and faulted conditions, are below the allowable limit of $1.2S_h$, where S_h is the allowable stress from ANSI B31.1.0 - 1967. Sufficient piping restraints are provided to withstand the postulated DBE and normal transient operating conditions.

5.4.11.2 Pressurizer Relief Tank

The PRT condenses and cools the discharge from the pressurizer safety and relief valves. Discharge from smaller relief valves located inside or outside the Containment is also piped to the relief tank. The tank normally contains water and a predominantly nitrogen atmosphere; however, provision is made to permit the gas in the tank to be periodically analyzed to monitor the concentration of hydrogen and/or oxygen.

The PRT, by means of its connection to the Waste Disposal System, provides a means for removing any noncondensable gases from the RCS which might collect in the pressurizer vessel.

Steam is discharged through a sparger pipe under the water level. This condenses and cools the steam by mixing it with water that is near ambient temperature. The tank is equipped with an internal spray and a drain which are used to cool the tank following a discharge. The tank is protected against a discharge exceeding the design value by two rupture disks which discharge into the Reactor Containment. The tank is carbon steel with a corrosion-resistant coating on the wetted surfaces. A flanged nozzle is provided on the tank for the pressurizer discharge line connection. This nozzle and the discharge piping and sparger within the vessel are austenitic stainless steel.

The tank design is based on the requirement to condense and cool a discharge of pressurizer steam equal to 110% of the volume above the full-power pressurizer water level setpoint. The tank is not designed to accept a continuous discharge from the pressurizer. The volume of water in the tank is capable of absorbing the heat from the assumed discharge, with an initial temperature of 120°F and increasing to a final temperature of 200°F. If the temperature in the tank rises above 120°F during plant operation, the tank is cooled by spraying in cool water and draining out the warm mixture to the Waste Disposal System.

The spray rate is designed to cool the tank from 200°F to 120°F in approximately one hour following the design discharge of pressurizer steam. The volume of nitrogen gas in the tank is selected to limit the maximum pressure following a design discharge to 50 psig.

The rupture disks on the relief tank have a relief capacity equal to the combined capacity of the pressurizer safety valves. The tank design pressure is twice the calculated pressure resulting from the maximum safety valve discharge described above. This margin is to prevent deformation of the disk. The tank and rupture disk holders are also designed for full vacuum to prevent tank collapse if the contents cool following a discharge without nitrogen being added.

Principal design parameters of the PRT are given in Table 5.4-22.

5.4.12 Valves

All valves in the RCS which are in contact with the coolant are constructed primarily of stainless steel. Other materials in contact with the coolant are special materials such as hard surfacing and packing.

All RCS valves which are 3 inches and larger, which contain radioactive fluid and which normally operate above 212°F, are provided with either double-packed stuffing boxes and stem intermediate lantern gland leakoff connections or a modified stem-packing arrangement and a capped leakoff line. All throttling control valves, regardless of size, are provided with double-packed stuffing boxes and stem leakoff connections with the exception of the pressurizer spray valves 1(2)PCV-RC06 and 1(2)PCV-RC07. These pressurizer spray valves have been modified to reduce the packing depth and apply live loading. All leakoff connections are piped to the PRT. Leakage to the Containment is essentially zero for these valves.

5.4.12.1 Reactor Coolant Loop Stop Valves

5.4.12.1.1 Design Description

The reactor coolant loop stop valves, shown on Figure 5.4-12, are remotely-controlled motor-operated gate valves which permit any loop to be isolated from the reactor vessel. One valve is installed on each hot leg and one on each cold leg. Coolant is circulated in an isolated loop through a bypass line which contains a remotely-controlled motor-operated stop valve. This bypass valve is closed during normal loop operation. To protect the RCP, a valve-pump interlock circuit prevents the starting of the RCP in a given loop being unisolated unless the cold leg (discharge) valve is closed and the bypass valve is open. The interlock also prevents pump operation if the bypass valve and either of the stop valves are closed.

To ensure against an accidental startup of an unborated and/or cold isolated loop, an additional valve interlock system is provided which meets the IEEE "Criteria for Nuclear Power Plant Protection Systems" (No. 279, August 1968). Reference to the RCS on Figure 5.1-1 indicates a relief line and bypass around the cold leg stop valve. The additional interlocks are for the purpose of ensuring that flow from the isolated loop to the remainder of the RCS takes place through the relief line stop valve (after system pressure is equalized through the loop drain header and the hot leg stop valve is opened) for a period of 105 minutes before the cold leg loop stop valve is opened.

The flow through the relief line is low (approximately 200 to 300 gpm) so that the temperature and boron concentration are brought to equilibrium with the remainder of the system at a relatively slow rate. The valve temperature relief line flow interlock:

1. Prevents opening of a hot leg loop stop valve unless the cold leg loop stop valve is closed;

2. Prevents starting a RCP unless:
   a. The cold leg loop stop valve is closed and the bypass valve is open, or
   b. Both the hot leg loop stop valve and cold leg loop stop valve are open; and

3. Prevents opening of a cold leg loop stop valve unless:

a. The hot leg loop stop valve has been opened a specified time,
b. The loop bypass valve has been opened a specified time,
c. Flow has existed through the relief line for a specified time, or
d. The cold leg temperature is within 10°F of the highest cold leg temperature, and the hot leg temperature is within 10°F of the highest hot leg temperature.

The parameters of each reactor coolant loop stop valve are shown in Table 5.4-24.

5.4.12.1.2 Design Evaluation

The primary loop isolation valves were designed and analyzed as ASME Section III Class A vessels including an experimental stress analysis per the guidelines of Article I-10. No emergency or faulted condition categories were defined or recognized by the design code identification Table 5.2-1.

The primary pressure boundary, such as the body and bonnet, are cast from CF8M steel having an allowable design stress intensity value (S_m) of 18,700 psi at 650°F. The other pressure-containing material within the valve is the disk with an allowable S_m of 16,700 psi also at 650°F.

Calculations confirm that the subject valve will withstand the loadings of the RCS piping. These calculations were also verified by a strain gage hydrostatic test on the body, bonnet, and bolting members of a typical valve.

The subject valves were subjected to cold and hot operational cycle tests by the manufacturer. All areas were re-inspected after the hot operational tests, and the results were recorded.

5.4.13 Safety and Relief Valves

5.4.13.1 Pressurizer Safety Valves

The pressurizer safety valves are totally enclosed pop-type valves. The valves are spring-loaded, self-activated and, with back-pressure compensation, designed to prevent system pressure from exceeding the design pressure by more than 110% in accordance with the ASME B&PV Code, Section III. The set pressure of the valves is 2485 psig.

A water seal is maintained below each safety valve seat to minimize leakage. The six-inch pipes connecting the pressurizer nozzles to their respective code safety valves are shaped in the form of a loop seal. Condensate, as a result of normal, ambient heat losses, will accumulate in the loop, thus flooding the valve seat. The water will prevent any leakage of hydrogen gas or steam through the safety valve seats. If the pressurizer pressure exceeds the set pressure of the safety valves, they will start lifting, and the water from the seal will discharge during the accumulation period. A temperature indicator in the safety valve discharge manifold alerts the operator to the passage of steam due to leakage or valves lifting.

A second method used to determine if a safety valve is lifting is the Babcock & Wilcox Company Valve Monitoring System (VMS). It is an acoustic-based system which monitors the valve and informs the operator whether the valve is opened or closed. The VMS utilizes accelerometers mounted on the valve to detect the noise caused by flow through the valve. This noise signal is conditioned and applied to an alarm monitor, indicator, and audio monitor. The system can distinguish between normal background noise (as when the valve is closed) and the much higher level when the valve is open.

The pressurizer safety valve piping system has been designed to accommodate the forces resulting from the motion of the water slug in the loop seal after sudden opening of the safety valve. The calculated impact load is 15 kips, which is less than the design value of 21 kips for the component support.

Design parameters for the pressurizer safety valves are given in Table 5.4-23.

5.4.13.2 Power-Operated Relief Valves

The pressurizer is equipped with PORVs which limit system pressure for a large power mismatch and thus prevent actuation of the fixed high-pressure reactor trip. The PORVs are operated automatically or by remote manual control. The operation of these valves also limits the undesirable opening of the spring-loaded safety valves. Remotely operated stop valves are provided to isolate the PORVs if excessive leakage occurs. A temperature alarm in the PORV relief line alerts the operator to passage of steam due to leakage or valves opening. Valve position is provided by stem-mounted position indicators.

The PORVs are designed to limit the pressurizer pressure to a value below the high-pressure trip setpoint for all design transients up to and including the design percent step load decrease with steam dump but without reactor trip.

Design parameters for the PORVs are given in Table 5.4-23.

5.4.14 Component Supports

The criteria applied in the design of the principal RCS component supports (i.e., supports, restraints, snubbers, and guides for vessels, piping, pumps and valves), including the design codes or standards applied to each type of support, are shown in Table 5.4-25.

All support steel consists of high-strength, low-alloy structural steel (ASTM A-588). Snubber-support connections made from quenched and tempered alloy steel (ASTM A-514) were employed.

The material specification is modeled after Section III, ASME Nuclear Vessel Code. Requirements include Charpy impact testing, ultrasonic testing, through-thickness tension testing, and traceability of all material. All welds received either radiographic (where possible), ultrasonic, or mag-particle testing.

See Figures 5.4-13 to 5.4-24 for support details.

5.4.14.1 Steam Generator Supports

Each steam generator is supported on a structural system consisting of four vertical support columns and upper and lower lateral restraints approximately 28 feet apart. The vertical columns have a universal pinned connection at each end to accommodate both the radial growth of the steam generator itself and the radial movement of the vessel from the reactor center.

The lower lateral support consists of an inner frame, keyed and shimmed into the four support feet to accommodate radial growth. The inner frame is surrounded by an outer frame which is embedded in both the reactor shield and crane wall concrete. The connection between the inner and outer frame consists of a series of shimmed points which act as both guides and limit stops to allow for expansion from the center of the reactor. The lower lateral support restrains both torsional and translational movements.

The upper lateral support consists of a ring band which is shimmed to the steam generator at twelve locations around the circumference. Attached to this band are lugs which are shimmed and guided to the structural framing system and are embedded directly in the operating level floor slab. Four hydraulic snubbers connect the lugs and the embedded frame in a direction coincident with the direction of movement away from the reactor center. The upper lateral support restrains rapid translational movements in all directions. When the Unit is in the cold shutdown condition (i.e., RCS $\leq$ 200°F and $\leq$ 275 psig) it is permissible to remove all four hydraulic snubbers concurrently (see Reference 8).

Potential for low fracture toughness and lamellar tearing of steam generator supports is minimal according to NUREG 0577, Part I, Section 3.

5.4.14.2 Reactor Vessel Supports

The reactor vessel is supported from four of eight nozzles by four individual weldments embedded in the reactor shield concrete. Each nozzle pad bears on a shoe, supported by a heavy U-shaped wide flange which wraps around the shoe. The U-shaped wide flange is water-cooled at the junction of the outer flange and the web by two continuous welded angles on either side of the web. On Unit 1, one of the lines (1CC188-1") to a reactor support pad has been cut and capped after it was found leaking during the performance of the 10-year inservice hydrostatic testing of the Component Cooling System. The U-shaped wide flange bears vertically on two shims and is restrained horizontally by a series of shims and bearing plates. These bearing plates and shims are connected to an outer weldment which completely surrounds the wide flange and is embedded in the concrete. The reactor support system allows the reactor to expand radially over the supports but resists translational and torsional movement by the combined tangential restraining action of each nozzle support.

5.4.14.3 Pressurizer Support

The pressurizer is supported on a ring girder which is in turn supported vertically by four wide flanges. Horizontally, the vessel is restrained at two elevations approximately 20 feet apart. The lower restraint consists of a weldment attached directly to the ring girder and is embedded in the crane wall. The upper restraint consists of four individual weldments embedded in concrete that allow the pressurizer to expand radially but resist torsional and translational movements.

5.4.14.4 Reactor Coolant Pump Support

The RCP is supported vertically by three universally pin-ended columns which rest on a heavy triangular steel platform. This structural column system resists both overturning and vertical movement while allowing for expansion from the center of the reactor. Translation movement is resisted by a combination of pin-connected tie rods and struts which are slotted at one end to allow for expansion movements. The potential for low fracture toughness and lamellar tearing of RCP supports is minimal according to NUREG 0577, Part I, Section 3.

5.4.15 Reactor Vessel Head Vent System

A Reactor Vessel Head Vent System (RVHVS) is provided to exhaust noncondensable gases from the reactor vessel that could inhibit natural circulation core cooling. The RVHVS consists of two parallel paths, each containing two solenoid-operated valves in series. These valves are operated from the Control Room and fail in the closed position. Temperature detectors, downstream of the solenoid valves, alarm on high temperature to provide indication of system actuation. The RVHVS piping and valves are designed in accordance with the parameters outlined in Table 5.1-2. The RVHVS valves are periodically tested as described in UFSAR Section 16.3. An associated safety-related system, the Reactor Vessel Level Instrumentation System (RVLIS), is discussed in Section 7.5.

5.4.16 References, Section 5.4

1. Ernest L. Robinson, "Bursting Tests of Steam-Turbine Disk Wheels," Transactions of the A.S.M.E., July 1944.

2. Application of the Griffith-Irwin Theory of Crack Propagation to the Bursting Behavior of Disks, Including Analytical and Experimental Studies by D.H. Winne and B.M. Wundt, ASME, December 1, 1957.

3. W.E. Berry and F.W. Fink, "The Corrosion of Inconel in High Temperature Water," Batelle Memorial Institute, April 1958.

4. W.J. O'Donnell, A Study of Perforated Plates with Square Penetration Patterns, Welding Research Council Bulletin No. 124, September 1967.

5. J.B. Mahoney and V.L. Salerno, Stress Analysis of a Circular Plate Containing a Rectangular Array of Holes, Welding Research Council Bulletin No. 106, July 1965.

6. M.M. Lemcoe, Feasibility Studies of Stresses in Ligaments, Welding Research Council Bulletin No. 65, November 1960.

7. O'Donnell, W.J. and C.M. Purdy, "The Fatigue Strength of Members Containing Cracks," ASME Transactions, Journal of Engineering for Industry, Vol. 86-B, 1964, pp. 205-213.

8. Belair, R.J., "Zion Units 1 and 2 Safety Evaluation Report: Cold Shutdown Seismic Analysis of RCL with All Steam Generator Snubbers Removed," CECo Nuclear Engineering Department, October 28, 1992.

TABLE 5.4-1

REACTOR COOLANT PUMPS DESIGN DATA

Number of Pumps	4
Design Pressure/Operating Pressure, psig	2485/2235
Hydrostatic Test Pressure (cold), psig	3107
Design Temperature (casing), °F	650
RPM at Nameplate Rating	1190
Suction Temperature, °F	539
Developed Head, ft	267
Capacity, gpm	87,500
Seal Water Injection, gpm	8
Seal Water Return, gpm	3
Pump Discharge Nozzle ID, in.	$27^1/_2$
Pump Suction Nozzle ID, in.	31
Overall Unit Height, ft-in.	25-5.05
Water Volume, ft^3	56
Pump-Motor Moment of Inertia, $lb\text{-}ft^2$	82,000
Motor Data:	
Type	AC Induction Single Speed, Air Cooled
Voltage	4000
Insulation Class	B Thermalastic Epoxy
Phase	3
Frequency, cps	60
Starting Current, amp	4800
Input (hot reactor coolant), kW	4870
Input (cold reactor coolant), kW	6310
Power, HP (nameplate)	6000
Pump Weight, lb. (dry)	169,200

TABLE 5.4-2 (1 of 2)

STEAM GENERATOR DESIGN DATA*

Number of Steam generators	4
Design Pressure, Reactor Coolant/Steam, psig	2485/1085
Reactor Coolant Hydrostatic Test Pressure (tube side-cold), psig	3107
Design temperature, Reactor Coolant/Steam, °F	650/600
Reactor Coolant Flow, lb/hr	33.8×10^6
Total Heat Transfer Surface Area, ft^2	51,500
Heat Transferred, Btu/hr	2772×10^6
Steam Conditions at Full Load, Outlet Nozzle:	
Steam Flow, lb/hr	3.5×10^6
Steam Temperature, °F	506.3
Steam Pressure, psig	705
Maximum Moisture Carryover, wt %	0.25
Feedwater, °F	428.6
Overall Height, ft-in.	67'8"
Shell OD, upper/lower, in.	$175^3/_4$ / 135
Number of U-tubes	3388
U-tube outer Diameter, in.	0.875
Tube Wall Thickness, (minimum), in.	0.050
Number of Manways/ID in.	4/16
Number of Handholes/ID, in.	2/6

TABLE 5.4-2 (2 of 2)

STEAM GENERATOR DESIGN DATA*

	Rated Load	No Load
Reactor Coolant Water Volume,* ft^3	1080	1080
Primary Side Fluid Heat Content, Btu	28.5×10^6	27.7×10^6
Secondary Side Water Volume, ft^3	1838	3524
Secondary Side Steam Volume, ft^3	4030	2344
Secondary Side Fluid Heat Content, Btu	5.627×10^6	9.628×10^7

* Quantities are for each steam generator.

TABLE 5.4-3

STRESSES DUE TO MAXIMUM STEAM GENERATOR TUBE SHEET PRESSURE DIFFERENTIAL (2485 PSIG)

Stress	(660°F) Computed Value	Allowable Value
Primary Membrane Stress	24,356 psi	37,000 psi (.9 S_y)
Primary Membrane plus Primary Bending Stress	54,946 psi	55,600 psi (1.35 S_y)

In addition to the foregoing evaluation, elasto-plastic limit analysis of the tube sheet-head-shell combination indicates a limit pressure of 3050 psi at operating conditions, giving a safety factor of 1.23 for the abnormal condition.

TABLE 5.4-4

RATIO OF ALLOWABLE STRESSES TO COMPUTED STRESSES
FOR A STEAM GENERATOR TUBE
SHEET PRESSURE DIFFERENTIAL OF 2485 PSIG

Component Part	Stress Ratio
Channel Head	1.34
Channel Head-Tube Sheet Joint	1.80
Tubes	1.20
Tube Sheet	
Max. Avg. Ligament	1.01
Effective Ligament	1.52

TABLE 5.4-5

PRIMARY-SECONDARY BOUNDARY COMPONENTS

CONDITION: 100% LOAD OPERATION - 2485/885 psi 650/600°F
Normal Operation Stress Limits

LOC	Description	Inside Limit Center Limit Outer Limit	Stress Limit Center Limit Stress Limit	Inside Surface Stress Center Surface Stress Outer Surface Stress
	JCT OF SHORT	3 S_m	80,100	-10,063 psi
7	CYL WITH	S_m	26,700	+ 8,597 psi
	TUBESHEET	3 S_m	80,100	+27,247 psi
	1/2 THROUGH	3 S_m	80,100	+ 9.514 psi
8	SHORT CYL	S_m	26,700	+ 8,597 psi
	DISCONTINUITY	3 S_m	80,100	+ 7,670 psi
	JCT OF SHORT	3 S_m	80,100	+10,740 psi
9	CYL WITH	S_m	26,700	+ 8,597 psi
	SHELL	3 S_m	80,100	6,443 psi
		3 S_m	80,100	+10,269 psi
10	ON SHELL	S_m	26,700	+ 8,597 psi
		3 S_m	80,100	+ 6,912 psi
		3 S_m	80,100	+ 9,746 psi
11	ON SHELL	S_m	26,700	+ 8,597 psi
		3 S_m	80,100	+ 7,435 psi
	JCT OF PRI	3 S_m	80,100	+58,701 psi
12	SHORT CYL WITH	S_m	26,700	+14,528 psi
	TUBE PLATE	3 S_m	80,100	-29,646 psi
	1/2 THROUGH	3 S_m	80,100	+50,836 psi
13	PRIM SHORT	S_m	26,700	+14,528 psi
	CYL DISCON.	3 S_m	80,100	-21,781 psi
	JCT OF PRI	3 S_m	52,200	+42,286 psi
14	SHORT CYL	S_m	19,400	+14,528 psi
	WITH HEAD	3 S_m	52,200	-13,231 psi

TABLE 5.4-6

PRIMARY-SECONDARY BOUNDARY COMPONENTS

CONDITION: PRIMARY HYDROTEST - 3107/0 psi

LOC	Description	Code Limit	Primary Membrane Stress Limit	Axial Primary Membrane Stress Intensity
7	JCT OF SHORT CYL WITH TUBESHEET	.9 S_y	45,000	0 psi
8	1/2 THROUGH SHORT CYL DISCONTINUITY	.9 S_y	45,000	0 psi
9	JCT OF SHORT CYL WITH SHELL	.9 S_y	45,000	0 psi
10	ON SHELL	.9 S_y	45,000	0 psi
11	ON SHELL	.9 S_y	45,000	0 psi
12	JCT OF PRI SHORT CYL WITH TUBE PLATE	.9 S_y	45,000	18,158 psi
13	1/2 THROUGH PRIM SHORT CYL DISCON.	.9 S_y	45,000	18,158 psi
14	JCT OF PRI SHORT CYL WITH HEAD	.9 S_y	36,000	18,158 psi

TABLE 5.4-7

PRIMARY-SECONDARY BOUNDARY COMPONENTS

CONDITION: SECONDARY CHAMBER HYDROTEST - 0/1356 psi

LOC	Description	Code Limit	Primary Membrane Stress Limit	Axial Primary Membrane Stress Intensity
7	JCT OF SHORT CYL WITH TUBESHEET	.9 S_y	45,000	13,169 psi
8	1/2 THROUGH SHORT CYL DISCONTINUITY	.9 S_y	45,000	13,169 psi
9	JCT OF SHORT CYL WITH SHELL	.9 S_y	45,000	13,169 psi
10	ON SHELL	.9 S_y	45,000	13,169 psi
11	ON SHELL	.9 S_y	45,000	13,169 psi
12	JCT OF PRI SHORT CYL WITH TUBE PLATE	.9 S_y	36,000	0 psi
13	1/2 THROUGH PRIM SHORT CYL DISCON.	.9 S_y	36,000	0 psi
14	JCT OF PRI SHORT CYL WITH HEAD	.9 S_y	36,000	0 psi

TABLE 5.4-8

PRIMARY-SECONDARY BOUNDARY COMPONENTS

CONDITION: LOSS OF SECONDARY PRESSURE (STEAMLINE BREAK) - FAULTED CONDITION
2485/0 psi 660°F

LOC	Description	Primary Membrane Stress Emergency Condition Limits Code Limit	Stress	Primary Membrane Stress
7	JCT OF SHORT CYL WITH TUBESHEET	S_y	41,112	0 psi
8	1/2 THROUGH SHORT CYL DISCONTINUITY	S_y	41,112	0 psi
9	JCT OF SHORT CYL WITH SHELL	S_y	41,112	0 psi
10	ON SHELL	S_y	41,112	0 psi
11	ON SHELL	S_y	41,112	0 psi
12	JCT OF PRI SHORT CYL WITH TUBE PLATE	S_y	41,112	14,528 psi
13	1/2 THROUGH PRIM SHORT CYL DISCON.	S_y	41,112	14,528 psi
14	JCT OF PRI SHORT CYL WITH HEAD	S_y	29,000	14,528 psi

* Complete Tubesheet Structure Complex also evaluated on Limit Analysis Basis

TABLE 5.4-9

51,500 SQ. FT. STEAM GENERATOR USAGE FACTORS (INDIVIDUAL TRANSIENTS)
PRIMARY AND SECONDARY BOUNDARY COMPONENTS

			JUNCTION OF HEAD & TUBESHEET & DIVIDER PLATE																		UNPERFORATED				JUNCTION OF		
			IN TUBESHEET						ON TUBESHEET FACE						IN HEAD						OUTER RING				SHELL TO TUBE		
			INLET			OUTLET			INLET			OUTLET			INLET			OUTLET			PRIM.		SEC.		SHEET		
No.	**Transient**	**No. of Cycles**	**SASR***	**SASH**	**SRSH**	**SASR**	**SASH**	**SRSH**	**SASR**	**SASH**	**SRSH**	**SASR**	**SASH**	**SRSH**	**SASR**	**SASH**	**SRSH**	**SASR**	**SASH**	**SRSH**	**INLET**	**OUTLET**	**INLET**	**OUTLET**	**HAHH**	**HAHR**	**HHHR**
1	Heatup-Cooldown	200	.008	.01	0	.008	.01	0	0	0	0	0	0	0	.009	.003	0	.009	.003	0	0	0	0	0	0	0	0
2	Loading-Unloading	18,300	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	.019	.082	.056	0
3	Small Step Increase	2,000	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
4	Small Step Decrease	2,000	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
5	Large Step Decrease	200	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	.002	.001	0
6	Loss of Load	80	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	.001	.001	0	0	0	.001	.002	.006	.005	0
7	Loss of Power	40	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
8	Loss of Flow	80	.008	.009	0	.008	.016	0	0	0	0	0	0	0	.011	.003	.001	.012	.006	0	0	0	.002	.004	.002	.002	0
9	Reactor Trip	400	0	0	0	0	.001	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	.001	.005	.012	.008	0
10	React. Cool. Pipe Break	1	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
11	Steamline Break	1	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	.001	0
12	Primary Hydrotest	5	.004	.007	0	.004	.007	0	0	0	0	0	0	0	.005	.002	0	.005	.002	0	0	0	.001	.001	.001	.001	0
13	Secondary Hydrotest	5	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	.001	.002	0
14	Turbine Roll Test	10	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0

* Principal Stress Difference Codes

TABLE 5.4-10

51,500 SQ. FT. STEAM GENERATOR USAGE FACTORS (INDIVIDUAL TRANSIENT)
CENTER OF TUBESHEET

No.	Transient	No. of Cycles	Primary Inlet Angle							Primary Outlet Angle							Secondary Inlet Angle							Secondary Outlet Angle						
			0°	15°	30°	45°	60°	75°	90°	0°	15°	30°	45°	60°	75°	90°	0°	15°	30°	45°	60°	75°	90°	0°	15°	30°	45°	60°	75°	90°
1	Heatup-Cooldown	200	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
2	Loading-Unloading	18,300	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
3	Small Step Increase	2,000	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
4	Small Step Decrease	2,000	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
5	Large Step Decrease	200	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
6	Loss of Load	80	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
7	Loss of Power	40	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
8	Loss of Flow	80	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
9	Reactor Trip	400	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
10	React. Cool. Pipe Break	1	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
11	Steamline Break	1	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
12	Primary Hydrotest	5	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
13	Secondary Hydro	5	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
14	Turbine Roll Test	10	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0

* Angular location around perforation

TABLE 5.4-11

TUBE SHEET STRESS ANALYSIS RESULTS
FOR 51,500 SQ. FT. STEAM GENERATORS

Conditions		Maximum Primary Membrane Plus Primary Bending Average Ligament Stress		Maximum Effective Ligament Membrane Stress
100% Normal Operation	2485/885 psi 650/600°F	33,979 psi	(40,050)[1]	15,853 psi (26,700)[2]
Primary Hydrotest	3107/0 psi 100°F	67,300 psi	(67,500)[3]	30,365 psi (45,000)[4]
Secondary Hydrotest	0/1356 psi 100°F	29,811 psi	(67,500)[3]	13,159 psi (45,000)[4]
Steamline Break (Fault Condition)	2485/0 psi 660°F	56,785 psi	(Limit)[5]	24,356 psi (Limit)[5]

Parenthesis Indicate Code Allowable Stress

1 $1.5\ S_m$

2 $1.0\ S_m$

3 $1.35\ S_y$

4 $.9\ S_y$

5 Limit Analysis Results Apply

TABLE 5.4-12

LIMIT ANALYSIS CALCULATION RESULTS
TABLE OF STRAINS, LIMIT PRESSURES, AND FATIGUE EVALUATIONS FOR 51,500 SQ. FT. STEAM GENERATORS

Case	Location	Meridional Strain, In/In	Circumferential Strain, In/In	Peak Stress Intensity, Psi	Allowable Number of Cycles, N_1	Number of Cycles, N_2	Usage Factor N_2/N_1	Limit Pressure Psi
Hot	Channel/Primary Shell	.0188	-.000559	508,000	46	10	.22	
2500/0 PSI	Tubesheet/Secondary Shell	-.00193	.00602	83,700	5,000	10	.0020	3,158
650°F	Tubesheet Center	.00159	.00159	77,400	6,600	10	.0015	
Cold Hydro.	Tubesheet/Primary Shell	.0145	-.000537	434,000	80	5	.053	
3105/0 PSI	Tubesheet/Secondary Shell	-.00220	.000684	106,000	3,500	5	.0014	3,887
70°F	Tubesheet Center	.00177	.00177	95,400	5,000	5	.0010	
Cold Hydro With	Tubesheet/Primary Shell	.00730	-.000348	218,000	500	5	.010	
Secondary Pressure	Tubesheet/Secondary Shell	-.000962	.000560	50,700	40,000	5	.0001	4,401
3105/700 PSI	Tubesheet Center	.00147	.00147	79,000	8,000	5	.0005	
70°F								
Hot Hydro	Tubesheet/Primary Shell	.00777	-.000407	222,000	400	50	.13	
2485/0 PSI	Tubesheet/Secondary Shell	-.00176	.000551	80,900	7,000	50	.0071	3,354
400°F	Tubesheet Center	.00148	.00148	76,300	8,500	50	.0059	

TABLE 5.4-13

REACTOR COOLANT PIPING DESIGN PARAMETERS

Reactor inlet piping, ID, in.	27 1/2
Reactor inlet piping, nominal thickness, in.	2.38
Reactor outlet piping, ID, in.	29
Reactor outlet piping, nominal thickness, in.	2.50
Coolant pump suction piping, ID, in.	31
Coolant pump suction piping, nominal thickness, in.	2.66
Pressurizer surge line piping, ID, in.	11.188
Pressurizer surge line piping, nominal thickness, in.	1.406
Design/operating pressure, psig	2485/2235
Hydrostatic test pressure, (Cold) psig	3107
Design temperature, °F	650
Design temperature (pressurizer surge line) °F	680
Water volume, (all 4 loops including surge line (ft^3)	1545
Design pressure, pressurizer relief line, psig	From pressurizer to safety valve - 2485 psig, 650°F
Design temperature, pressurizer relief lines, °F	From safety valve to pressurizer relief tank - 600 psig, 600°F

TABLE 5.4-14

OPERATING CONDITIONS AND STRESS LIMITS PRESSURE PIPING (INCLUDING RCS PIPING)

Design Conditions	Loading Combination	Stress Limit
Normal	$P_1=P_m$ (Design Pressure)+P_b (Weight)	$\leq S_m$
Upset	$P_2=P_m$ (Design Pressure)+P_b (Weight + OBE)	$\leq 1.5\ S_m$
Emergency	$P_3=P_m$ (Design Pressure)+P_b (Weight + DBE)	$\leq 2.25\ S_m$ or $1.2S_h$
Faulted Case		
- Primary Stress	$P'_4=P_m$ (Design Pressure) + P_b (Weight + DBE) + P_L (Gross Local Stress)	$\leq 2S_h$
- Secondary Stress	$P''_4=P_m$ (Design Pressure) + P_b (Weight + DBE) + P_L (Gross Load Stress) + P_e (Thermal) + Q (Differential Anchor Movement)	$\leq S_u$ or $3S_m$

where:

S_h = allowable stress from ANSI B31.1.0 - 1967
S_u = allowable ultimate strength of piping material corresponding to the temperature for the operating condition
S_m = allowable stress from ANSI B31.7.0 - 1969, Nuclear Piping Code
P_m = primary general membrane stress
P_L = primary local membrane stress
P_b = primary flexural stress
P_e = secondary stress
Q = secondary membrane plus bending stress

TABLE 5.4-15

RCS PIPING STRESSES

STRESS LIMIT (FSAR):

P_M (or P_L) + P_B < S

where:

P_M = primary general membrane stress
P_B = primary bending stress
P_L = primary local membrane stress
S = 15,000 psi = allowable stress per code case N-10

RESULTS:

LOCATION STRESS (psi)	HOT LEG	CROSSOVER LEG	COLD LEG
DEADWEIGHT	523	139	723
PRESSURE	7,650	7,680	7,624
TOTAL	8,173	7,819	8,347
ALLOWABLE	15,000	15,000	15,000

TABLE 5.4-16

RCS PIPING STRESSES

STRESS LIMIT (USAS-B31.1.0 - 1967):

$$S_E < S_A = f(1.25\ S_C + 0.25\ (S_H))$$

where:

S_E = thermal expansion stress
S_A = allowable stress range
S_C = 17,500 psi = allowable stress (cold)
S_H = 15,000 psi = allowable stress (hot)
f = 1.0

RESULTS:

LOCATION STRESS (PSI)	HOT LEG	CROSSOVER LEG	COLD LEG
S_E	15,235	3,901	7,376
S_A	27,332	27,420	27,288

TABLE 5.4-17

RCS PIPING STRESSES

STRESS LIMIT (FSAR):

$P_M + P_B < 1.2S$

where:

P_M = primary general membrane stress
P_B = primary bending stress
S = 15,000 psi = allowable stress

RESULTS:

LOCATION STRESS (psi)	HOT LEG	CROSSOVER LEG	COLD LEG
DEADWEIGHT	523	139	723
PRESSURE	7,650	7,680	7,624
SEISMIC (OBE)	2,709	942	1,092
TOTAL	10,882	8,811	9,439
ALLOWABLE	18,000	18,000	18,000

TABLE 5.4-18

RCS PIPING STRESSES

STRESS LIMIT (FSAR):

DESIGN LIMIT CURVES

BREAK LOCATION STRESS (psi)	HOT LEG	CROSSOVER LEG	COLD LEG
NORMAL + SEISMIC (DBE) + BLOWDOWN	43,137	31,664	32,139
ALLOWABLE*	44,000	44,000	44,000

* From most conservative portion of the design limit curves.

TABLE 5.4-19

RESIDUAL HEAT REMOVAL SYSTEM CODE REQUIREMENTS

Residual Heat Exchangers (Tube Side)	ASME III, Class C
(Shell Side)	ASME VIII
Residual Heat Removal Piping and Valves	USAS B31.1

TABLE 5.4-20 (1 of 2)

RESIDUAL HEAT REMOVAL SYSTEM DESIGN PARAMETERS

GENERAL

Plant design life, years	40
Component cooling water supply temperature design, °F	95
Reactor coolant temperature at startup of decay heat removal, °F	350
Time to cool Reactor Coolant System from 350°F to 140°F, hr (design basis)	16
Refueling water storage temperature, °F	Ambient
Decay heat generation at 20 hours after shutdown, Btu/hr	68.7×10^6
H_3BO_3 concentration in refueling water storage tanks, ppm boron	2000

RESIDUAL HEAT EXCHANGERS

Number, per unit	2
Design heat transfer Btu/hr	28.0×10^6

	Shell Side	Tube Side
Design pressure, psig	150	600
Design temperature, °F	200	400
Design flow rate, lb/hr	2.475×10^6	1.85×10^6
Design outlet temperature, °F	118.5	122.3
Design inlet temperature, °F	107.1	137.5
Fluid	Component cooling water	Reactor coolant (borated demin-eralized water)
Material of construction	Carbon steel	Austenitic stainless steel

TABLE 5.4-20 (2 of 2)

RESIDUAL HEAT REMOVAL SYSTEM DESIGN PARAMETERS

RESIDUAL HEAT REMOVAL PUMPS

Number (per unit)	2
Type	Vertical centrifugal
Design pressure, psig	600
Design temperature, °F	400
Shutoff head, psi	170
Design flow rate, gpm	3000
Design head, ft	350
Maximum flow, gpm	4500
Available NPSH at maximum flow rate, ft	22
Available NPSH at design flow rate, ft	25
Temperature of pump fluid, °F	40/350
Normal fluid	Reactor coolant
Fluid during LOCA recirculation phase	Radioactive borated water with H_2 and NaOH in solution
Material of construction	Austenitic stainless steel

PIPING AND VALVES

Residual heat removal loop (piping and valves in isolated loop):	
Design pressure, psig	600
Design temperature, °F	400
Residual heat removal loop isolation valves and piping:	
Design pressure, psig	2485
Design temperature, °F	650

TABLE 5.4-21 (1 of 2)

RESIDUAL HEAT REMOVAL SYSTEM MALFUNCTION ANALYSIS

Component	Malfunction	Comments and Consequences
Residual Heat Removal Pumps	- Rupture of a pump casing	The casing and shell are designed for 600 psi and 400°F. The pump is protected from overpressurization by two normally closed valves in the pump suction line, and by a relief valve which discharges back to the pressurizer relief tank. The pump can be inspected and is located in the Auxiliary Building in an area protected against credible missiles. Rupture is considered unlikely, but in any event the pump can be isolated.
	- Pump fails to start	One operating pump furnishes half of the flow required to meet design cooldown rate. This increases the time necessary for plant cooldown.
	- Manual valve on pump suction is closed	This is prevented by prestartup and operational check.
	- Stop valve on discharge line closed or check valve sticks closed	Stop valve is locked open. Prestartup and operational checks confirm position of valves.
Remote operated valves inside containment in pump suction line	- Valve fails to open	In the improbable event that one of the remote-operated valves on the suction line to the residual heat removal pumps is inoperable, an attempt will be made to open it manually. If this is impossible, the plant will be cooled to about 280°F by using steam dumps. The unit will be kept at that temperature for several weeks until decay heat could be matched by the letdown heat exchangers and by feed and bleed. Feed and bleed through the CVCS will be done intermittently to prevent heat transfer through the regenerative heat exchanger. The pressurizer level will be brought to minimum during the bleed operation and to maximum during the feed operation. It is estimated that plant cooldown may be accomplished within a month.

TABLE 5.4-21 (2 of 2)

RESIDUAL HEAT REMOVAL SYSTEM MALFUNCTION ANALYSIS

Component	Malfunction	Comments and Consequences
Remote operated valves inside containment on pump discharge line	- Valve fails to open	Pump discharge pressure gauge shows pump shutoff head indicating no flow. The low head safety injection lines may be opened and utilized to direct flow to the RCS cold legs.
Residual Heat Exchanger	- Tube or shell rupture	Rupture is considered unlikely, but in any event the faulty heat exchanger may be isolated.
Residual Heat Exchanger vent or drain valve	- Left open	This is prevented by prestartup operational checks.

TABLE 5.4-22

PRESSURIZER AND PRESSURIZER RELIEF TANK DESIGN DATA

Pressurizer	
Design/Operating Pressure, psig	2485/2235
Hydrostatic Test Pressure (cold), psig	3107
Design/Operating Temperature, °F	680/653
Water Volume, Full Power, ft^3	624 Actual
Steam Volume, Full Power, ft^3	1176 Actual
Surge Line Nozzle Diameter, in.	14
Shell ID, in	84
Electric Heaters Capacity, kW*	1800
Heatup rate of Pressurizer using Heaters only, °F/hr	55 (approximately)
Maximum spray rate, gpm	800
Pressurizer Relief Tank	
Design pressure, psig	100
Rupture Disk Release Pressure, psig	100
Design temperature, °F	340
Normal water temperature, °F	Containment Ambient (120°F max.)
Total volume, ft^3	1800
Total Rupture Disk Relief Capacity, lb/hr	1.60×10^6

* As result of abandoned heaters, Unit 1 capacity is reduced by 23.08 kW and Unit 2 capacity is reduced by 92.32 kW.

TABLE 5.4-23

PRESSURIZER VALVES DESIGN PARAMETERS

Pressurizer Spray Control Valves	
Number	2
Design pressure, psig	2485
Design temperature, °F	650
Design flow for valves full open, each, gpm	400
Fluid temperature, °F	545
Pressurizer Safety Valves	
Number	3
Maximum relieving capacity, each at 3% accumulation, lb/hr	420,000
Set pressure, psig	2485
Fluid	Saturated steam
Constant backpressure:	
Normal, psig	3-5
Expected during discharge, psig	350
Pressurizer Power-Operated Relief Valves	
Number	2
Design pressure, psig	2485
Design temperature, °F	650
Relieving capacity at 2350 psig, lb/hr	210,000
Fluid	Saturated steam

TABLE 5.4-24

LOOP STOP VALVES

Design/Normal Operating Pressure, psig	2485/2235
Hydrostatic Test Pressure Shop/Loop, psig	3350/3107
Design Temperature, °F	650
Hot Leg Valve Size, Nominal, in.	29
Cold Leg Valve Size, Nominal, in.	27.50
Open/Close Travel Time, sec	210

TABLE 5.4-25

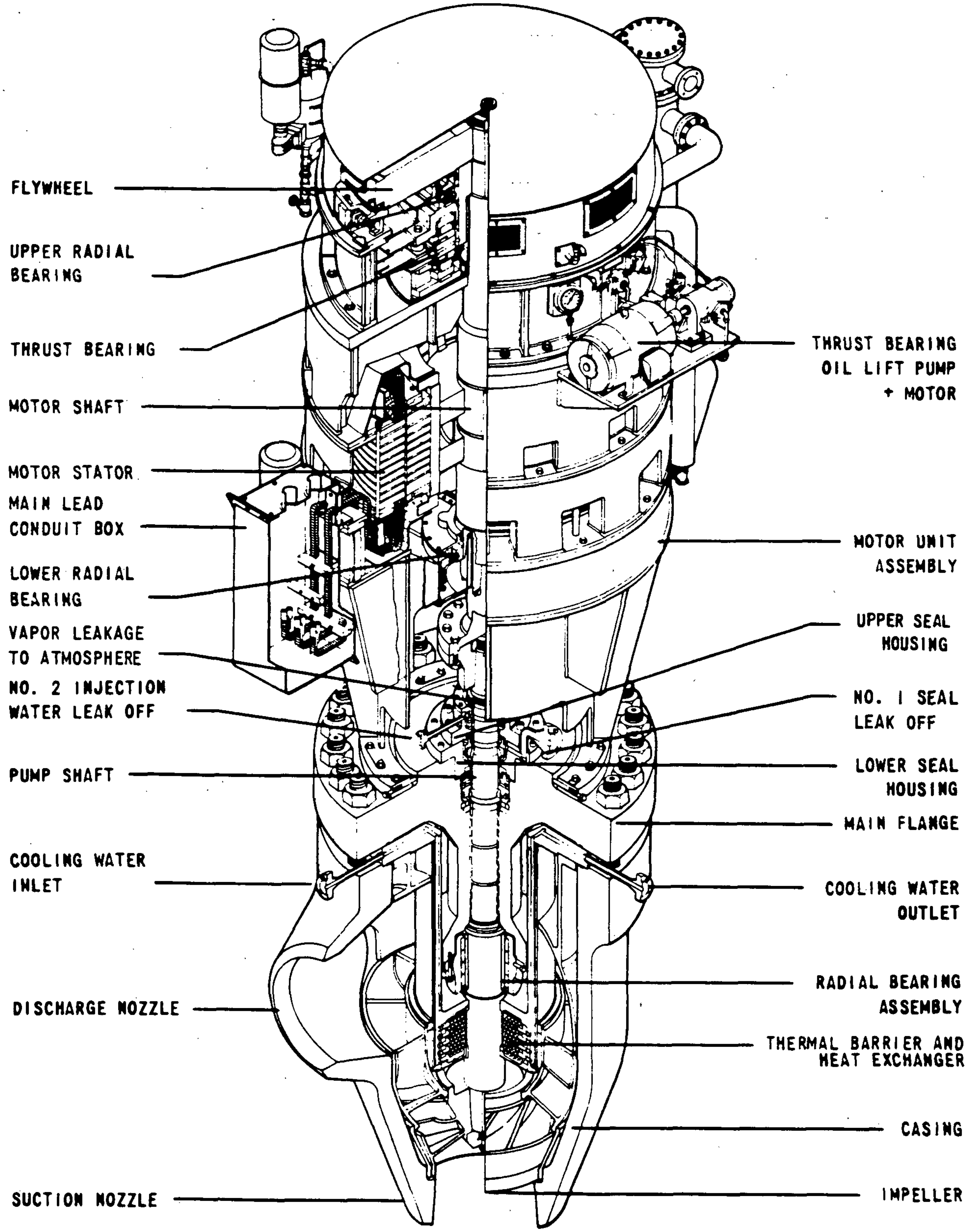
NSSS SUPPORT CRITERIA

LOADING CONDITION	CRITERIA FOR DESIGN OF SUPPORTS
1. Dead Loads + Operating Loads + Thermal Loads	1. Working allowable stresses. ACI-318-63 AISC Manual of Steel Construction (6th edition) 2. Equipment & piping within allowable stresses.
2. 1 + Design Earthquake Forces	Same as 1
3. 1 + Maximum Credible Earthquake Forces	1. Stresses in Supports limited to yield. 2. Equipment & piping within allowable stresses.
4. 1 + Pipe rupture	Same as 3 except strains exceeding yield are allowed in limited, controlled areas of support system to prevent pipe failing. Yielding is allowed only in isolated cases where elastic pipe stress analysis is not a consideration.
5. 3 + Pipe rupture	Same as 4

TABLE 5.4-26

PREOPERATIONAL TEST PROGRAM TRANSIENTS
EVALUATED FOR POSSIBLE VIBRATION PROBLEMS

	Preoperational Test	Specific Transients
1.	RCS Heatup	Operational Tests of Centrifugal Charging Pump (Step Changes) RCP Start Operation of Pressurizer Power-Operated Relief Valves (PORVs) Operation of Pressurizer Spray Valves Operation of Letdown Isolation Valves
2.	RCS at Temperature	Operation of Pressurizer PORVs RCPs (Stopping and Starting)
3.	RCS Cooldown	Initiation of RHR
4.	Reactor Coolant Loop Isolation Valve Tests	Operation of Reactor Coolant Loop Isolation Valves and Bypass line Valves (Open and Close)
5.	Emergency Core Cooling Full Flow Test	Initiation and Termination of the Following: A. Safety Injection (SI Pumps) B. Boron Injection (Centrifugal Charging Pumps with Primary Water) C. Safety Injection (RHR Pumps)
6.	CVCS Test	Operational Tests of Positive Displacement Charging Pump (Stop and Start)



ZION STATION UFSAR

Figure 5.4-1

REACTOR COOLANT PUMP

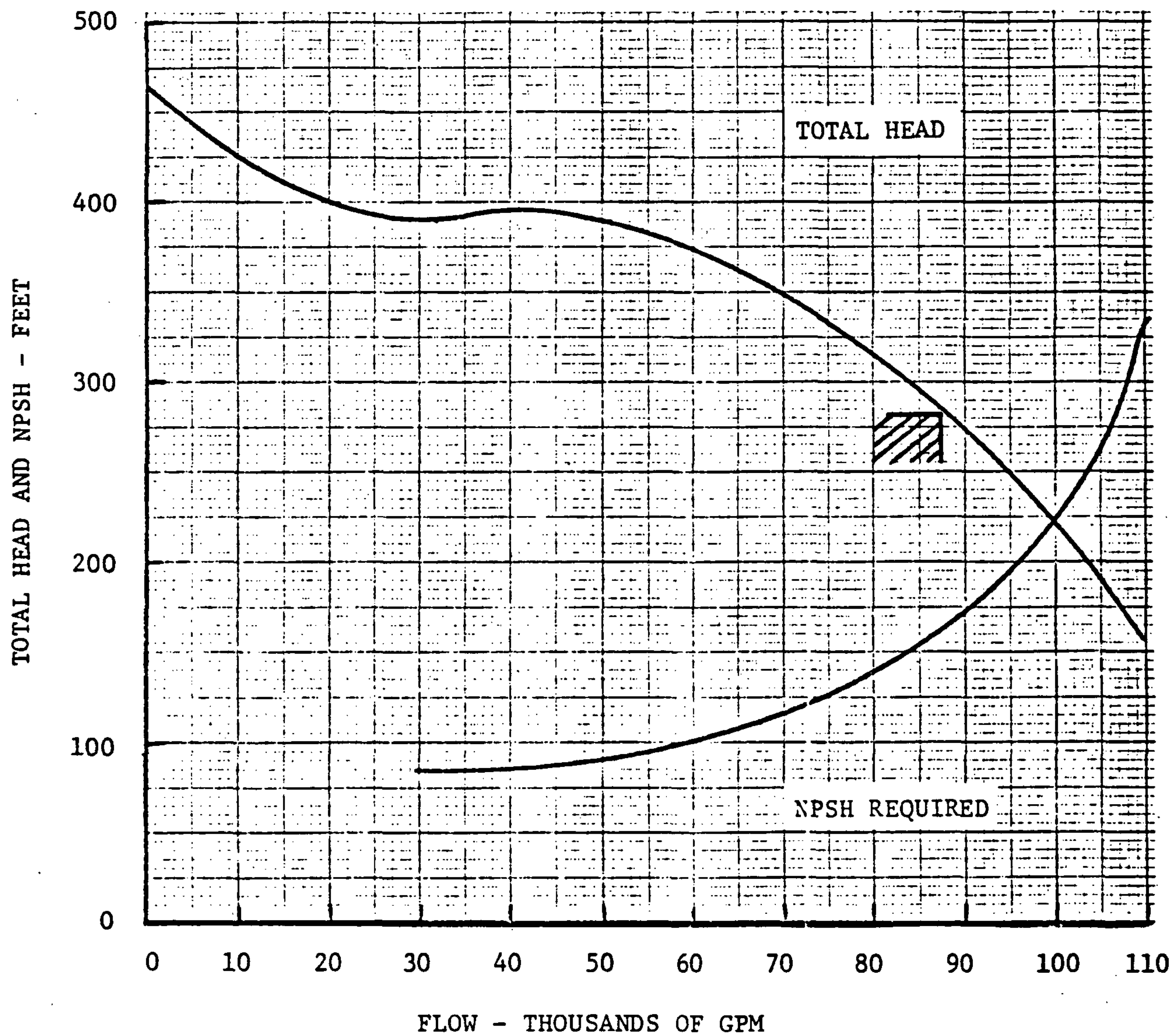


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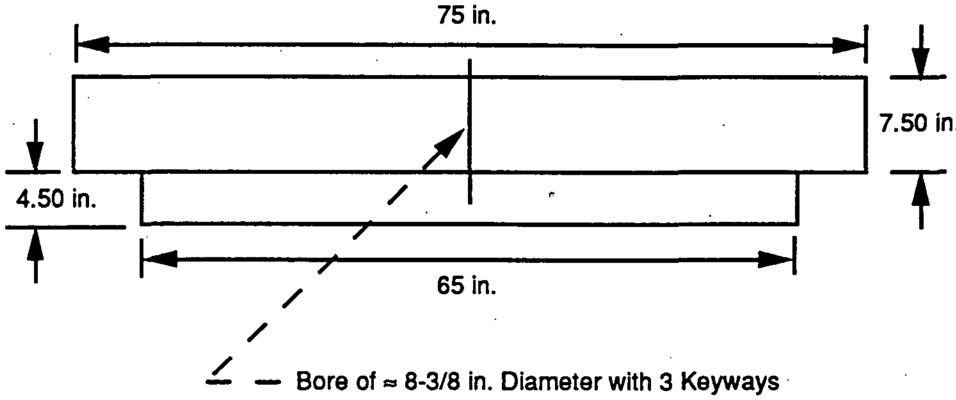
Figure 5.4-2

REACTOR COOLANT PUMP PERFORMANCE CHARACTERISTICS



NOTE:
The plates are bolted together with the bolts aligned perpendicular to the planes of the plates.

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Figure 5.4-3

REACTOR COOLANT PUMP FLYWHEEL

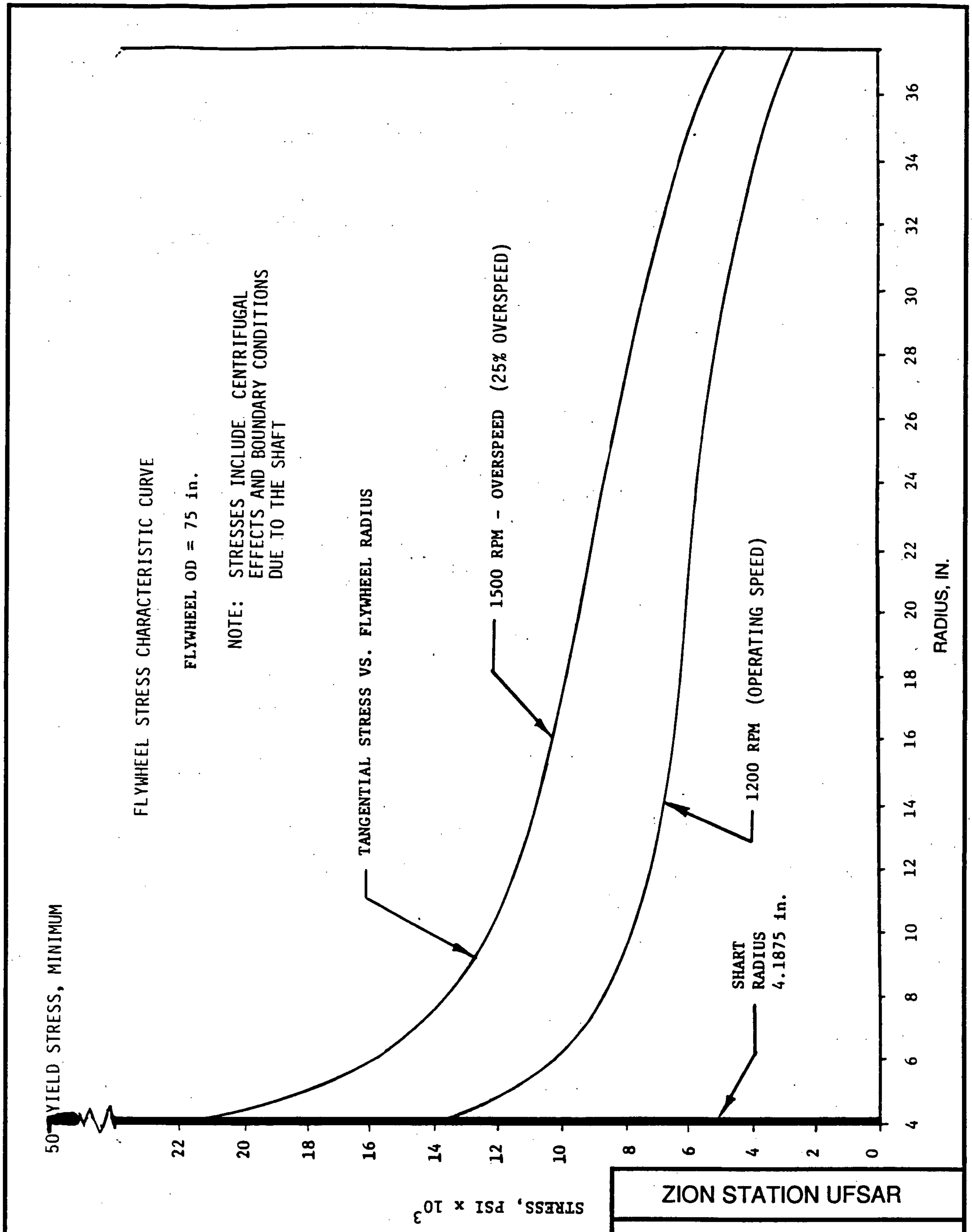

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Figure 5.4-4

FLYWHEEL STRESS CHARACTERISTIC CURVE

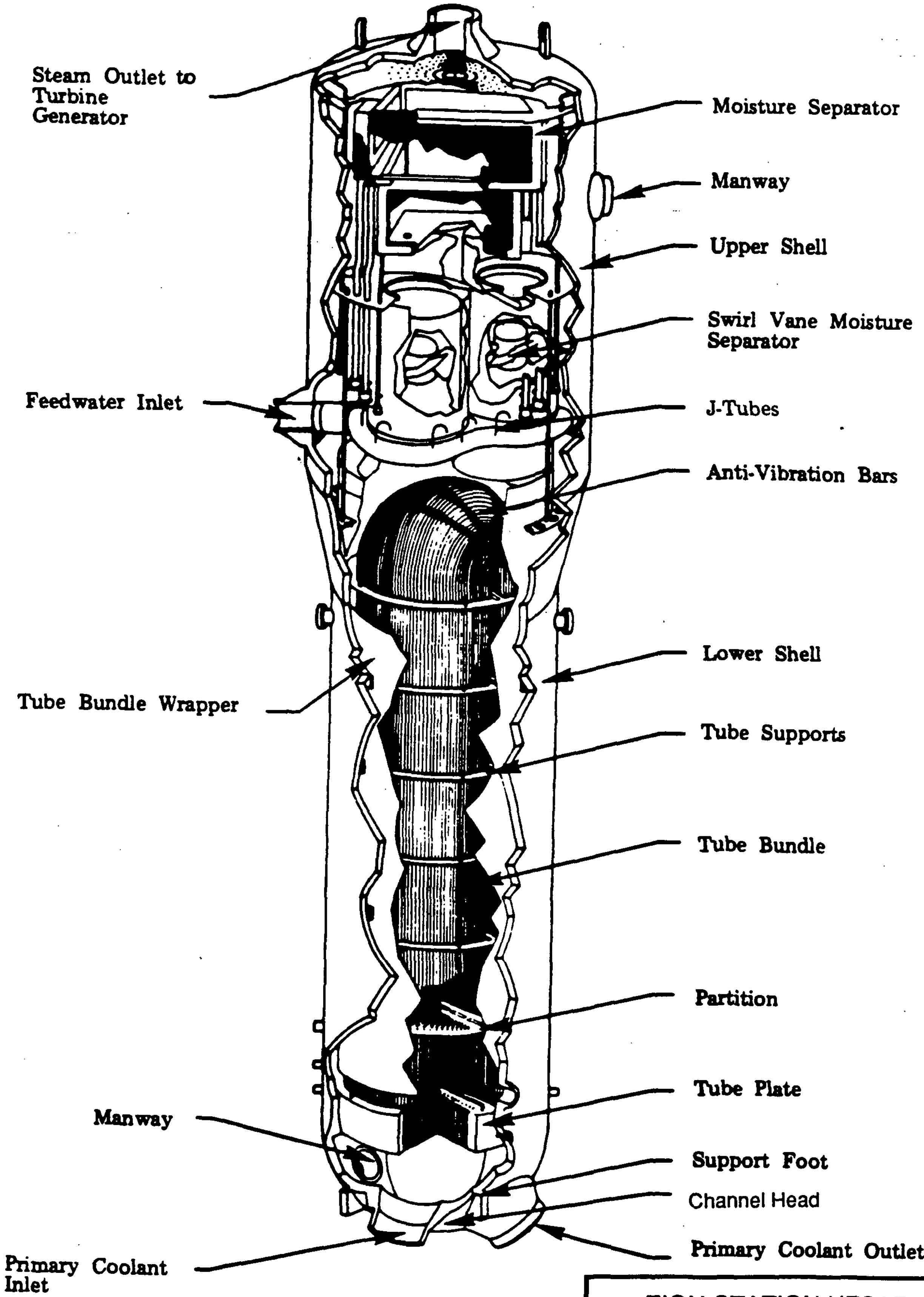

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Figure 5.4-5

STEAM GENERATOR

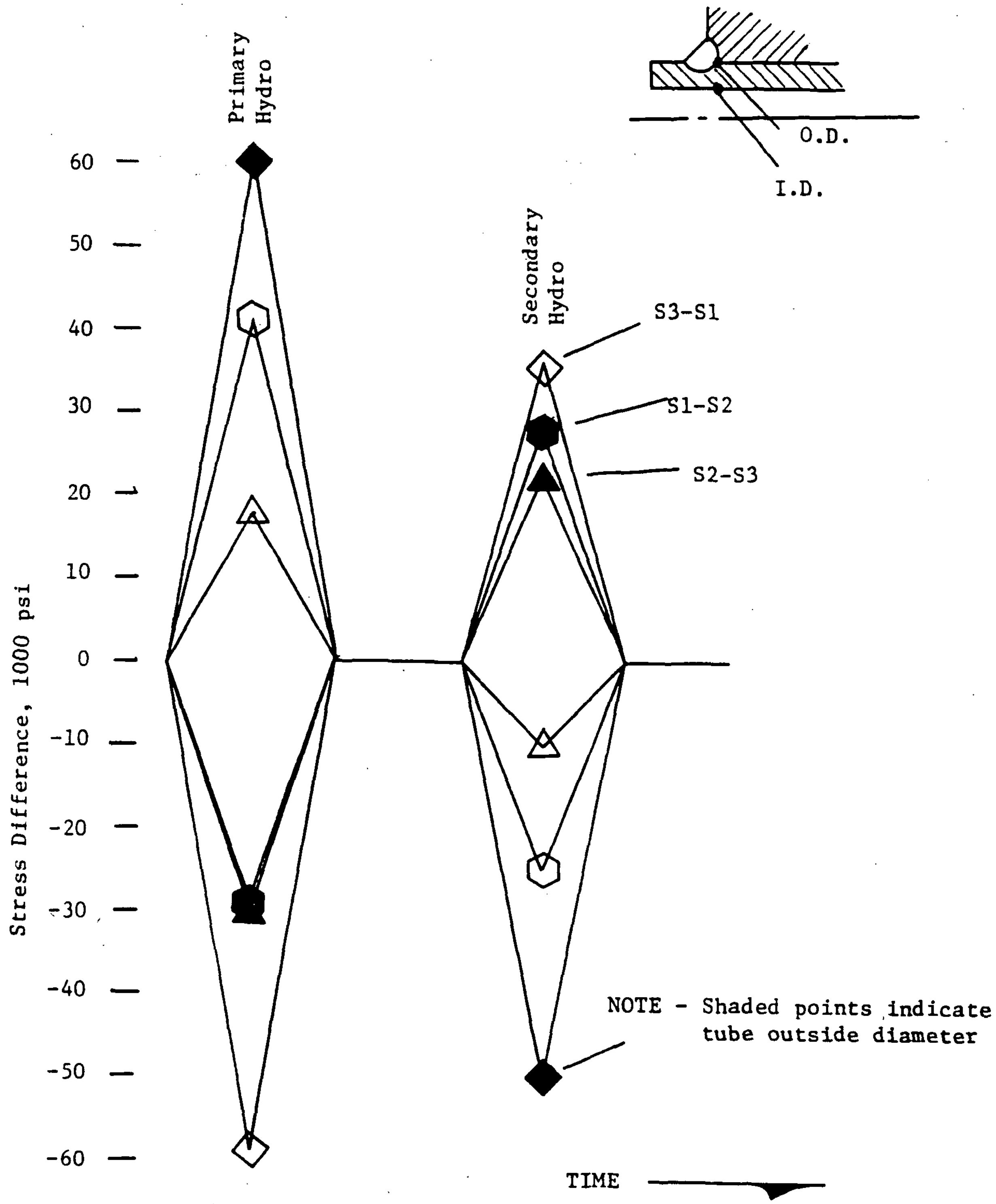

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Figure 5.4-6
PRIMARY AND SECONDARY HYDROSTATIC TEST STRESS HISTORY FOR THE CENTER HOLE LOCATION

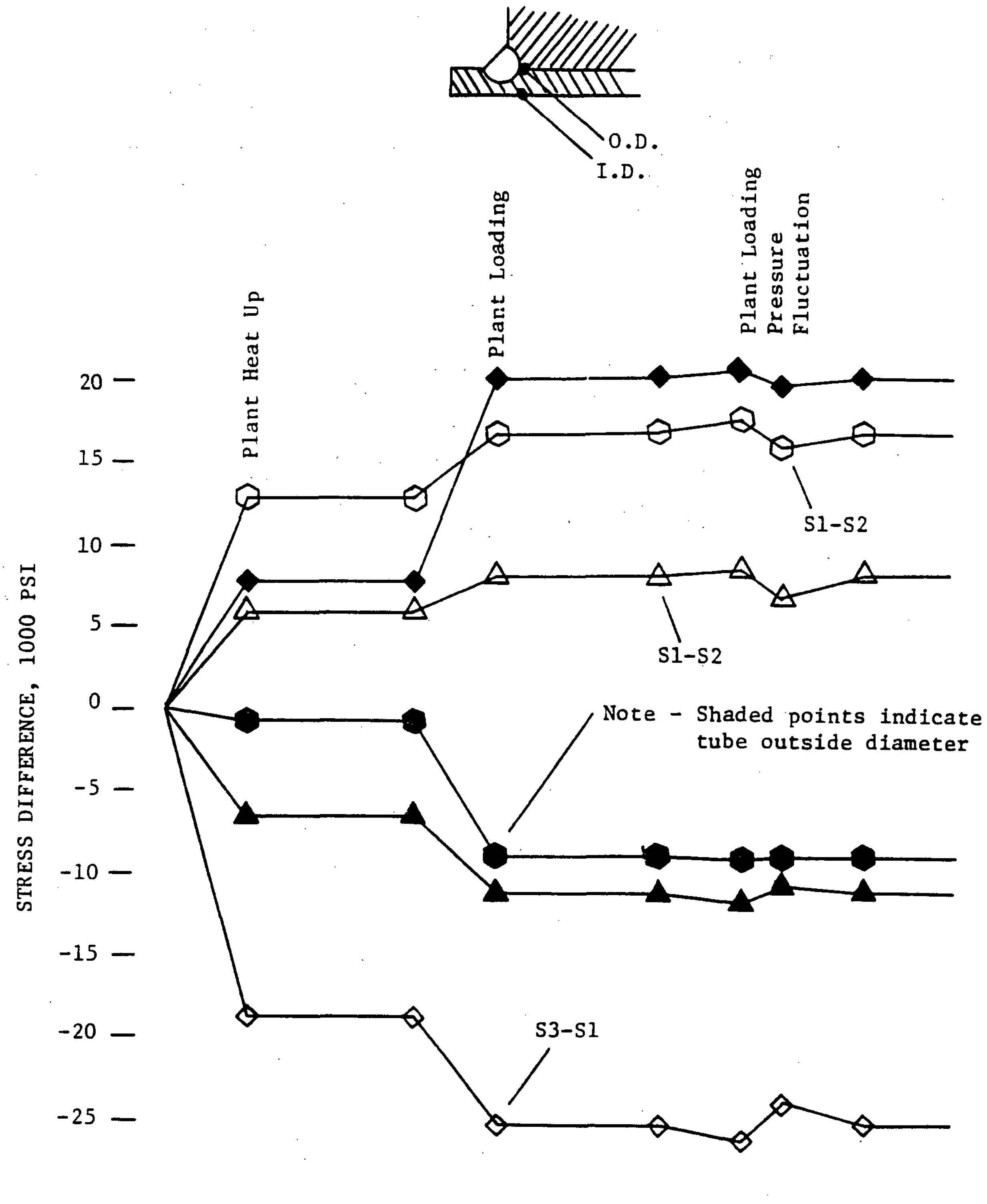

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Figure 5.4-7
PLANT HEATUP AND LOADING OPERATIONAL TRANSIENTS (WITH STEADY-STATE PLATEAU) STRESS HISTORY FOR THE HOT SIDE CENTER HOLE LOCATION

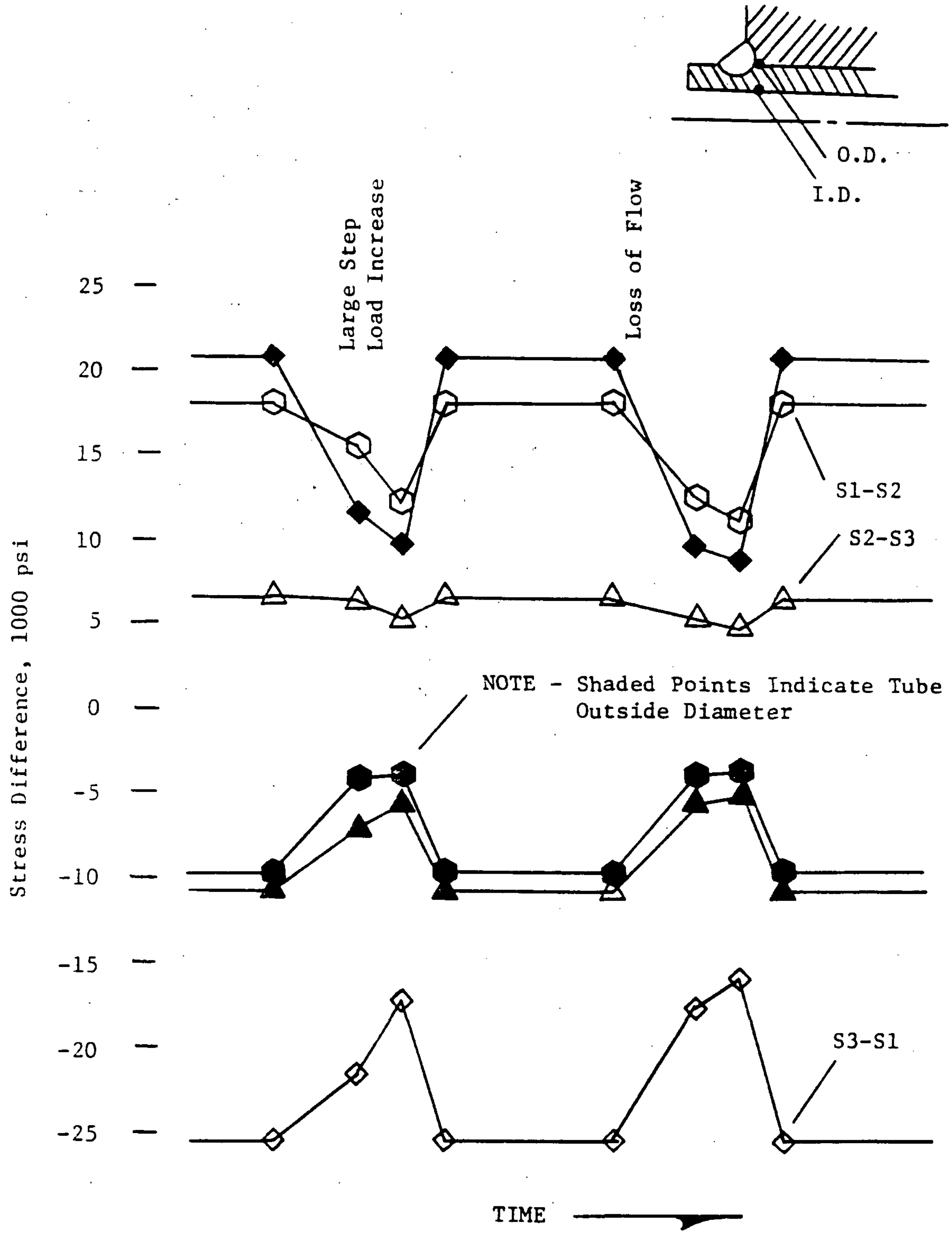

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Figure 5.4-8

LARGE STEP LOAD DECREASE AND LOSS OF FLOW STRESS HISTORY FOR THE HOT SIDE CENTER HOLE LOCATION

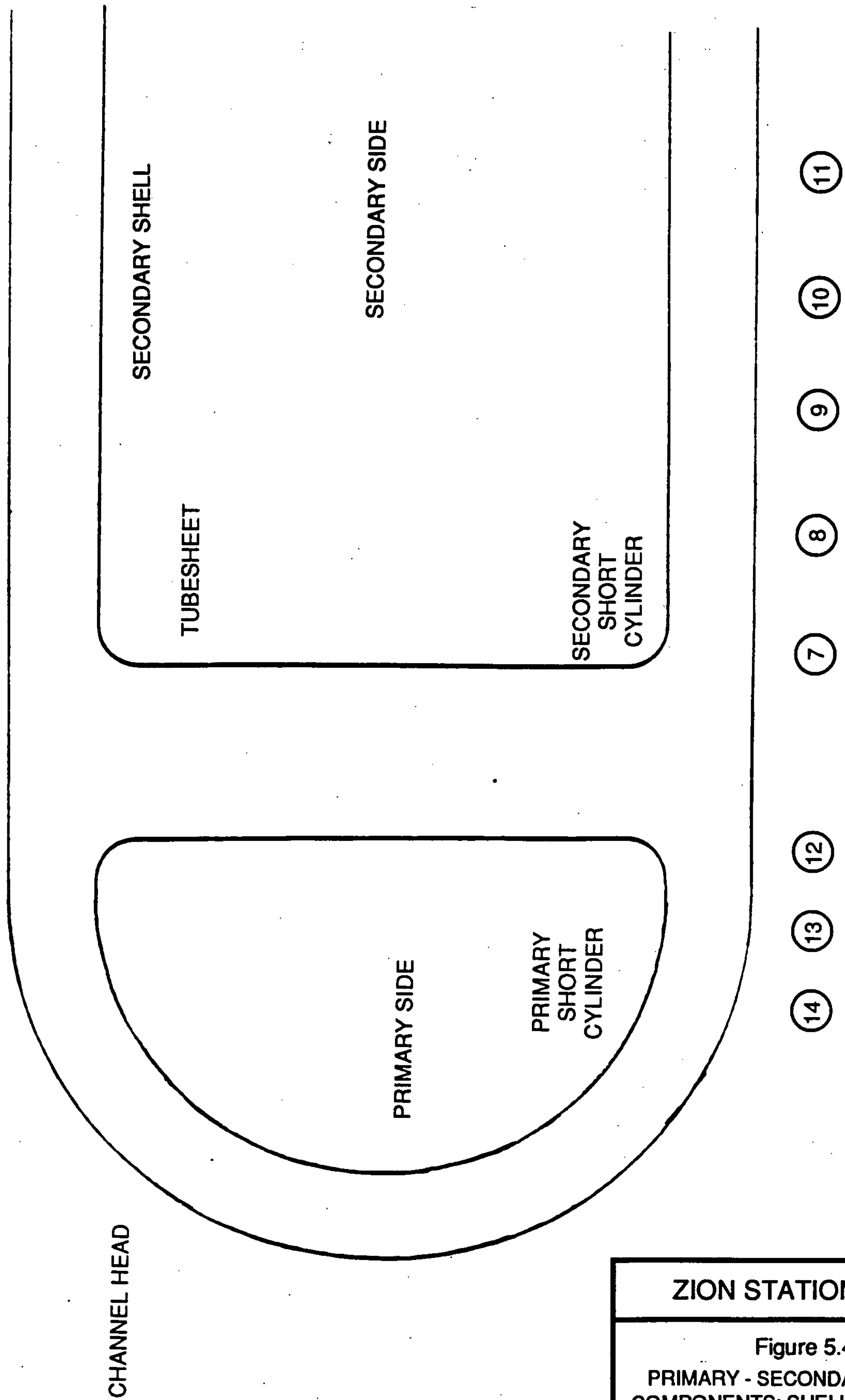

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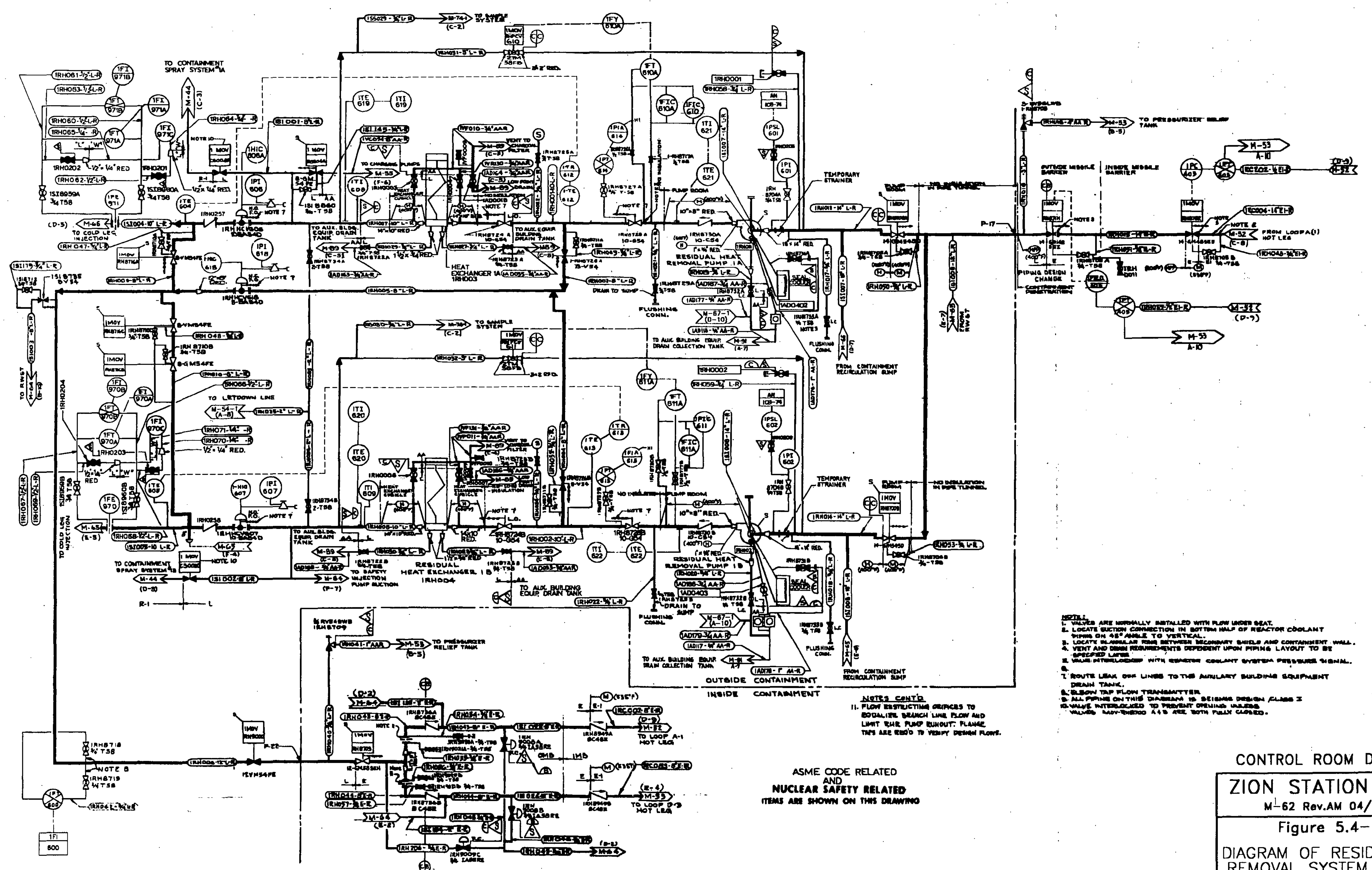
Figure 5.4-9

PRIMARY - SECONDARY BOUNDARY COMPONENTS; SHELL LOCATIONS OF STRESS INVESTIGATIONS



CONTROL ROOM DRAWING

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M-62 Rev.AM 04/11/96

Figure 5.4-10

DIAGRAM OF RESIDUAL HEAT REMOVAL SYSTEM (UNIT 1)

MAY 1996

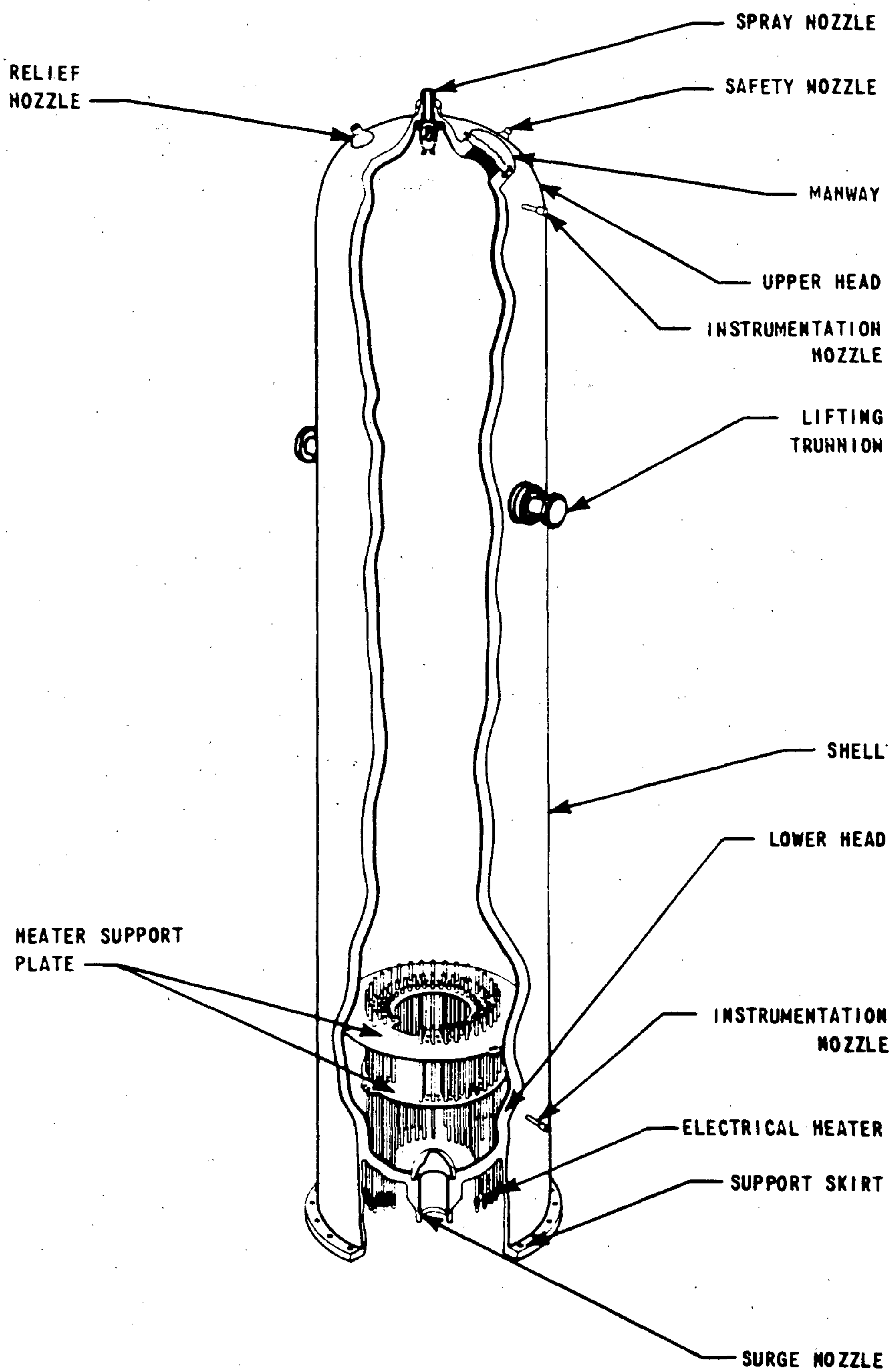

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Figure 5.4-11

PRESSURIZER

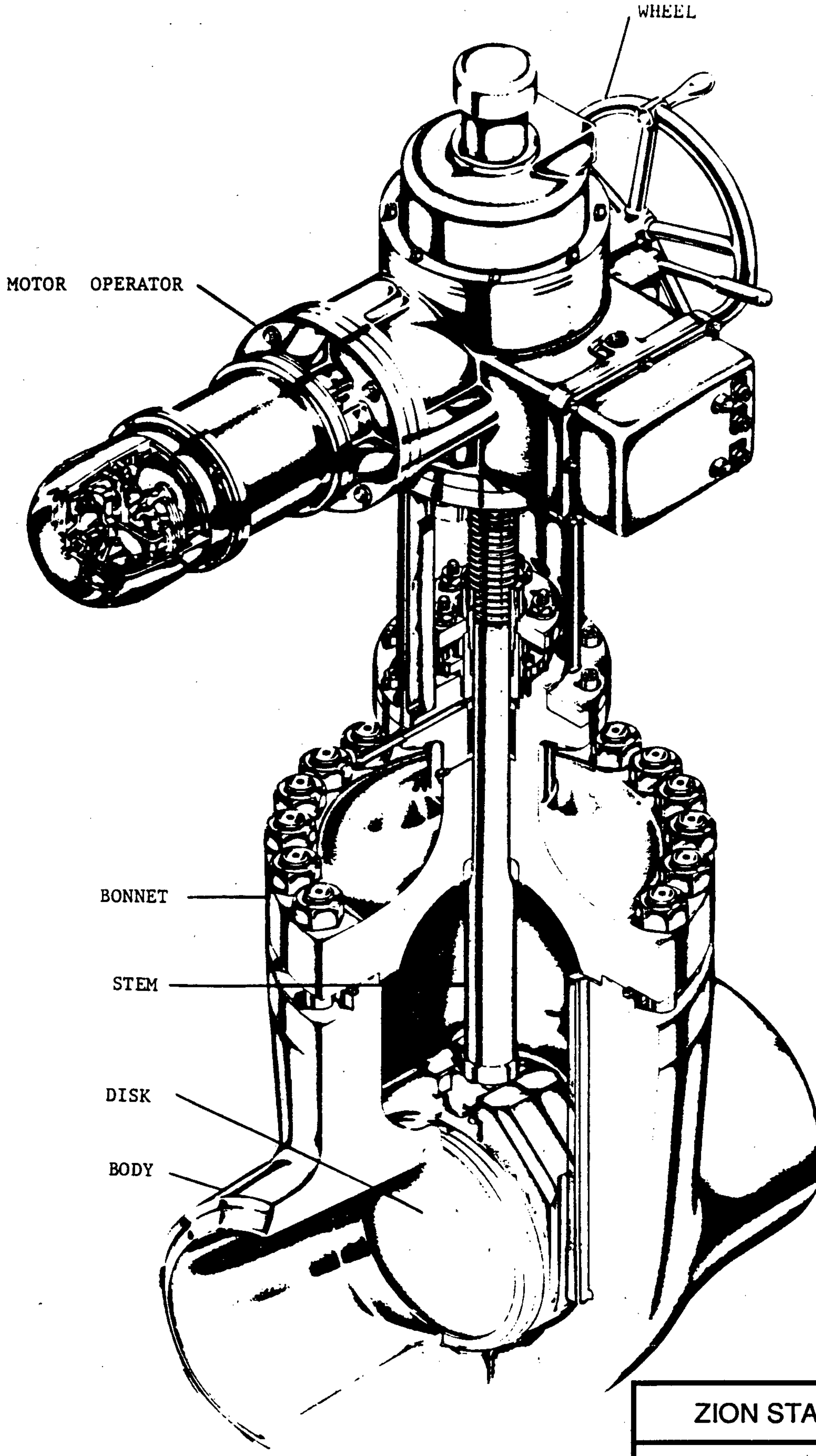

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Figure 5.4-12

REACTOR COOLANT LOOP
STOP VALVE

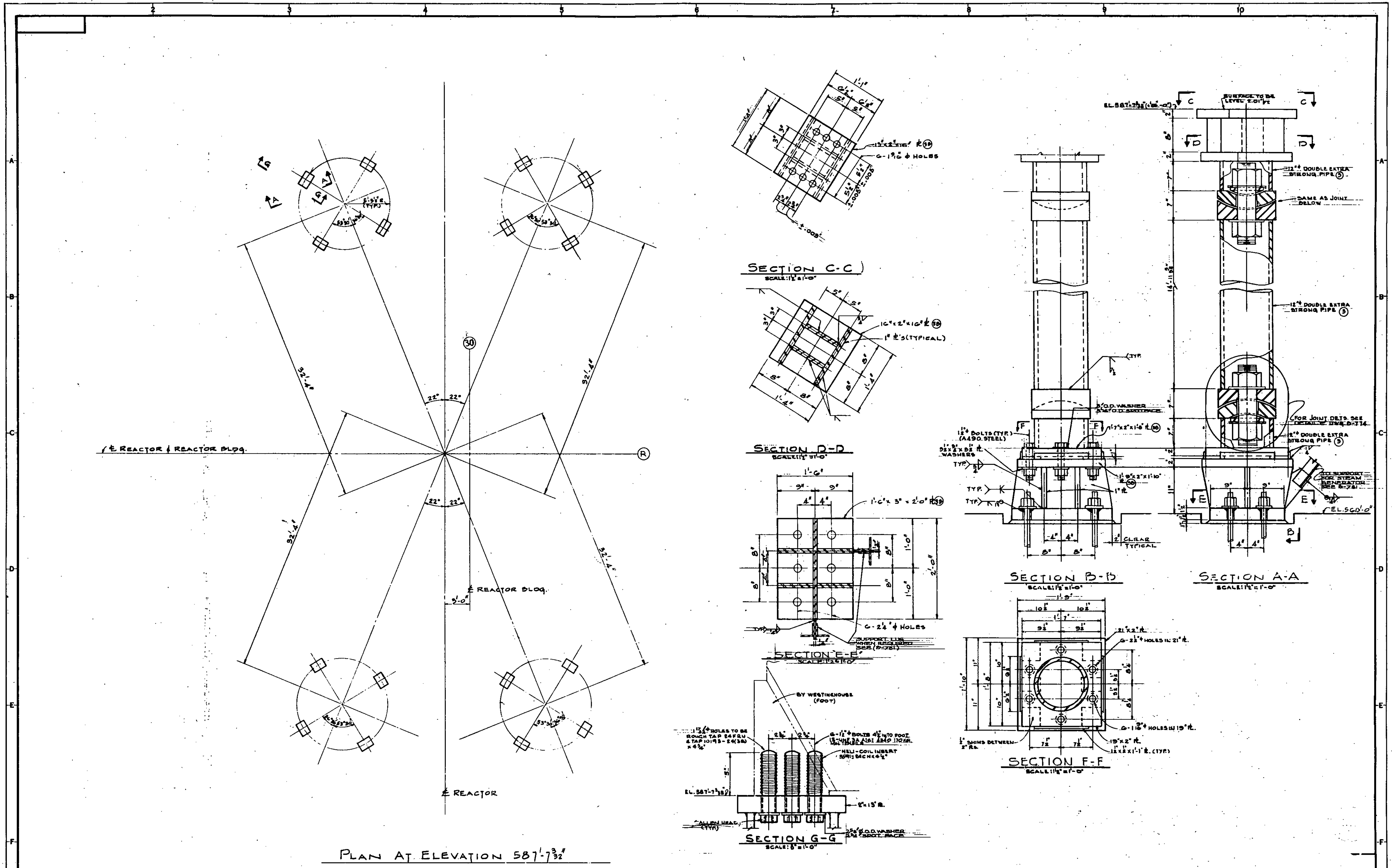

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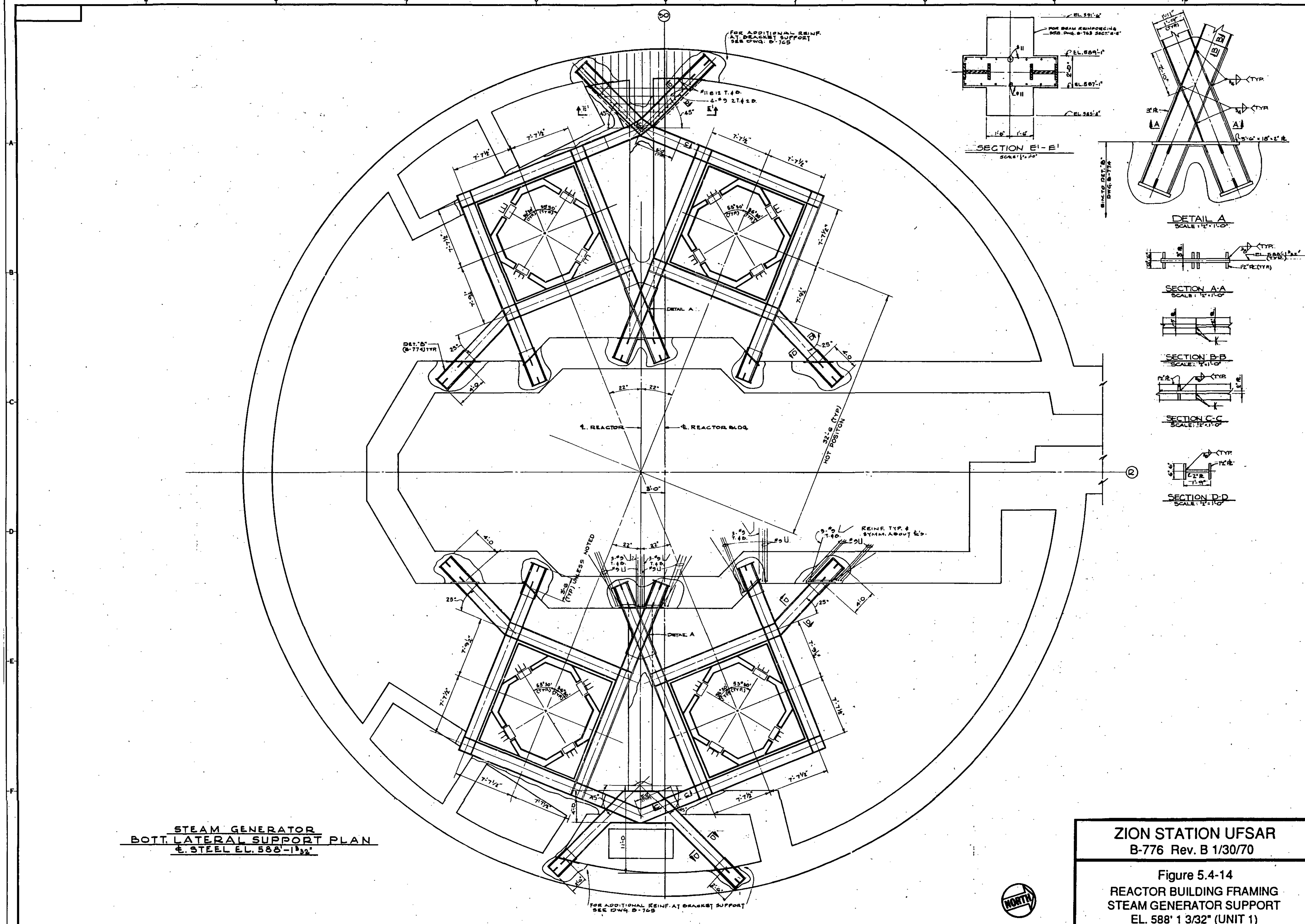
Figure 5.4-13
REACTOR BUILDING FRAMING
STEAM GENERATOR SUPPORT
EL. 587' 1 3/32" (UNIT 1)



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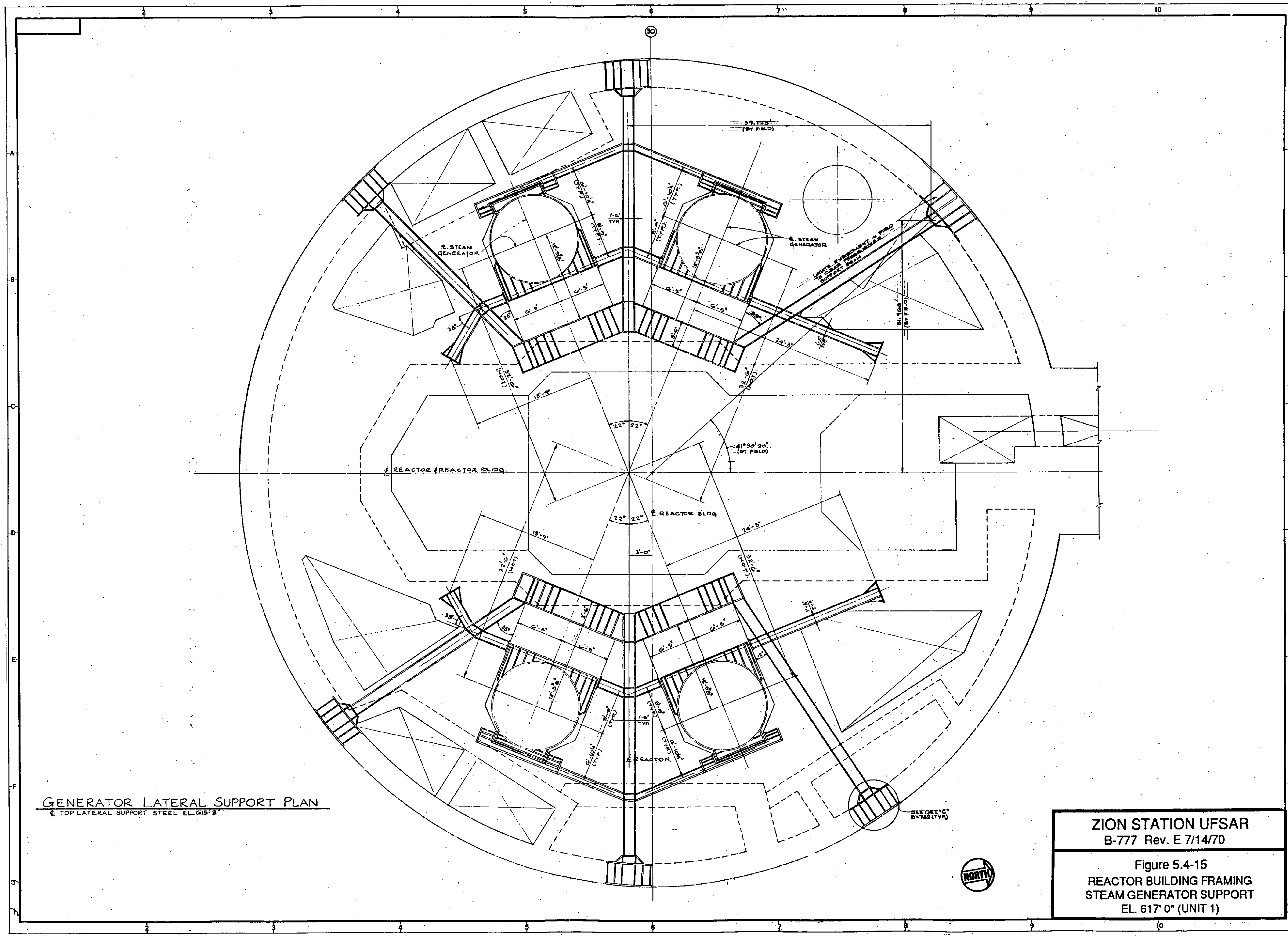
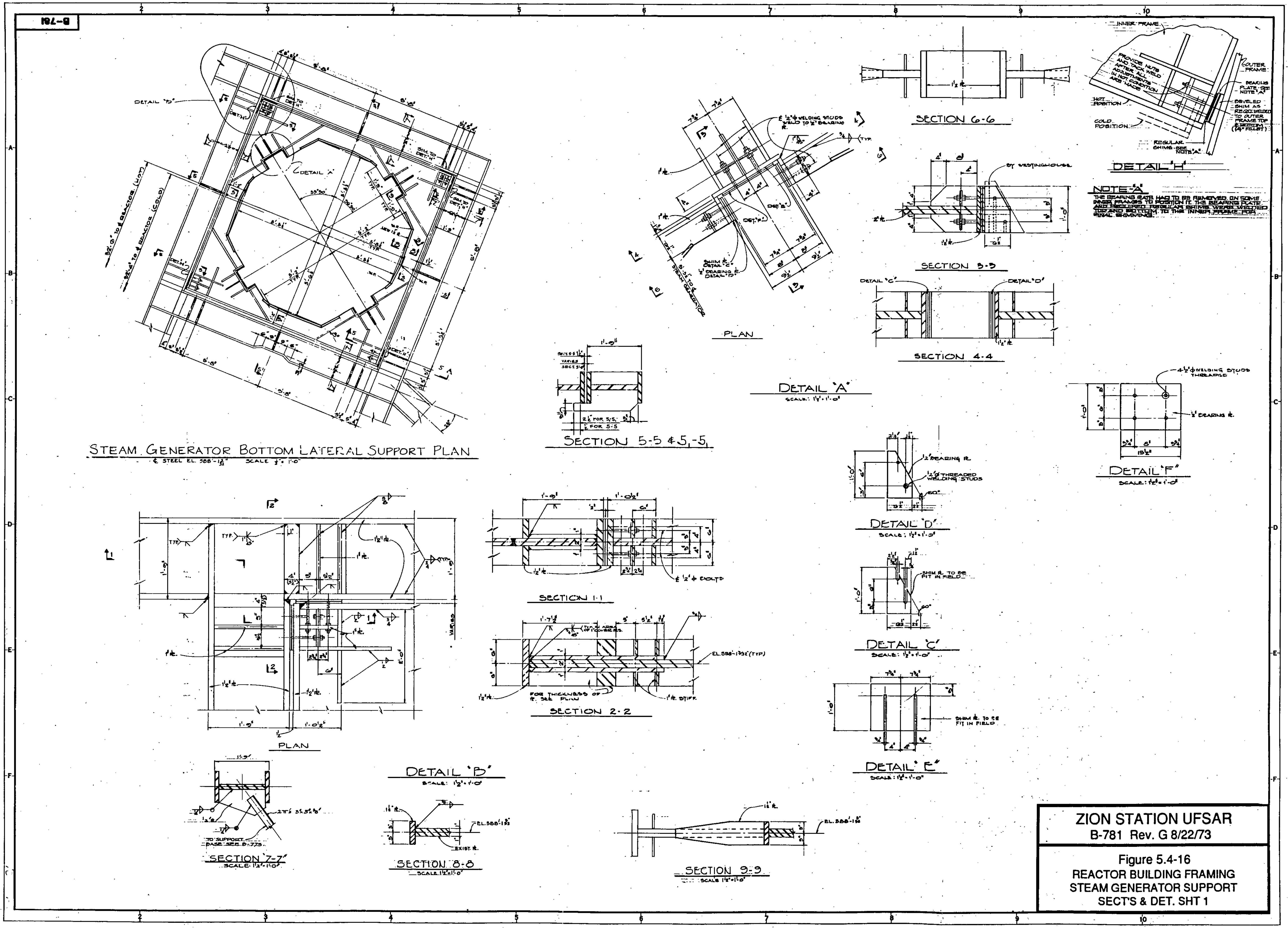
B-776 Rev. B 1/30/70

Figure 5.4-14
REACTOR BUILDING FRAMING
STEAM GENERATOR SUPPORT
EL. 588' 1 3/32" (UNIT 1)



ZION STATION UFSAR
B-777 Rev. E 7/14/70

Figure 5.4-15
REACTOR BUILDING FRAMING
STEAM GENERATOR SUPPORT
EL. 617' 0" (UNIT 1)



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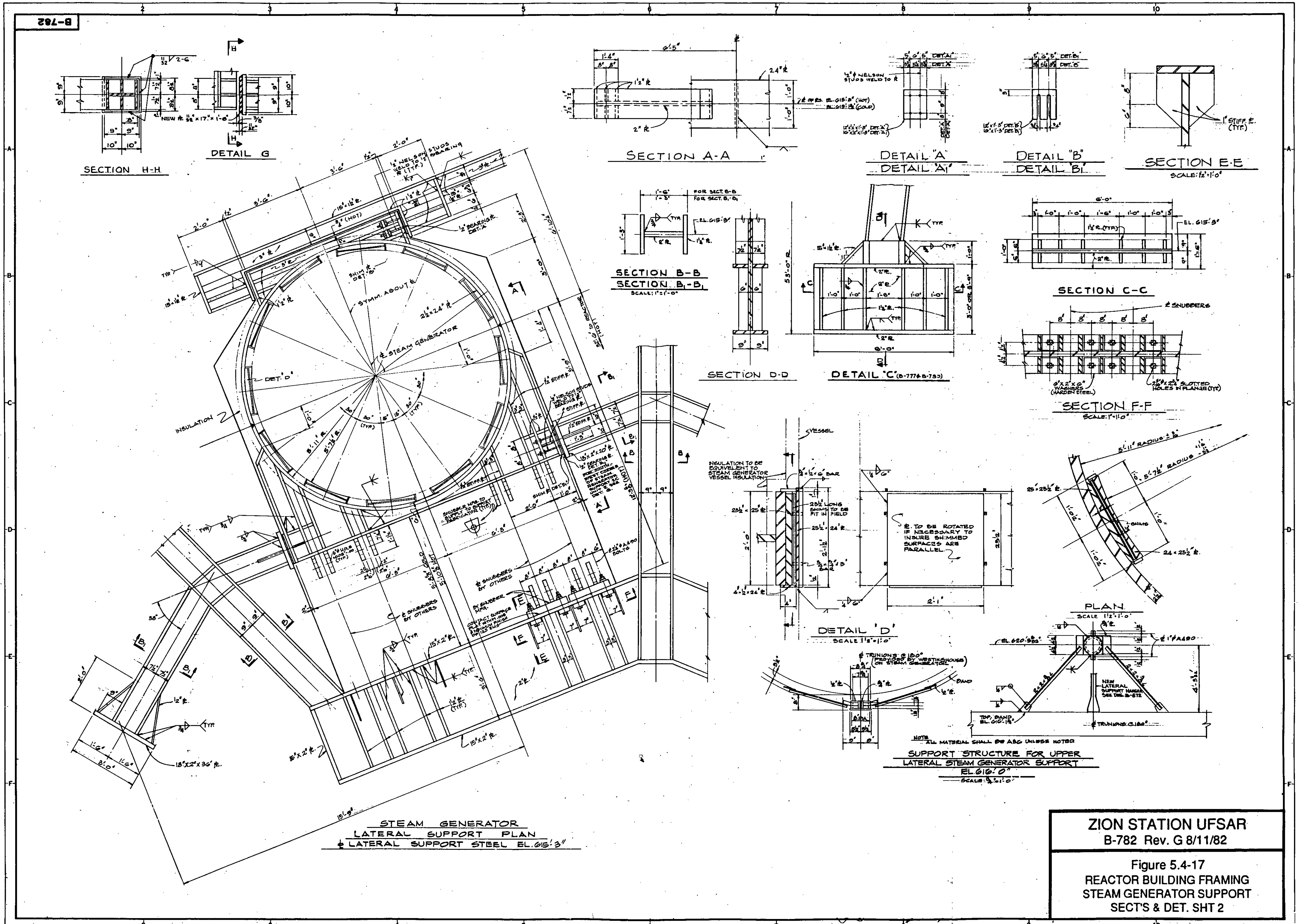
B-781 Rev. G 8/22/73

Figure 5.4-16
REACTOR BUILDING FRAMING
STEAM GENERATOR SUPPORT
SECT'S & DET. SHT 1



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B-782 Rev. G 8/11/82

Figure 5.4-17
REACTOR BUILDING FRAMING
STEAM GENERATOR SUPPORT
SECT'S & DET. SHT 2

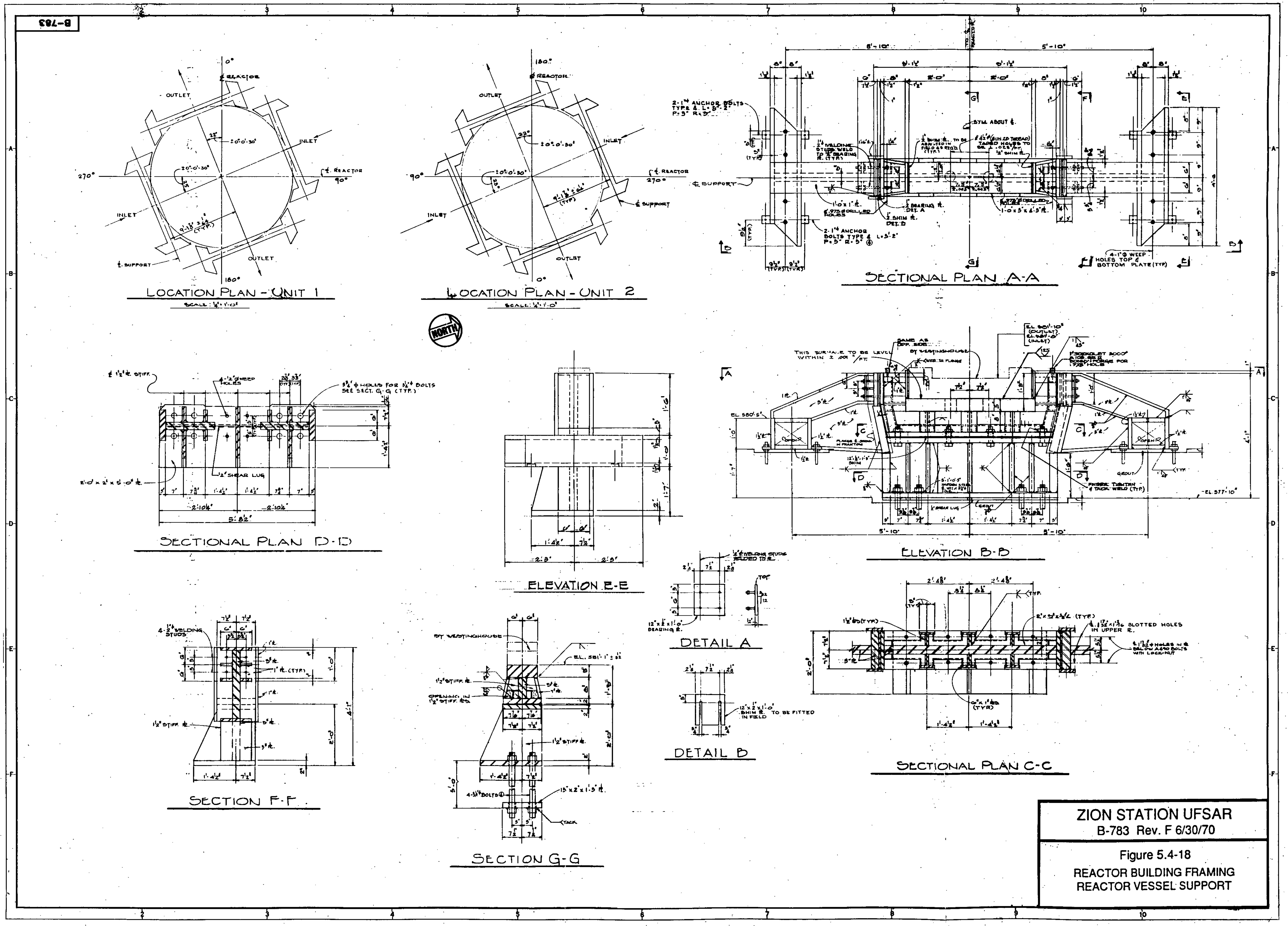

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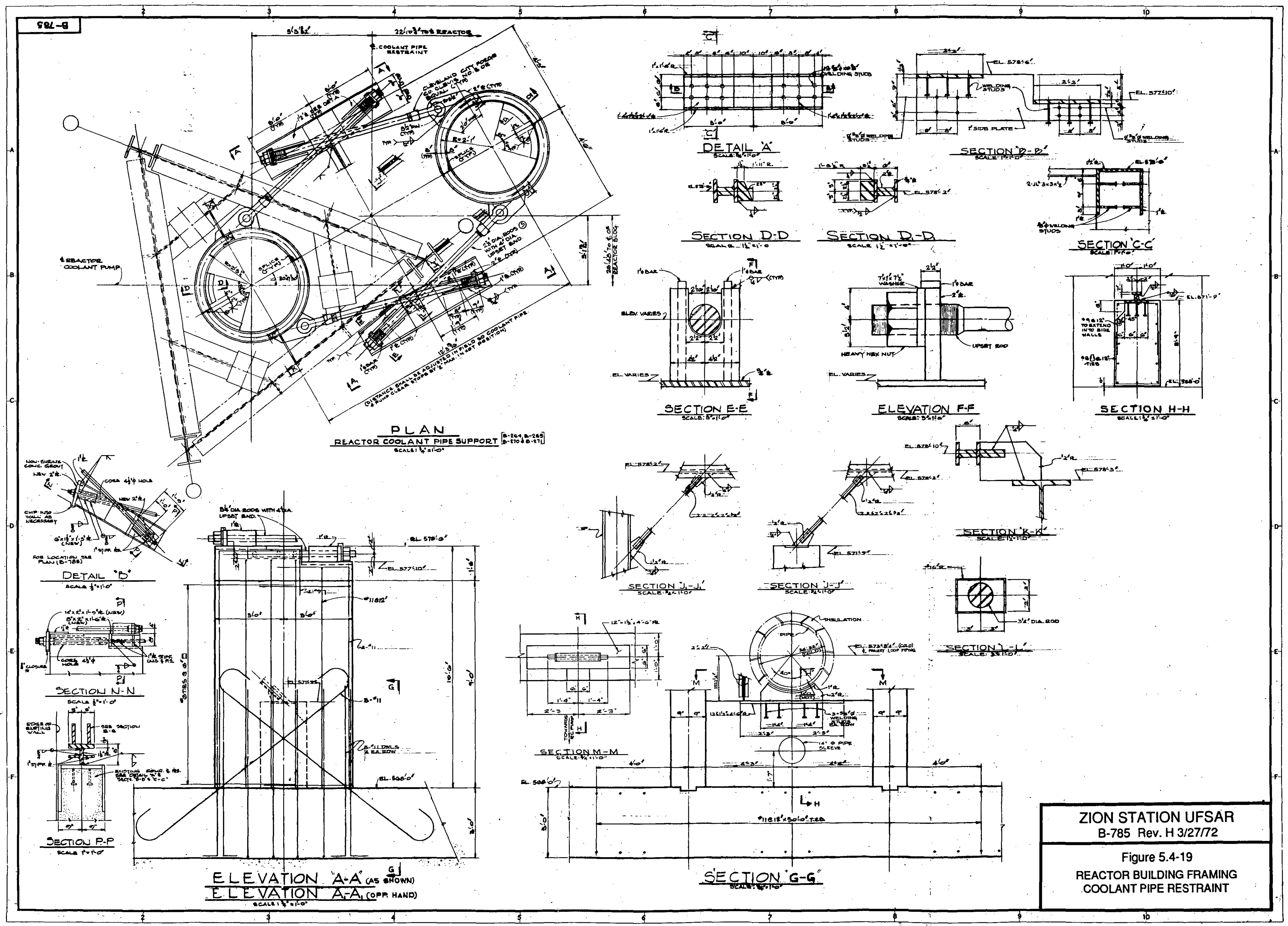
Figure 5.4-18
REACTOR BUILDING FRAMING
REACTOR VESSEL SUPPORT



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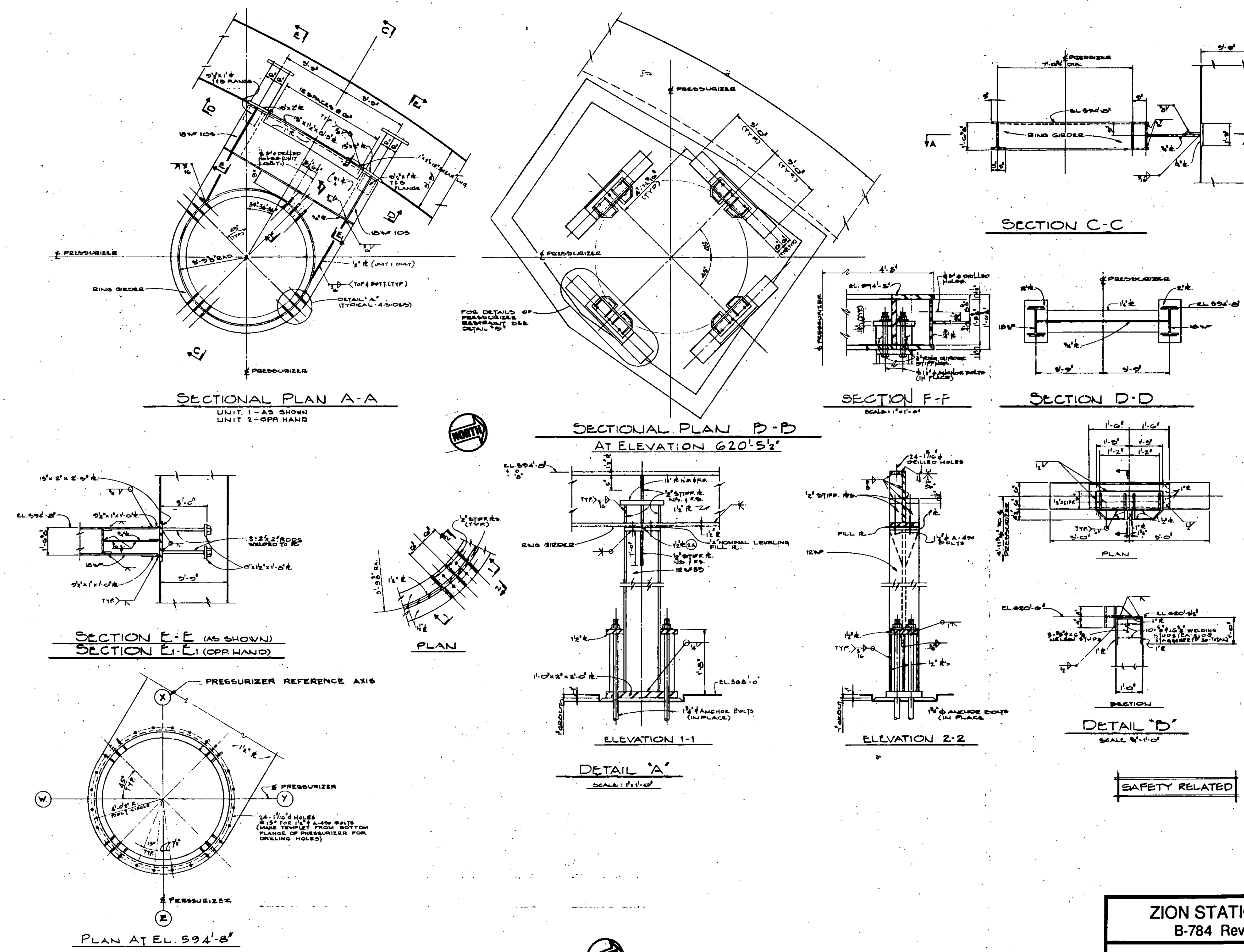
B-785 Rev. H 3/27/72

Figure 5.4-19
REACTOR BUILDING FRAMING
COOLANT PIPE RESTRAINT



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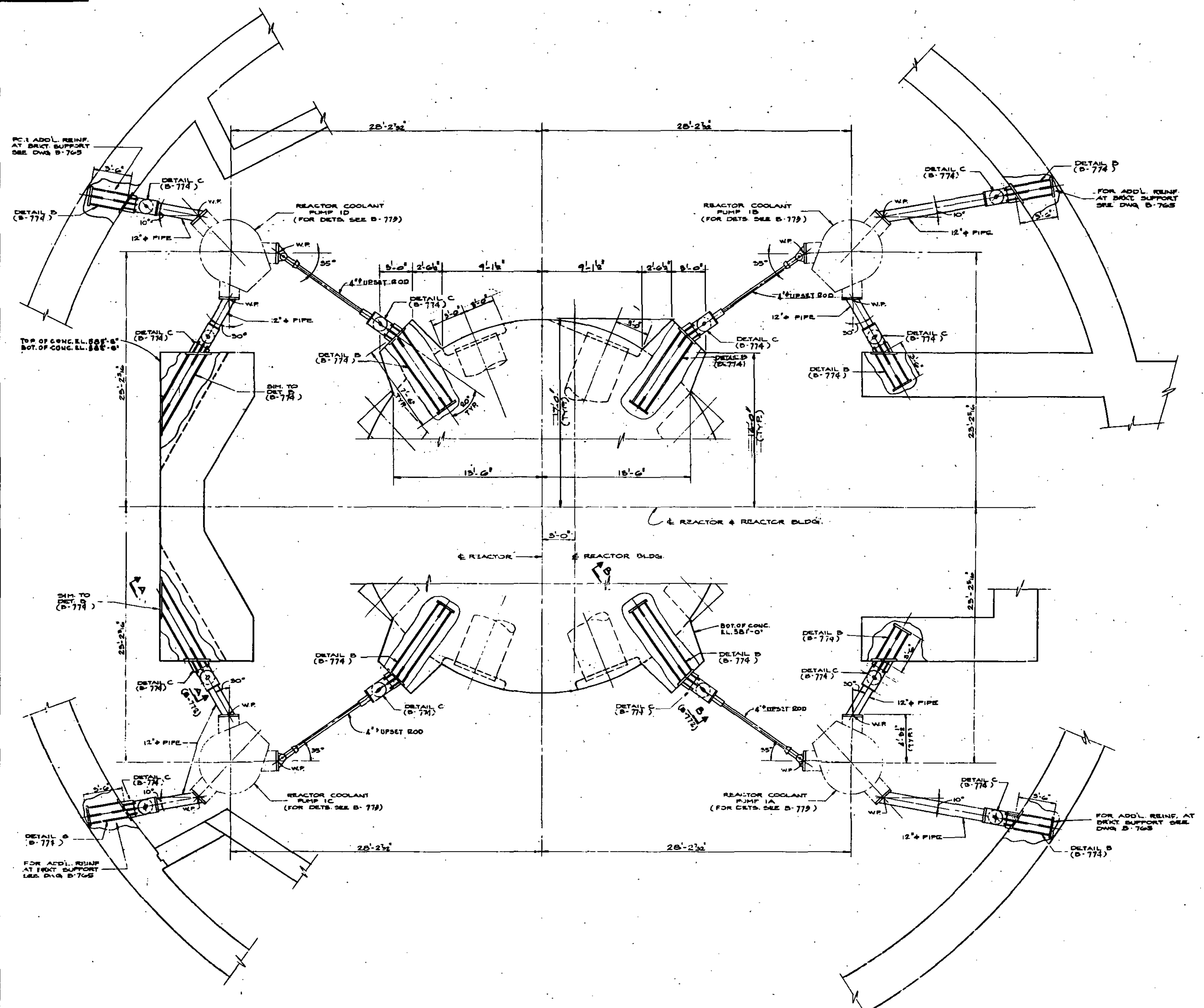
B-784 Rev. G 5/3/76

Figure 5.4-20
REACTOR BUILDING FRAMING
PRESSURIZER

B-771



REACTOR COOLANT PUMP

LATERAL SUPPORT PLAN EL. 584'-0"

FOR ADDITIONAL REINFORCING FOR SUPPORT BRACKETS NOT NOTED SEE DWG. B-772

MATERIAL SPECIFICATIONS

Material Designation	ASTM Designation	Deliver Yield (KSI) Max.	Deliver Yield (KSI) Min.	Product	Thickness Group
1	A7		33	Plate Bars Shapes	to 8"
1A	(A7)[1]	36	33	Shapes Bolt Mat.	to 8"
2	A36		36	Plate Bars Shapes	to 8"
2A	A36		36	Plate	to 4"
3	A588		50	Plate Bars Shapes	to 4"
3A	A588		50	Plate	to 4"
3B	A588		50	Plate	to 4"
4	A193 GRB7		Per ASTM	Bolt Mat.	
5	(A441)[1]	50	45	Bolt Mat.	

NORTH

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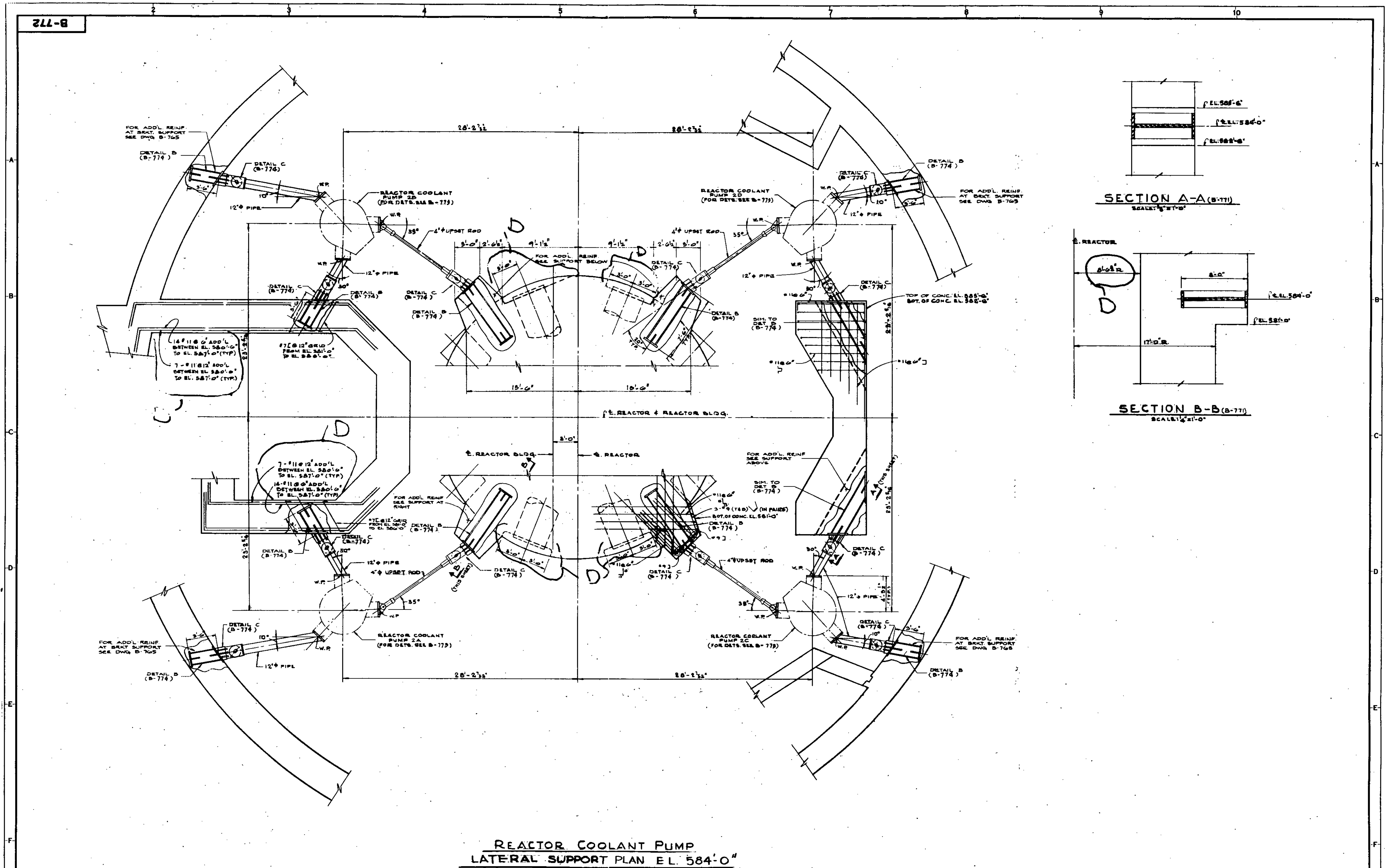
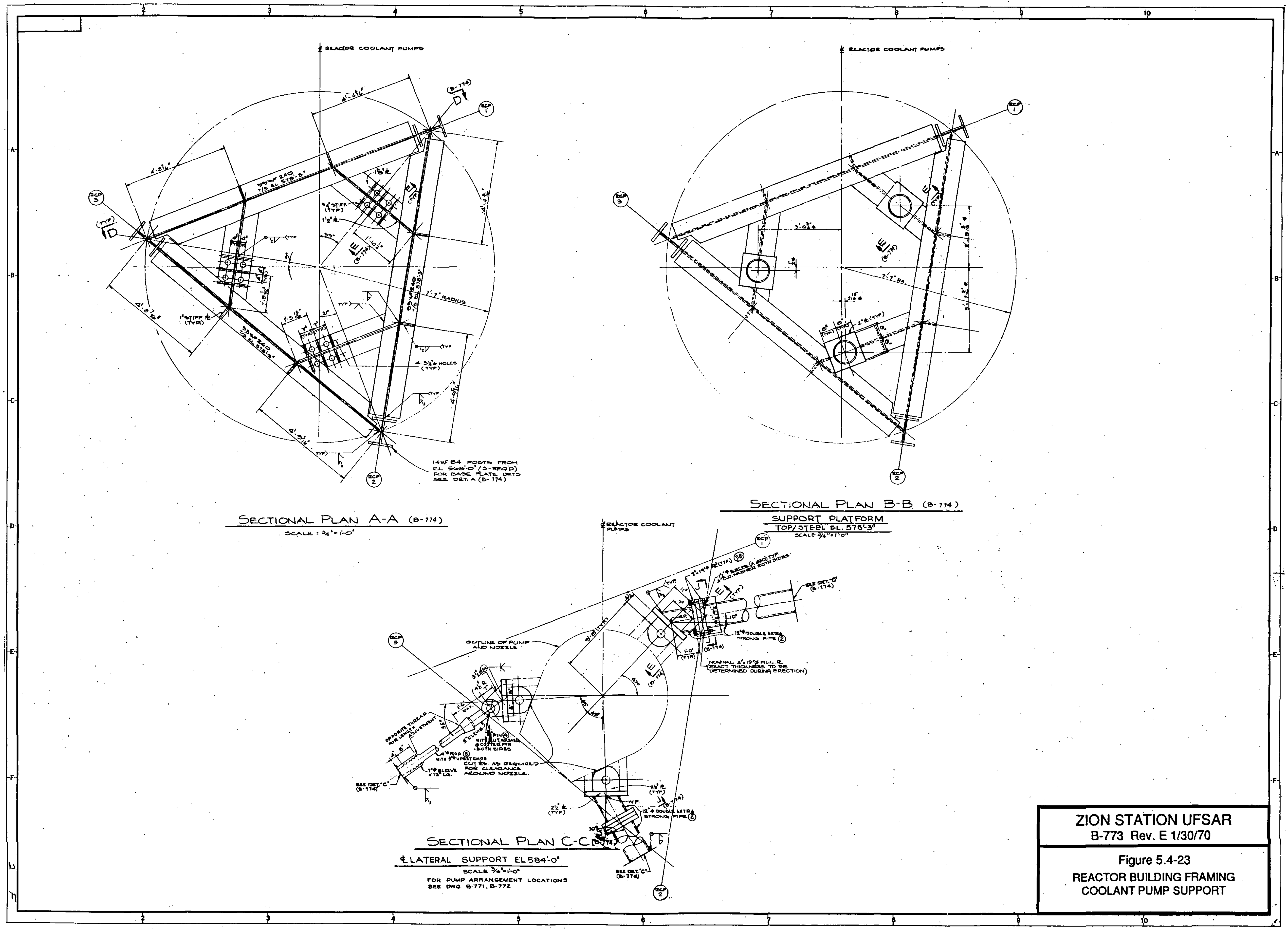
B-771 Rev. F 6/30/70

Figure 5.4-21
REACTOR BUILDING FRAMING
REACTOR COOLANT PUMP
SUPPORT (UNIT 1)



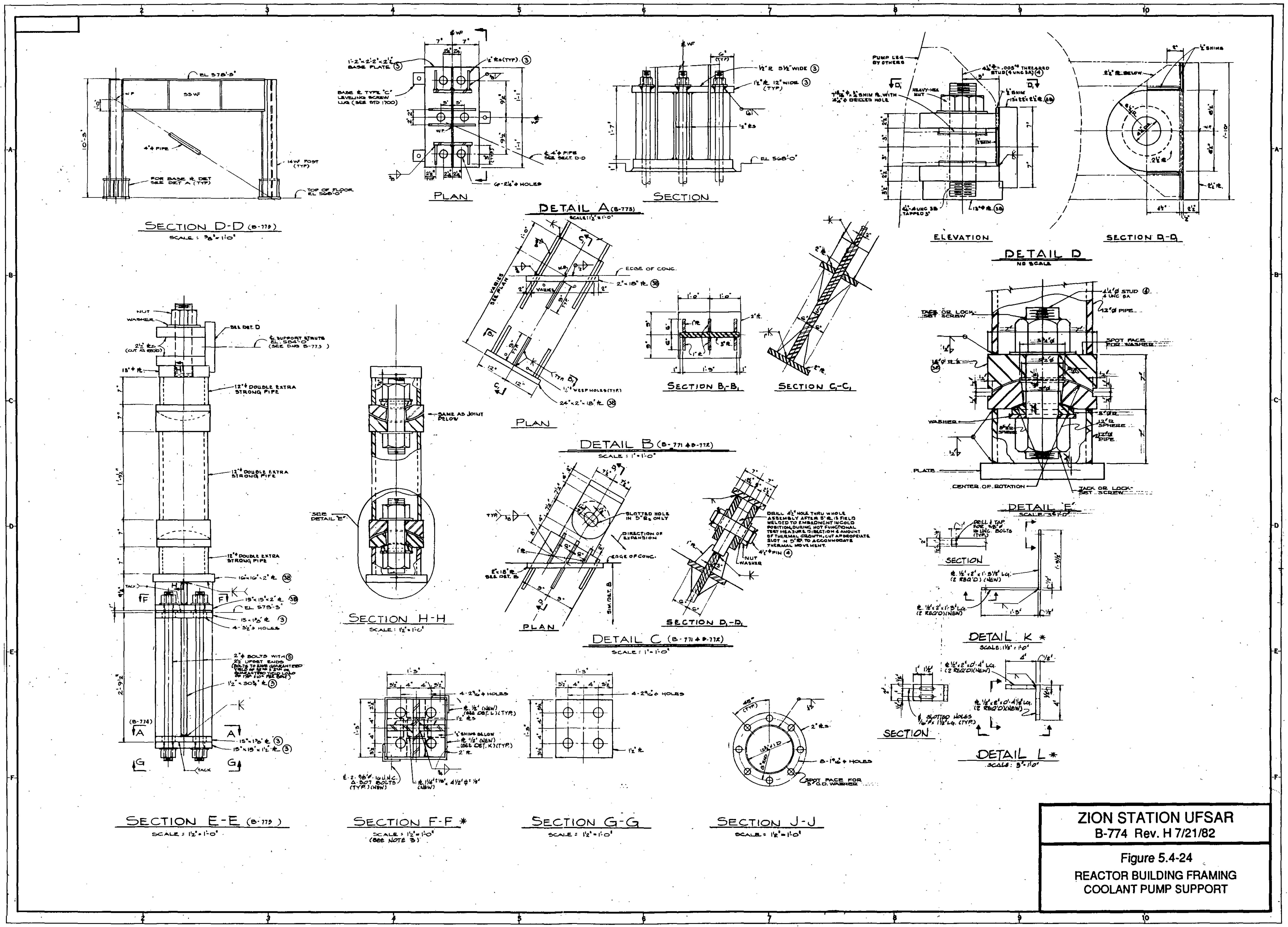
ZION STATION UFSAR
B-772 Rev. D 6/30/70

Figure 5.4-22
REACTOR BUILDING FRAMING
REACTOR COOLANT PUMP
SUPPORT (UNIT 2)



ZION STATION UFSAR
B-773 Rev. E 1/30/70

Figure 5.4-23
REACTOR BUILDING FRAMING
COOLANT PUMP SUPPORT



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B-774 Rev. H 7/21/82

Figure 5.4-24
REACTOR BUILDING FRAMING
COOLANT PUMP SUPPORT

APPENDIX 5A: CRITERIA FOR VESSELS AND PIPING WITHIN REACTOR COOLANT SYSTEM PRESSURE BOUNDARY

Note: This document was retyped for clarity in the 1992 UFSAR Update.

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LIST OF TABLES

LIST OF FIGURES

APPENDIX 5A

CRITERIA FOR VESSELS AND PIPING WITHIN REACTOR COOLANT SYSTEM PRESSURE BOUNDARY

In addition to the loads imposed on the system under normal operating conditions, the design of equipment and equipment supports requires that consideration also be given to abnormal loading conditions such as seismic and pipe rupture. Two types of seismic loadings are considered: Operating Basis Earthquake (OBE) and Design Basis Earthquake (DBE).

For the DBE loading condition, the nuclear steam supply system is designed to be capable of continued safe operation. Therefore, for this loading condition critical structures and equipment needed for this purpose are required to operate within design limits. The seismic design for the DBE is intended to provide a margin in design that assures capability to shutdown and maintain the nuclear facility in a safe condition. In this case, it is only necessary to ensure that required critical structures and components do not lose their capability to perform their safety function. This has come to be referred to as the "no-loss-of-function" criteria and the loading condition as the "Design Basis Earthquake" loading condition.

Not all critical components have the same functional requirements for safety. For example, the reactor containment must retain capability to restrict leakage to an acceptable level. Therefore, based on present practice, general elastic behavior of this structure under the "Design Basis Earthquake" loading condition must be ensured. On the other hand, many components can experience significant permanent deformation without loss of function. Piping and vessels are examples of the latter where the principal requirement is that they retain their contents and allow fluid flow.

The normal as well as abnormal loads for vessels and piping are considered singly and in combination (see Table 5A-1), and the allowable stress limits for each of the possible combinations are limited to those specified in Table 5A-2. The design limit curves that give the allowable stresses for faulted conditions are developed by using the approach presented in WCAP 5890 Rev. 1. This report develops limit curves by using 50 percent of the ultimate strain as the maximum allowable membrane strain. Design limit curves were developed by using the following procedure:

a. Use material data to develop stress-strain curves.

   Stress-strain curves of Type 304 stainless steel, Inconel 600 and SA302B low alloy steel at 600°F were generated from tests using graphs of applied load versus cross-head displacement as automatically plotted by the recorder of the tensile test apparatus. The scale and sensitivity of the test apparatus recorder assured accurate measurement of the uniform strain.

For other materials, stress-strain curves were developed by conservative use of pertinent available material data (i.e., lowest values of uniform strain and initial strain hardening). If the available data was not sufficient to develop a reliable stress-strain curve, three standard ASTM tensile tests of the material in question were performed at design temperature. These data would conservatively apply in developing a stress-strain curve as described above.

b. Normalize the ordinate (stress) of the stress-strain curves to the measured yield strength (Figure 5A-1).

c. Use 20 percent of the uniform strain as defined on the curve developed under item (a) as the allowable membrane strain.

d. Establish the normalized stress ratio at 20 percent of uniform strain on the normalized stress ratio-strain curves developed under item (b).

e. Establish the value of the membrane stress limit.

   Multiply the normalized stress ratio in item (d) by the applicable code yield strength at the design temperature to get the membrane stress limit. Alternatively, for certain materials, the actual physical properties were used.

f. Develop limit curves for the combination of local membrane and bending stresses.

   The limit curves were developed by using the analytical approach presented in WCAP 5890, Rev. 1, and the stress-strain curve up to the membrane stress limit as developed under item (e). In addition, dynamic and stability analyses were performed where required.

Examples of design limit curves as developed by using the above procedure are given in Figures 5A-2 and 5A-3.

TABLE 5A-1 (Sheet 1 of 2)

	LOAD COMBINATION	STRESS LIMIT (NOTE 1)
1.	Normal (deadweight, thermal and pressure)	Normal Conditions
2.	Normal and Operating Basis Earthquake	Upset Condition
3.	Normal and Design Basis Earthquake	Faulted Condition
4.	Normal and Pipe Rupture	Faulted Condition
5.	Normal and Design Basis Earthquake and Pipe Rupture	Faulted Condition

NOTE 1: Definition of Terms from Summer 1968 Addenda to the ASME Boiler and Pressure Vessel Code, Section III.

The Operating Condition categories are defined as follows:

(1) Normal Condition - Any condition in the course of system startup, operation in the design power range, and system shutdown, in the absence of Upset, Emergency, or Faulted Conditions.

(2) Upset Condition - Any deviations from Normal Conditions anticipated to occur often enough that design should include a capability to withstand the conditions without operational impairment. The Upset Condition includes those transients caused by a fault in a system component requiring its isolation from the system, transients due to a loss of load or power and any system upset not resulting in a forced outage. The estimated duration of an Upset Condition shall be included in the Design Specifications. The Upset Conditions include the effect of the specified earthquake for which the system must remain operational or must regain its operational status.

(3) Emergency Condition - Any deviations from normal conditions which require shutdown for correction of the conditions or repair of damage in the system. The conditions have a low probability of occurrence but are included to provide assurance that no gross loss of structural integrity will result as a concomitant effect of any damage developed in the system. The total number of postulated occurrences for such events shall not exceed 25.

(4) Faulted Condition - Those combinations of conditions associated with extremely low probability postulated events whose consequences are such that the integrity and operability of the nuclear energy system may be impaired to the extent where considerations require compliance with safety criteria as may be specified by jurisdictional authorities. Among the Faulted Conditions may be a specified earthquake for which safe shutdown is required.

LOADING CONDITIONS AND STRESS LIMITS: PRESSURE VESSELS

LOADING CONDITIONS	STRESS INTENSITY LIMITS	NOTE
1. Normal Condition	(a) $P_m \leq S_m$	
	(b) P_m (or P_L) + $P_B \leq 1.5\ S_m$	1
	(c) P_m (or P_L) + $P_B + Q \leq 3.0S_m$	2
2. Upset Condition	(a) $P_m \leq S_m$	
	(b) P_m (or P_L) + $P_B \leq 1.5S_m$	1
	(c) P_m (or P_L) + $P_B + Q \leq 3.0S_m$	2
3. Emergency Condition	(a) $P_m \leq 1.2S_m$ or S_y whichever is larger	
	(b) P_m (or P_L) + $P_B \leq 1.5\ (1.2S_m)$ or $1.5S_y$ whichever is larger	3
4. Faulted Condition	Design Limit Curves as discussed in the text and attached	4

P_m = primary general membrane stress intensity

P_L = primary local membrane stress intensity

P_B = primary bending stress intensity

Q = secondary stress intensity

S_m = stress intensity from ASME B&PV Code, Section III, Nuclear Vessels

S_y = minimum specified material yield (ASME B&PV Code, Section III, Table N-421 or equivalent)

LOADING CONDITIONS AND STRESS LIMITS: PRESSURE PIPING

LOADING CONDITIONS	STRESS LIMITS
1. Normal Condition	(a) $P_m \leq S$
	(b) P_m (or P_L) + $P_B \leq S$
2. Upset Condition	(a) $P_m \leq 1.2S$
	(b) P_m (or P_L) + $P_B \leq 1.25S$
3. Emergency Condition	(a) $P_m \leq 1.2S$
	(b) P_m (or P_L) + $P_B \leq 1.5$ (1.2S)
4. Faulted Condition	Design Limit Curves as discussed in the text and attached

P_m = primary general membrane stress

P_L = primary local membrane stress

P_B = primary bending stress

S = allowable stress from USASI B31.1 Code for Pressure Piping

LOADING CONDITIONS AND STRESS LIMITS: EQUIPMENT SUPPORTS

LOADING CONDITIONS	STRESS INTENSITY LIMITS
1. Normal Condition	Working Stresses or Applicable Factored Load Design Values
2. Upset Condition	Working Stresses or Applicable Factored Load Design Values
3. Emergency Condition	Within yield after load redistribution
4. Faulted Condition	Permanent Deflection of Supports Limited to Maintain Supported Equipment Within Design Limit Curves as discussed in the text and attached

Notes for Tables 5A-2

Note 1: The limits on local membrane stress intensity ($P_L \leq 1.5S_m$) and primary membrane plus primary bending stress intensity (P_m (or P_L) + $P_B \leq 1.5S_m$) need not be satisfied at a specific location if it can be shown by means of limit analysis or by tests that the specified loadings do not exceed 2/3 or the lower bound collapse load as per paragraph N-4217.6(b) of the ASME B&PV Code, Section III, Nuclear Vessels.

Note 2: In lieu of satisfying the specific requirements for the local membrane ($P_L \leq 1.5S$) or the primary plus secondary stress Intensity ($P_L + P_B + Q \leq 3S_m$) at a specific location, the structural action may be calculated on a plastic basis and the design will be considered to be acceptable if shakedown occurs, as opposed to continuing deformation, and if the deformations which occur prior to shakedown do not exceed specified limits, as per paragraph N-417,6(a) (2) of the ASME B&PV Code, Section III, Nuclear Vessels.

Note 3: The limits on local membrane stress intensity ($P_L \leq 1.5S_m$) and primary membrane plus primary bending stress intensity (P_m (or P_L) + $P_B \leq 1.5S_m$) need not be satisfied at a specific location if it can be shown by means of limit analysis or by tests that the specified loadings do not exceed 120 percent of 2/3 of the lower bound collapse load as per paragraph N-417.10(c) of the ASME B&PV Code, Section III, Nuclear Vessels.

Note 4: As an alternate to the design limit curves which represent a pseudo plastic instability analysis, a plastic instability analysis may be performed in some specific cases considering the actual strain-hardening characteristics of the material, but with yield strength adjusted to correspond to the tabulated value at the appropriate temperature in Table N-424 or N-425, as per paragraph N-417.11(c) of the ASME B&PV Code, Section III, Nuclear Vessels. These specific cases will be justified on an individual basis.

TYPICAL STRESS STRAIN CURVE
STANDARD ASTM TENSILE TEST
MATERIAL: 304 STAINLESS STEEL
TEMPERATURE: $600^{o}F$

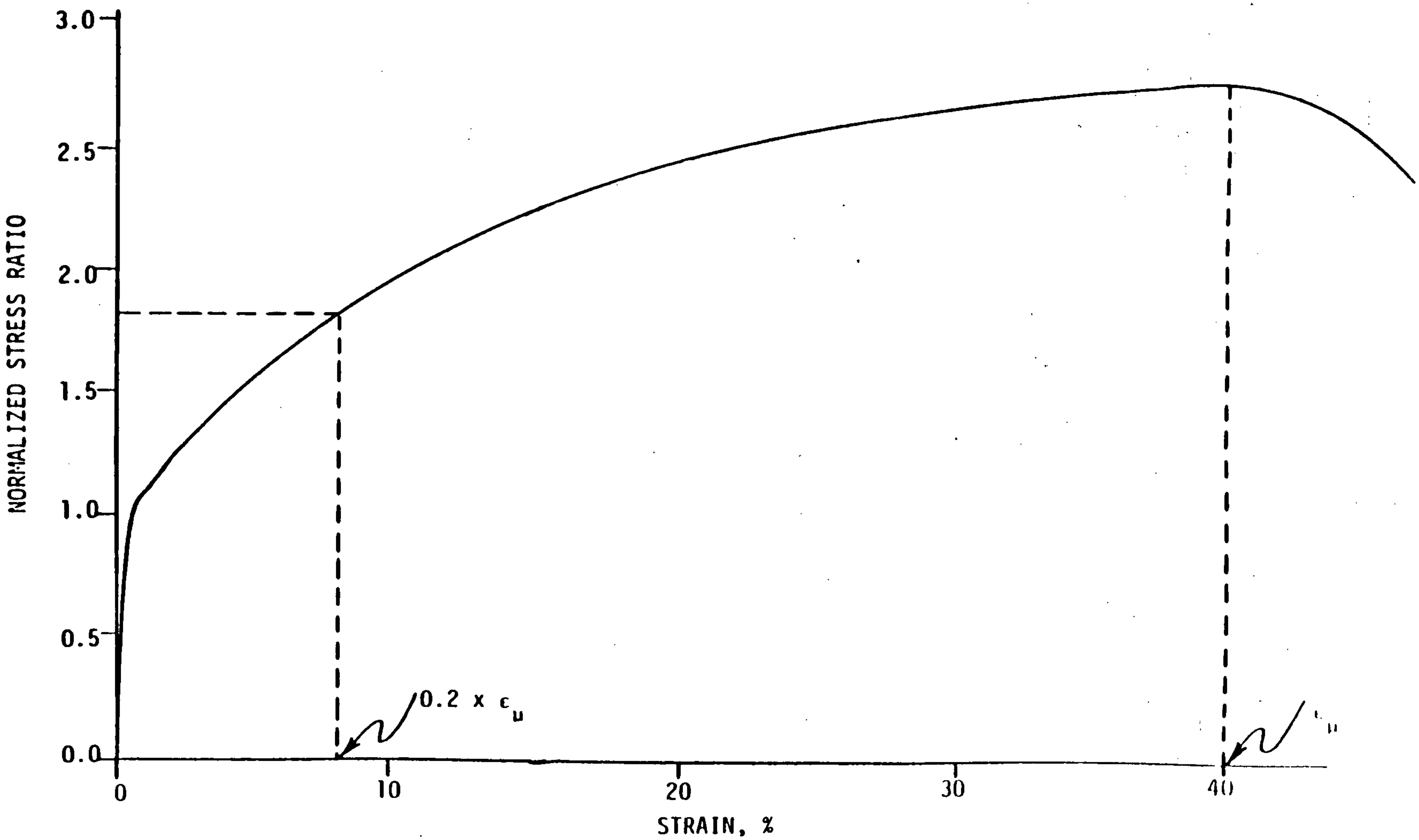
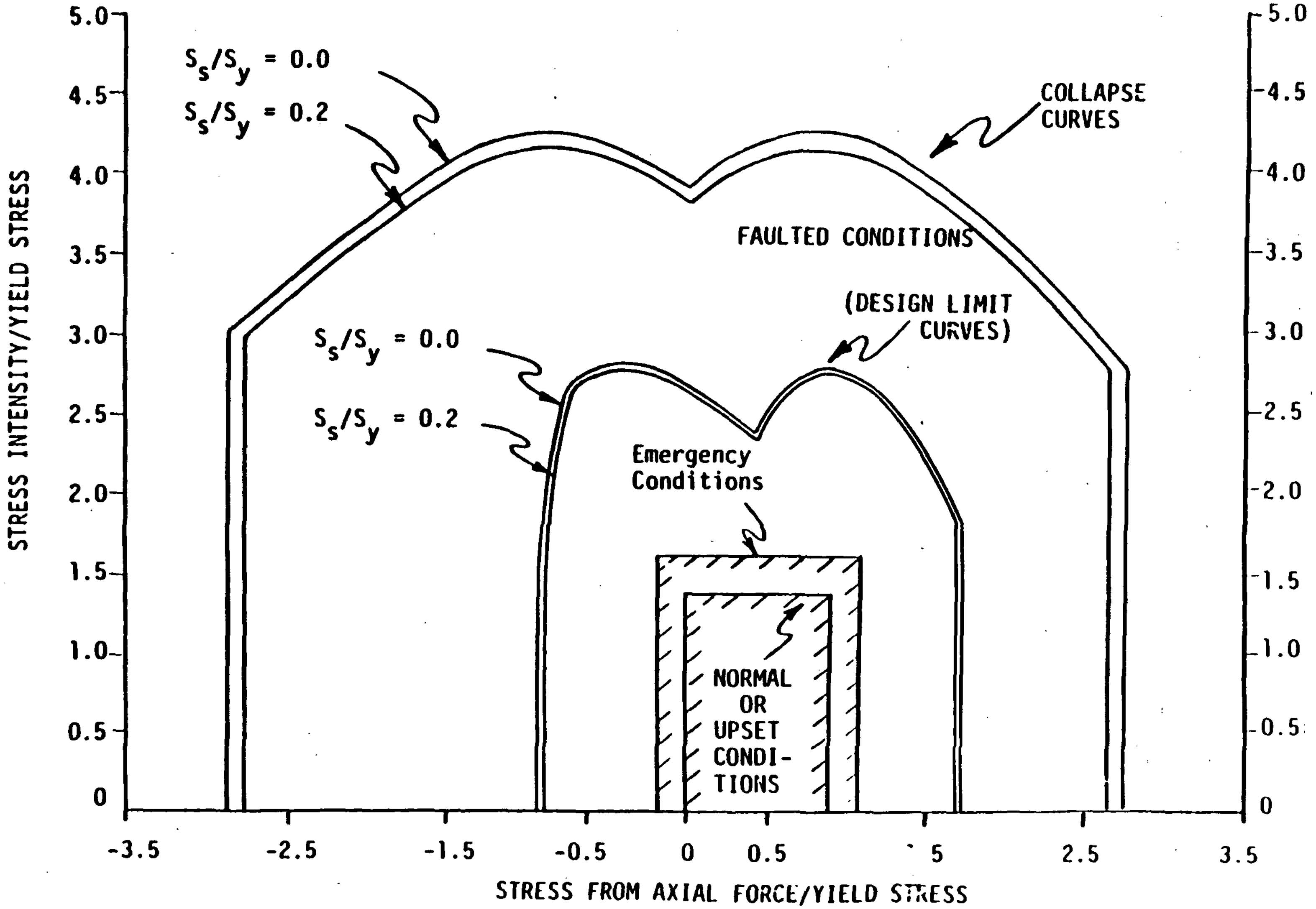


Figure 5A-1

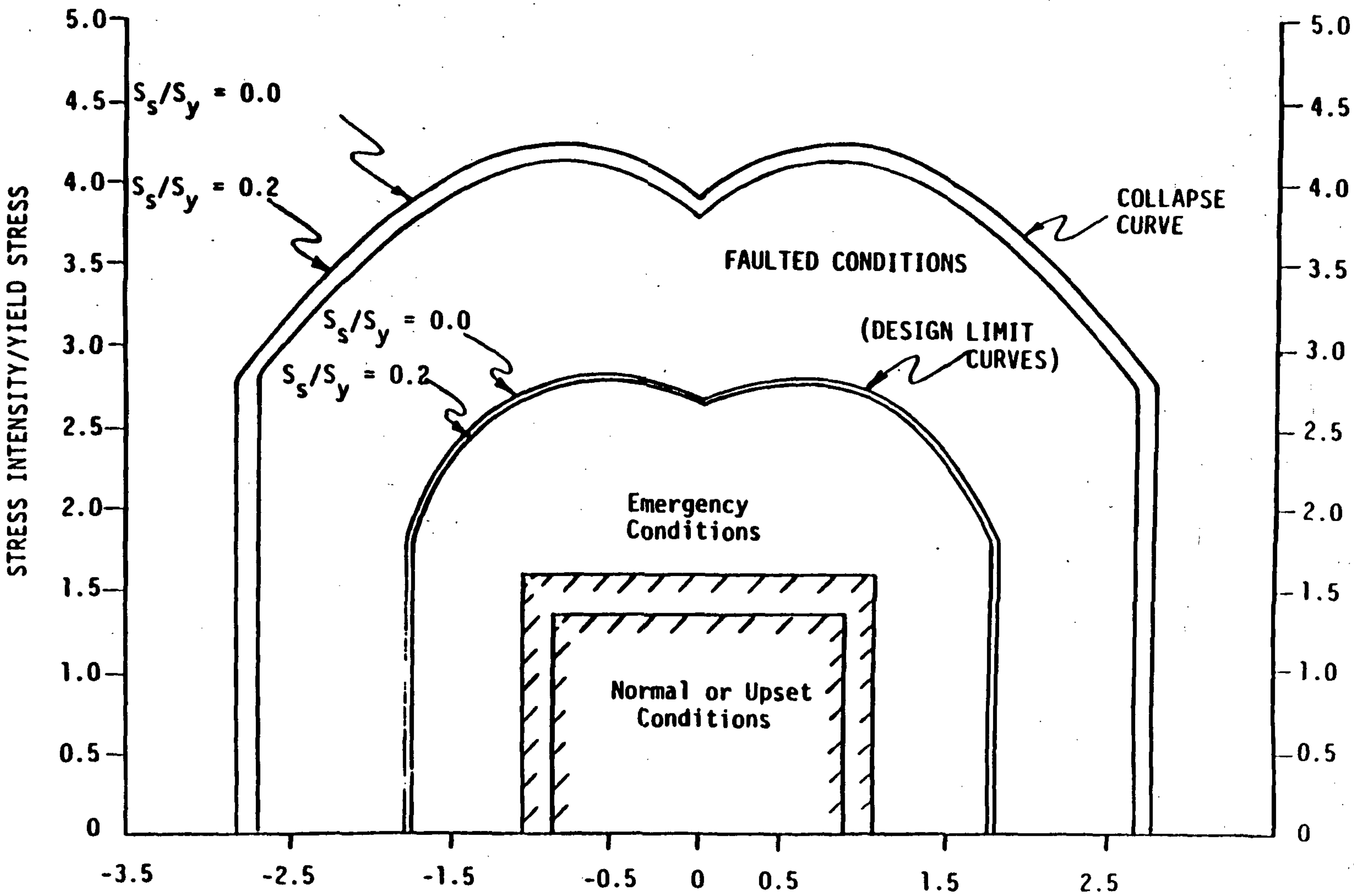
Mat:rial: SA 376 Tp 316
Cross-section: Hollow-Circular
Hoop Stress: 0.90 S_y



COMPARISON BETWEEN DESIGN AND COLLAPSE CONDITIONS

Figure 5A-2

Material: SA 376 Tp 316
Cross-section: Hollow-Circular
Hoop Stress: 0.00 S_y



COMPARISON BETWEEN DESIGN AND COLLAPSE CONDITIONS

Figure 5A-3

APPENDIX 5B: DETERMINATION OF REACTOR PRESSURE VESSEL NDTT

NOTE: This document was retyped for clarity in the 1992 UFSAR Update.

APPENDIX 5B

DETERMINATION OF REACTOR PRESSURE VESSEL NDTT

1. MEASUREMENT OF INTEGRATED FAST NEUTRON (E > 1.0 MEV) FLUX AT THE IRRADIATION SAMPLES

Information on the spectrum of neutron fluxes at the location of the irradiation samples is obtained from the multigroup diffusion code P1MG[1]. Dosimeters including U-238, Np-237, Co-Al, Cu, Ni, Cd shielded Co Al, and Fe from specimens are contained in the capsule assemblies.

The procedure for measurement of fast neutron flux by the ^{54}Fe (n, p)^{54}Mn reaction is described below. The measurement technique for the other dosimeters, which are sensitive to different portions of the neutron spectrum, is similar.

The ^{54}Mn product of this reaction has a half life of 314 days and emits gamma rays of 0.84 Mev energy which are easily detected using a NaI scintillator. In irradiated steel samples, chemical separation of the ^{54}Mn may be performed to ensure freedom from interfacing activities. This separation is simple and very effective, yielding sources of very pure ^{54}Mn activity. In some samples all the interferences may be corrected for by the gamma spectrometric methods without any chemical separation. The count data is used to give the specific activity of ^{54}Mn per gram of iron. Because of the relatively long half life of ^{54}Mn the flux may be calculated for irradiation periods up to about two years. Beyond this time the dosimeter begins to reflect the later stages of the irradiation. Calculation of total dose is from flux and integrated power output. The burnout of the ^{54}Mn produced is not significant until the thermal flux is about 10^{14} neutrons cm^{-2}sec^{-1}.

The analysis of the sample requires that two steps are completed: one the measurement of ^{54}Mn disintegration rate per unit mass of sample and second measurement of iron content of the sample. Having completed these analyses the calculation of the flux is as follows:

For an irradiation the activity of any activation product (A) is given by:

$$A = \phi \sigma N \left(1 - e^{-\lambda t_i}\right) e^{-\lambda t_d} \qquad (1)$$

Where ϕ is the neutron flux, n/cm^2 sec

σ the cross-section, barns

N number of target atoms

λ decay constant of product, sec^{-1}

t_i irradiation time, sec

t_d decay time from end of irradiation to counting time, sec

Then for a power reactor operating at various power levels over some long period we allow for flux changes by dividing the exposure period into several parts and normalizing the flux in each part as that fraction of full power represented. Then for τ periods:

$$A = \phi_m \sigma N \sum_{1}^{\tau} (1 - e^{-\lambda t_{i_n}}) e^{\lambda t_{d_n}} F_n \tag{2}$$

Where ϕ_m = flux at maximum power, n/cm^2, sec

t_{i_n} = cooling time for end of n^{th} period, sec

t_{d_n} = cooling time for end of n^{th} period, sec

F_n = flux normalizing factor which is

$$\frac{actual\ power\ output\ in\ n^{th}\ period}{maximum\ possible\ in\ n^{th}\ period}$$

If now we write

$$\phi_m \sigma N = c \sum_{1}^{55} \phi_{P1MG} (E, r) \cdot \sigma_{Fe} (E) \tag{3}$$

Where E is the energy

r radial distance from core center line.

where the right hand side of equation (3) is the sum of the products of P1MG fluxes and the ^{54}Fe (n,p) ^{54}Mn cross section [2] averaged over the P1MG energy groups, then the measured neutron flux (E > 1 Mev) is given by:

$$\phi\ (E > 1\ Mev) = C \sum_{E=1.0}^{10\ Mev} \phi_{P1MG}\ (E, r)$$

where C is a constant

The error involved in the measurement of the specific activity of the dosimeter after irradiation is estimated to be ± 5%.

2. CALCULATION OF INTEGRATED FAST NEUTRON (E > 1.0 MEV) FLUX AT THE IRRADIATION SAMPLES

The method to be described herein is an approximation to the ideal 3 dimensional neutron transport solution but correlations between its predictions and measurements on samples irradiated in the Yankee and Saxton cores indicate good agreement.

The spectrum of neutron fluxes at the capsule location is obtained from the one dimensional multigroup diffusion code P1MG[1] for the array of annular shields surrounding a cylindrical core of infinite height. The cylindrical core has a cross-sectional area equal to that of the actual core. The radial source distribution chosen for the core represents the expected average over the life of the station. The magnitude of the neutron fluxes generated by the P1MG Code, which does not treat transport effects, is adjusted by application of a spatial correction factor. This factor is the ratio of the fast neutron dose rate calculated by the SPIC-1[3] code for an all water medium surrounding a typical Westinghouse PWR to the fast neutron dose rate obtained by P1MG in the identical geometry. The SPIC-1 fast neutron dose rate calculation uses an empirical fast neutron attenuation kernel in the form of a linear combination of single exponentials which are fitted to the experimental fast neutron dose rate distribution in pure water.

The axial and azimuthal variations of neutron flux at the capsule location are determined separately. The axial distribution is expressed as the ratio of the normalized results of two calculations using PDQ4,[4] a two dimensional 4 group (r,z) diffusion code. In the first of these an infinitely high equivalent cylindrical core with a fission neutron source strength S_l, per unit height is surrounded by an all water medium containing the capsule location. In the second, the finite height is surrounded by an all water medium. The fixed source option of the PDQ4 code is selected so that the axial variation of source strength in the core represents a good approximation to the average over the core life. The radial distribution is identical to that chosen for P1MG. The ratio,

$$\frac{\phi(E,r,z)_F}{S_F} \times \frac{S_I}{\phi(E,r)_I}$$

where subscripts F and I denote finite and infinite core representations respectively, is the required axial correction term.

The azimuthal distributions of neutron fluxes at the sample location are derived from a comparison of the results of the two dimensional 4 group (x,y) code PDQ3[5] and the one dimensional 4 group diffusion program AIM-5[6]. In the PDQ3 calculation the core, whose shape can be specified exactly, is surrounded by an all water medium. The radial and azimuthal source distributions in the core are both reasonable approximations to the averages expected during the core life. The radial source distribution in the AIM-5 calculation, in which the equivalent cylindrical core is surrounded by an all water medium, is identical to that chosen for P1MG.

The product of,

1) The spatially corrected P1MG results,

2) Axial correction term, and

3) Azimuthal correction term,

defines the three dimensional verification of neutron flux at the sample locations.

The technique indicated above overpredicts Saxton measurement by 30 percent and the Yankee measured values by 14 percent. In both reactors the measured results are averages for a set of specimens in a capsule located outside the thermal shield opposite a core corner. More recently, results from SELNI specimens were overpredicted by 10 percent.

The reported technique also gives excellent agreement with measured data reported for the PM2A reactor. Based on the above evidence, it is concluded that the P1MG calculation, corrected as described, is conservative by approximately 20 percent.

3. MEASUREMENT OF THE INITIAL NDTT OF THE REACTOR PRESSURE VESSEL BASE PLATE AND FORGINGS MATERIAL

The unirradiated or initial NDT temperature of pressure vessel base plate and forgings material is presently measured by two methods. These methods are the drop weight test per ASTM E208 and the Charpy V-notch impact test (Type A) per ASTM E23. The NDT temperature is defined in ASTM E208 as "the temperature at which a specimen is broken in a series of tests in which duplicate no break performance occurs at 10°F higher temperature". Using the Charpy V-notch test, the NDTT is defined as the temperature at which the energy required to break the specimen is a certain "fixed" value. For SA 533B Class I and A508 Class 2 and A508 Class 3 steel the ASME III Table N-421 specifies an energy value of 30 ft-lb. This value is based on a correlation with the drop weight test and is referred to as the "30 ft-lb-fix". A curve of the temperature versus energy absorbed in breaking the specimen is plotted. To obtain this curve, 15 tests are performed which include three tests at five different temperatures. The intersection of the energy versus temperature curve with the 30 ft-lb ordinate is designated as the NDTT.

As part of the Westinghouse surveillance program referred to above, Charpy V-impact tests, tensile tests, and fracture mechanics specimens are taken from the core region plates and forgings, and core region weldments including heat-affected zone material. The test locations are similar to those used in the tests by the fabricator at the plate mill.

The uncertainties of measurement of the NDTT of base plate are:

1) Differences in Charpy V-notch foot pound values at a given temperature between specimens.

2) Variation of impact properties through plate thickness.

The fracture toughness technology for pressure vessels and correlation with service failures based on Charpy V-notch impact data are based on the averaging of data. The Charpy V-notch 30 ft-lb "fix" temperature is based on multiple tests by the material supplier, the fabricator, and by Westinghouse as part of the surveillance program. The average of sets of three specimens at each test temperature is used in determining each of five data points (total of 15 specimens). In the review of available data, differences of 0°F to approximately 40°F are observed in comparing curves plotted through the minimum and average values respectively. The value of NDTT derived from the average curve is judged to be representative of the material because of the averaging of at least 15 data points, consistent with the specified procedures of ASTM E23. In the case of the assessment of NDTT shift due to fast neutron flux, the displacement of transition curves is measured. The selection of maximum, minimum or average curves for this assessment is not significant since like curves are used.

There are quantitative differences between the NDT temperature measurements at the surface, 1/4 thickness or the center of a plate. Differences in NDT temperature between 1/4T and the center in heavy plates had been observed to vary from improvement in the NDT temperature to increases up to 85°F. The NDT temperature at the surface had been measured to be as much as 85°F lower than at 1/4T.

The 1/4T location is considered conservative since the enhanced metallurgical properties of the surface are not used for the determination of NDT temperature. In addition, the limiting NDT temperature for the reactor vessel after operation is based on the NDT temperature shift due to irradiation. Since the fast neutron dose is highest at the inner surface, usage of the 1/4T NDT temperature criterion is conservative.

Data are being accumulated on the variation of NDT across heavy section steels at Westinghouse Nuclear Energy Systems. Similarly, the Pressure

Vessel Research Committee sponsors an evaluation of properties of pressure vessel steels in plates and forgings greater than 6 inches thick. Preliminary data show NDT temperature differences between 1/4T and center of less than 20°F. The present criteria of using NDT temperature + 60°F at the 1/4T location without taking advantage of the enhanced properties at the surface of reactor vessel plates is conservative.

To assess any possible uncertainties in the consideration of NDT temperature shift for welds heat affected zone, the base metal, test specimens of these three "material types" are included in the reactor vessel surveillance program.

4. CALCULATION OF THE REACTOR PRESSURE VESSEL REFERENCE TEMPERATURE FOR PRESSURIZED THERMAL SHOCK (RT_{PTS}) VALUES

Calculations have been made in order to determine Reference Temperature for Pressurized Thermal Shock (RT_{PTS}) values for the Units 1 and 2 reactor vessels to meet the requirements of the NRC Rule for Pressurized Thermal Shock (8). These calculations are based on a neutron exposure evaluation and a reactor vessel material study (7). Conclusions reveal that at end-of-license both Units 1 and 2 will be under the NRC RT_{PTS} screening values (270°F for plates, forging, axial welds and 300°F for circumferential welds) and at 32 EFPY Unit 1 will be at or just below the screening values for plates, forgings, axial welds) and Unit 2 will be under all the NRC RT_{PTS} screening values. These conclusions are based on using actual and projected fluence values.

In performing the fast neutron exposure evaluations, two sets of transport calculations were utilized. A single computation in the conventional forward mode was used as the first set of transport calculations to provide baseline data derived from a design basis core power distribution against which cycle by cycle plant specific calculations can be compared. The forward transport calculation was accomplished using R, θ geometry in the DOT discrete ordinates code and the SAILOR cross-section library. SAILOR library is a 47 group, ENDF-BIV based data set produced specifically for light-water reactor applications. Anisotropic scattering is treated with a P_3 expansion of the cross-sections. The design basis core power distribution used in the forward analysis was derived from statistical studies of long-term operation of Westinghouse 4-loop plants. The use of this design basis distribution is expected to yield somewhat conservative results, especially where low leakage fuel management has been employed. The second set of transport calculations, the adjoint analysis, was also utilized using the P_3 cross-section approximation from the SAILOR library. Source locations for the adjoint analysis were chosen at positions along the inner diameter of the reactor vessel as well as at the center of each surveillance capsule. These calculations were also run in R, θ geometry to provide power distribution importance functions for exposure parameters of interest (neutron flux >1.0 MeV). The response of interest is then calculated as:

$$R_{R,\theta} = \int_R \int_\theta I(R,\theta) F(R,\theta) R dR d\theta$$

where

$R_{R,\theta}$ = Response of interest (ϕ(E>1.0MeV)) at radius R and azimuthal angle θ.

$I(R,\theta)$ = Adjoint importance function at radius R and azimuthal angle θ.

$F(R,\theta)$ = Full power fission density at radius R and azimuthal angle θ.

The calculated fast neutron exposure results are given in Tables II.2-1 through II.2-12 and in Figures II.2-1 through II.2-6 in WCAP-10962, (ref. 7). Measured fluence data from previously withdrawn surveillance capsules are additionally presented in WCAP-10962 for comparison with analytical results. The comparisons revealed that excellent agreement exists between the calculated and measured fast neutron fluence levels and are well within the uncertainty of the experimental results.

The reactor vessel material study consists of the best estimate copper and nickel chemical compositions of the reactor vessel belt line material needed for the calculation of RT_{PTS}. Material property values for the shell plates were derived from vessel fabrication test certificate results which have been docketed with the NRC, (ref. 5 in WCAP-10962). The weld property data, however, is not straight-forward as the shell plates in that the weldments are compounded with variabilities of copper concentrations. Babcock & Wilcox (B & W) performed a reactor vessel beltline weld chemistry study in which the results were reported in BAW-1799. In addition, the Westinghouse Owners Group (WOG) Reactor Vessel Beltline Region Weld Metal Data Base was also utilized for the material chemistry study. The statistical analysis evaluation resulted in average mean composition contents of 0.32 wt% copper and 0.56 wt% nickel, (section III of WCAP-10962).

Using both the neutron exposure and reactor vessel material evaluations, determination of RT_{PTS} values for all beltline region materials on both reactor vessels can be made. The calculation of RT_{PTS} is obtained by utilizing two equations:

Equation 1:

$$RT_{PTS} = I+M+[-10+470(Cu)+350(Cu)(Ni)]f^{0.270}$$

Equation 2:

$$RT_{PTS} = I+M+283f^{0.194}$$

where

I = initial reference transition temperature of unirradiated material measured as defined in ASME Code, NB-2331. If a measured value is not available, the following generic mean values must be used: 0°F for welds made with Linde 80 flux, and -56°F for welds made with Linde 0091, 1092, 124, and ARCOS B-5 weld fluxes.

M = margin to be added to cover uncertainties in the values of initial RT_{NDT}, copper and nickel content, fluence, and calculation procedures. In equation 1, M=48°F if a measured value of I was used, M=59°F if generic mean value of I was used. In equation 2, M=0°F if a measured value of I was used and M=34°F if the generic mean value of I was used.

Cu,Ni = Best estimate weight percent copper and nickel in the material.

f = Maximum neutron fluence, units of 10^{19} n/cm^2 (E $\geq$1MeV), at the clad-base-metal interface on the inside surface of the vessel at the location where the material in question received the highest fluence for the period of service in question.

The most limiting values at end-of-license (25.8 EFPY for Unit 1 and 25.3 EFPY for Unit 2) are 284°F for the circumferential weld (Unit 1) and 238°F for the longitudinal welds in the lower shell of Unit 2. At 32 EFPY, the most limiting value for Unit 1 is the circumferential weld which would be at or just below the NRC screening values and Unit 2 would have all its RT_{PTS} values below the screening values. These conclusions are based on using actual and projected fluence values. These values obtained for RT_{PTS} are dependent upon the fact that the present low leakage pattern fuel management will be used through the 32 EFPY period (section IV in WCAP-10962).

References, Appendix 5B

1. "PlMG, A one-dimensional multigroup Pl Code for the IBM-704," Bohl, H., Jr. et al, WAPD-TM-135 (1959).

2. "Radiation Damage Exposure and Embrittlement of Reactor Pressure Vessels," Shure K. Nuclear Applications, Vol. 2 (April 1966).

3. "SPIC-I, An IBM-704 Code to calculate the uncollided flux outside a right circular cylinder." Gillis P.A. et al, WAPD-TM-176 (1959).

4. "PDQ4, A program for the solution of the neutron diffusion equations in two dimensions on the Philco-2000," Cadwell, W.R., WAPD-TM-230 (1961).

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6. "AlM-5, A multigroup one dimensional diffusion equation code," Flatt, H.P. and Baller, D.C., NAA-SR-4694 (March 1960).

7. "Zion Units 1 and 2 Reactor Vessel Fluence and RT_{PTS} Evaluations," WCAP-10962 (December 1985).

8. "The Pressurized Thermal Shock Rule," U.S. Nuclear Regulatory Commission, 10CFR 50.34 (July, 1985).

APPENDIX 5C: INVESTIGATION OF INDICATIONS REVEALED BY ULTRASONIC BASE LINE INSPECTION OF ZION 1 REACTOR VESSEL 610-0144-51

NOTE: Portions of this document were retyped for clarity in the 1992 UFSAR Update.

APPENDIX 5C

INVESTIGATION OF INDICATIONS REVEALED BY ULTRASONIC BASE LINE INSPECTION OF ZION 1 REACTOR VESSEL 610-0144-51

REPORT NO. 7

PREPARED BY:

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G.A. WALTON

DATED 3/16/71

APPROVED BY (signature)
J.A. Van Meter

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INTRODUCTION

IN ACCORDANCE WITH A CHANGE NOTICE UNDER THE CONTRACT WITH WESTINGHOUSE ELECTRIC COMPANY, B&W PERFORMED A BASELINE EXAMINATION OF THE PRESSURE BOUNDARY WELDS USING ULTRASONIC TESTING AS DESCRIBED BY SECTION XI OF THE ASME CODE, ENTITLED "IN SERVICE INSPECTION OF REACTOR COOLANT SYSTEMS", AND WESTINGHOUSE REQUIREMENTS. AS A RESULT OF THIS EXAMINATION, THE CIRCUMFERENTIAL WELD, WR-16, BETWEEN THE MK 6 TRANSITION FORGING AND MK 5 LOWER HEAD WAS SUBJECTED TO FURTHER EXPLORATORY EXAMINATIONS.

THE FOLLOWING REPORT IS A REVIEW OF THE FABRICATION HISTORY, NON-DESTRUCTIVE TESTING, PROBING AND EVALUATION OF THE EXPLORED AREAS.

FABRICATION HISTORY

BASE MATERIALS

THE MK 5 LOWER HEAD IS FABRICATED FROM A-533 GRADE B, CLASS 1 CODE CASE-1339-2 MATERIAL PURCHASED FROM LUKENS STEEL COMPANY, HEAT NO. C-4007-2, WHICH WAS HOT PRESSED, QUENCHED AND TEMPERED AND ULTRASONICALLY EXAMINED AT THE BARBERTON WORKS OF THE BABCOCK AND WILCOX COMPANY.

THE MK 6 LOWER HEAD RING IS A-508-64, CLASS 2 MATERIAL, AS MODIFIED BY ASME SECTION III, ARTICLE 3 AND CODE CASE 1332-3 PURCHASED FROM MIDVALE-HEPPENSTALL COMPANY. THE HEAT NO. IS ZV-3779, BV-2847, FORGING NO. IS FV-3425. REQUIRED NON-DESTRUCTIVE TESTING INCLUDING ULTRASONIC EXAMINATION WAS PERFORMED BY THE VENDOR.

MAGNETIC PARTICLE INSPECTION OF THE MACHINED WELD PREPARATION OF THE MK 5 LOWER HEAD DISCLOSED FIVE DEFECTS 1 1/2" TO 4" LONG, 1/4" TO 5/8" WIDE, AND 1/8" TO 3/8" DEEP, AS SHOWN IN SKETCH NO. 1, THESE WERE REMOVED AND REPAIRS MADE PRIOR TO ASSEMBLY WITH THE MK 6 IN ACCORD WITH SPECIFICATION WS-69 REV. 5. TO MINIMIZE MISMATCH CONDITIONS BETWEEN THE MK 5 AND MK 6, THE LOWER HEAD WAS MACHINED TO A HEIGHT OF 35 15/32" IN LIEU OF THE PROCESS DIMENSION OF 36 59/64". THE ROOT GAP BETWEEN THE MK 5 AND MK 6 WAS INCREASED FROM 3/4" TO 1 11/16". WHEN ASSEMBLED AND ACCEPTED FOR WELDING, THE WELD GAP AND MISMATCH CONDITIONS WERE AS SHOWN ON SKETCH NO. 2.

FABRICATION HISTORY

WELDING OF MK 5 TO MK 6

THE INITIAL WELDING OF CIRCUMFERENTIAL WELD SEAM WR-16 WAS PERFORMED WITH THE SUBMERGED ARC PROCESS. WELDING WAS STARTED ON AUGUST 11, 1969, AND COMPLETED ON AUGUST 19, 1969. THE SURFACE OF THE LOW ALLOY WELD WAS THEN EXAMINED USING THE MAGNETIC PARTICLE METHOD AND ACCEPTED PRIOR TO CLADDING WITH THE SIX WIRE SUBMERGED ARC PROCESS.

THE BACKING STRAP AND ARCH BARS WERE REMOVED BY ARC AIR. THE ROOT OF THE WELD AND THE ARCH BAR AREAS WERE GROUND AND EXAMINED BY THE MAGNETIC PARTICLE METHOD. A 3:1 TAPER IN THE MISMATCH AREAS OF BOTH THE OD AND ID SURFACES WAS BUILT UP BY THE MANUAL METAL ARC PROCESS WITH ALL OF THE OPERATIONS BEING COMPLETED ON SEPTEMBER 11, 1969. PREHEAT WAS MAINTAINED FROM THE INITIATION OF WELDING THRU THE INTERMEDIATE POST WELD HEAT TREATMENT ON SEPTEMBER 26, 1969 IN FURNACE RUN Q 238 IN ACCORD WITH SPECIFICATION HT-11.

THE INITIAL WELD WAS RADIOGRAPHICALLY EXAMINED AND THE FOLLOWING DEFECTS WERE REPORTED:

DEFECT LOCATION RADIOGRAPHIC (RT) STATION NO.	LENGTH OF DEFECT	TYPE OF DEFECT
15-16	12"	SLAG
16-17	12"	"
17-18	12"	"
18-19	12"	"
19-20	2 1/2"	"
24-25	3/4"	"

FABRICATION HISTORY

SEQUENCE OF WELD REPAIRS

RT STATIONS 15-16, 16-17, 17-18, AND 18-19 WERE REPAIRED, RADIOGRAPHICALLY EXAMINED, AND ACCEPTED AFTER THE FIRST REPAIR CYCLE; RT STATION 24-25 AFTER THE SECOND REPAIR CYCLE; AND RT STATION 19-20 AFTER THE THIRD REPAIR CYCLE. THE REPAIR CYCLES ARE CORRELATED WITH THE REPAIR AREAS ON DRAWINGS MT. V 12071C SHEETS 2 OF 4 AND 3 OF 4. WELD REPAIRS WERE MADE IN CONFORMANCE WITH SPECIFICATION WS-69 REV. 5. INTERMEDIATE POST WELD HEAT TREATMENTS, MADE IN ACCORD WITH HT-11, ARE IDENTIFIED AS FURNACE RUNS HT-705, P675 AND Q254. THE REPAIR SEQUENCES WERE COMPLETED AND ACCEPTED ON NOVEMBER 6, 1969.

RADIOGRAPHIC EXAMINATIONS

ALL RADIOGRAPHIC EXAMINATIONS WERE PERFORMED IN ACCORDANCE WITH THE APPROVED SPECIFICATION AND THE ASME CODE, SECTION III. A COBALT 60, 1/4 X 1/4 IN. SOURCE SIZE, 100 CURIES, WITH A SOURCE TO FILM DISTANCE OF 90" AND KODAK AA FILM WERE USED. THE FILM WAS PLACED ON THE OUTSIDE OF THE LOWER HEAD ASSEMBLY WITH THE SOURCE LOCATED ON THE CENTERLINE OF THE ASSEMBLY. THE ORIGINAL WELD WAS RADIOGRAPHED USING A PANORAMIC EXPOSURE. RADIOGRAPHY OF REPAIRED AREAS WAS PERFORMED USING INDIVIDUAL EXPOSURES. THE ASME CODE ACCEPTANCE RADIOGRAPHS WERE REVIEWED AND ACCEPTED BY QUALIFIED B&W RADIOGRAPHERS C. WHITE AND C. MATHIS W. OWENS FOR WESTINGHOUSE AND R. MASON, THE AUTHORIZED CODE INSPECTOR AT THE MT. VERNON WORKS.

FABRICATION HISTORY

COMPLETION OF THE VESSEL

FABRICATION OF THE REACTOR VESSEL WAS CONTINUED IN ACCORD WITH THE PROCESS SHEETS AND CULMINATED IN THE FINAL POST WELD HEAT TREATMENT PERFORMED IN FURNACE RUN HT-791 ON OCTOBER 28, 1970.

THE HYDROSTATIC TEST WAS SUCCESSFULLY PERFORMED ON DECEMBER 15, 1970.

REVIEW OF PERTINENT RECORDS

A REVIEW OF ALL POST WELD HEAT TREATMENTS DISCLOSED NO DEVIATIONS FROM THE APPROVED HEAT TREATING PROCEDURES.

THE QUALIFICATION RECORDS OF ALL SUBMERGED ARC AND MANUAL METAL ARC WELDERS HAVE BEEN REVIEWED AND FOUND TO BE IN ORDER.

THE WELDING MATERIALS TESTS FOR THE WIRE/FLUX COMBINATION USED IN SUBMERGED ARC WELDING AS WELL AS THOSE FOR THE MANUAL METAL ARC ELECTRODES WERE REVIEWED AND FOUND TO BE ACCEPTABLE.

ATTACHMENTS "A" AND "B" PROVIDE THE WELDING RECORDS.

PREOPERATIONAL INSPECTION

THE BASE LINE INSPECTION OF THE REACTOR VESSEL AND CLOSURE HEAD WAS PERFORMED AFTER THE HYDROSTATIC TEST USING ULTRASONIC TESTING PROCEDURES IN ACCORDANCE WITH B&W SPECIFICATION BLI-1. (ATTACHMENT C) THIS INSPECTION INCLUDED ALL FULL PENETRATION PRESSURE BOUNDARY WELDS AND BASE METAL ADJACENT TO THE WELDS FOR A DISTANCE EQUAL TO THE THICKNESS OF THE PRESSURE PART AT THE WELD. THE INSPECTION WAS PERFORMED FROM THE INSIDE OF THE REACTOR VESSEL AND THE OUTSIDE OF THE CLOSURE HEAD SINCE FUTURE INSPECTIONS WILL BE MADE IN THIS WAY.

CALIBRATION PROCEDURE

THE CALIBRATION BLOCKS WERE FABRICATED TO THE REQUIREMENTS OF PARAGRAPH IX 343 OF THE 1968 EDITION OF ASME SECTION III. THE BLOCK WAS CLAD TO DUPLICATE THE REACTOR VESSEL INSIDE SURFACE. A SPERRY UM 721 REFLECTOSCOPE WAS COUPLED TO A 1" X 1" BRANSON Z, 45 DEGREE ANGLE BEAM TRANSDUCER. THE PRIMARY REFERENCE LEVEL USED WAS A DISTANCE AMPLITUDE CURVE CONSTRUCTED AS DESCRIBED IN B&W SPECIFICATION BLI-1.

INSPECTION RESULTS

TWENTY THREE INDICATIONS WHICH EXCEEDED 20% OF THE PRIMARY REFERENCE LEVEL WERE DETECTED IN THE MK 5 TO MK 6 WELD. EIGHT OF THESE INDICATIONS EXCEEDED THE 100% PRIMARY REFERENCE LEVEL. ALL TWENTY THREE INDICATIONS WERE RECORDED.

EVALUATION OF INSPECTION RESULTS

IN ACCORDANCE WITH PARAGRAPH IX 347 OF SECTION III AND IS 312 OF SECTION XI OF THE ASME CODE, THESE INDICATIONS WERE EVALUATED FURTHER TO DETERMINE THEIR SHAPE, IDENTITY AND LOCATION. THIS EVALUATION WAS ACCOMPLISHED AS FOLLOWS:

ULTRASONICS

THE TWENTY THREE RECORDABLE INDICATIONS FOUND BY THE 45 DEGREE SHEAR WAVE EXAMINATION WERE EVALUATED USING LONGITUDINAL AND VARIOUS SHEAR WAVE ANGLE BEAMS. IT WAS DETERMINED THAT THE 60 DEGREE SHEAR WAVE GAVE THE MAXIMUM RESPONSE. WITH THIS TECHNIQUE, INDICATIONS NUMBERED 9 THROUGH 20 WERE EVALUATED AS BEING NON FUSION. INDICATIONS NUMBER 1 THROUGH 8 AND INDICATIONS 22 AND 23 WERE EVALUATED AS SLAG. THE DIFFERENCE BETWEEN A SLAG OF NON FUSION TYPE INDICATION IS DETERMINED FROM THE PRESENTATION ON THE CATHODE RAY TUBE. THE NON FUSION TYPE INDICATION IS CHARACTERIZED BY A SHARP, HIGH AMPLITUDE SIGNAL WHICH WILL TRAVEL ALONG THE HORIZONTAL TIME SCALE OF THE CATHODE RAY TUBE AS THE SOUND WAVE TRAVERSES THE INDICATION. THE SLAG TYPE OF INDICATION PRESENTS A SMALL AMPLITUDE SIGNAL, BULBOUS IN NATURE, WHICH DOES NOT TRAVEL ALONG THE HORIZONTAL TIME AXIS SINCE THE SLAG PARTICLES ARE USUALLY ROUNDED AND OFFER ONLY A SINGLE POINT REFLECTOR SURFACE. IT SHOULD BE NOTED THAT THE DATA RECORDED ON DRAWING MT V 12071C SHEETS ONE THRU FOUR REPRESENT THE TRANSDUCER MOVEMENT AND NOT THE ACTUAL SIZE, WHICH WOULD BE SMALLER. INDICATIONS IDENTIFIED ABOVE AS SLAG ARE PERMITTED BY BLI-1 AND ASME SECTION III, PARAGRAPH N-625-3.

THEIR SIZE AND LOCATION HAVE BEEN RECORDED FOR FUTURE REFERENCE. INDICATIONS WHICH WERE EVALUATED AS NON FUSION HAVE BEEN REMOVED.

RADIOGRAPHY

THE ASME CODE SECTION III ACCEPTANCE RADIOGRAPHS OF THE MK 5 AND MK 6 WELD WERE AGAIN REVIEWED PAYING PARTICULAR ATTENTION TO THE AREAS IN WHICH THE TWENTY THREE RECORDABLE INDICATIONS WERE REPORTED. NONE OF THE INDICATIONS FOUND BY ULTRASONIC EXAMINATION WAS OBSERVED. BOTH INDIVIDUAL SHOTS AND PANORAMIC EXPOSURE RESOLVED THE 2 T HOLE IN THE PENETRAMETERS AS REQUIRED BY THE ASME CODE SECTION III. TO AID IN EVALUATING THE TWENTY THREE RECORDABLE INDICATIONS, ADDITIONAL RADIOGRAPHY WAS PERFORMED USING THE FOLLOWING SPECIAL METHODS, NONE OF WHICH ARE REQUIRED UNDER THE APPLICABLE PORTION OF THE ASME CODE.

METHOD "A"

SOURCE: 200 CURIES COBALT 60
FILM: KODAK AA
FILM LOCATION: OUTSIDE OF VESSEL

THIS TECHNIQUE WAS THE SAME AS THAT USED PREVIOUSLY EXCEPT THE SOURCE STRENGTH WAS 200 CURIES RATHER THAN 100 AND THE PLANE OF RADIATION WAS PARALLEL TO THE MK 5 FUSION ZONE, THUS PROVIDING THE MOST IDEAL GEOMETRY FOR FINDING NON FUSION IN THE MK 5 SIDE WALL OF THE WELD GROOVE.

METHOD "B"

SOURCE: 7-1/2 MEV LINEAR ACCELERATOR
FILM: KODAK TYPE M
FILM LOCATION: INSIDE OF THE VESSEL
PLANE OF RADIATION: CENTERLINE OF WELD

THIS TECHNIQUE WAS THE SAME AS THAT NORMALLY USED EXCEPT THAT TYPE M FILM WAS USED WHICH IS EXTRA FINE GRAINED.

METHOD "C" THE SAME AS "B" EXCEPT THE PLANE OF RADIATION WAS PARALLEL TO THE MK 5 FUSION ZONE, THUS PROVIDING THE MOST IDEAL GEOMETRY FOR FINDING NON FUSION I THE MK 5 SIDE WALL OF THE WELD GROOVE.

ALL OF THE FOREGOING SPECIAL METHODS WERE DESIGNED TO RESOLVE THE 1 T HOLE IN THE PENETRAMETER IN ADDITION TO THE 2 T HOLE REQUIRED BY THE ASME CODE SECTION III. EACH METHOD CONFIRMED THE CONDITION AT RADIOGRAPHIC STATION 14-15 WHICH, ON ULTRASONIC EXAMINATIONS, SHOWED A RESPONSE EXCEEDING THE 100% REFERENCE LEVEL.

IT SHOULD BE NOTED AT THIS POINT THAT, WHILE IT WAS POSSIBLE TO SHOW THE CONDITION AT STATION 14-15, THE METHODS EMPLOYED REQUIRED SPECIAL KNOWLEDGE, GAINED BY ULTRASONIC TESTING, WITH RESPECT TO THE INDICATIONS AND THEIR LOCATION. IT IS ALSO IMPORTANT TO NOTE THAT, EVEN WITH THIS KNOWLEDGE, IT WAS NOT POSSIBLE TO SHOW, BY RADIOGRAPHY, ALL OF THE INDICATIONS FOUND BY ULTRASONIC TESTING.

THE ABILITY OF RADIOGRAPHY TO DETECT NON FUSION TYPE INDICATIONS IS DEPENDENT UPON:

A. ORIENTATION OF THE INDICATION TO THE PLANE OF RADIATION I.E., THE THICKNESS RATIO AND GEOMETRY OF THE INDICATION WITH RESPECT TO TOTAL CROSS SECTION THICKNESS.

B. DEGREE OF SEPARATION.

THE EFFECT OF THE ABOVE FACTORS CAN BE EXPLAINED AS FOLLOWS: CONSIDER THAT A CERTAIN HOLE ON A PENETRAMETER IS VISIBLE ON THE RADIOGRAPH, A NON FUSION TYPE INDICATION OF THE SAME SEPARATION AND THICKNESS MAY NOT BE VISIBLE. THE PENETRAMETER HOLES HAVING SHARP BOUNDARIES, RESULT IN AN

ABRUPT, THOUGH SMALL CHANGE IN METAL THICKNESS WHEREAS THE NON FUSION TYPE INDICATION HAVING MORE OR LESS CURVED SURFACES CAUSES A GRADUAL CHANGE. THEREFORE, THE IMAGE OF THE PENETRAMETER HOLE IS SHARPER AND MORE EASILY SEEN IN THE RADIOGRAPH THAN IS THE IMAGE OF THE NON FUSION INDICATION. SIMILARLY A NON FUSION INDICATION MAY BE ORIENTED SUCH THAT AS THE X-RAYS (LINEAR ACCELERATOR) OR GAMMA RAYS (COBALT 60) PASS FROM THE SOURCE TO FILM ALONG THE THICKNESS OF THE INDICATION, ITS IMAGE ON THE FILM MAY NOT BE VISIBLE BECAUSE OF THE VERY GRADUAL TRANSITION IN PHOTOGRAPHIC DENSITY IN THE RADIOGRAPH.

FIGURE IX 333 (A) OF SECTION III OF THE ASME BOILER AND PRESSURE VESSEL CODE FOR NUCLEAR VESSELS GOVERNS THE RADIATION ENERGIES TO BE USED FOR RADIOGRAPHY. FOR EXAMPLE, 7.5 MEV (LINATRON) ENERGY SHOULD NOT BE USED FOR STEEL THICKNESS LESS THAN 2 1/4" WHEREAS THE MINIMUM THICKNESS FOR COBALT 60 IS 1.5" AND FOR IRIDIUM 192 1". THE UPPER LIMIT FOR IRIDIUM 192 IS 3" AND FOR COBALT 60 11". THE UPPER LIMITS ARE PRIMARILY DICTATED BY EXPOSURE TIME AND ARE NOT MANDATORY.

FILM DENSITY, SOMETIMES REFERRED TO AS PHOTOGRAPHIC DENSITY TO DISTINGUISH IT FROM THE MASS DENSITY OF THE OBJECT BEING RADIOGRAPHED, IS THE DARKENING OF THE RADIOGRAPH CAUSED BY EXPOSURE TO RADIATION AND IS AN IMPORTANT ATTRIBUTE AFFECTING RADIOGRAPHIC CONTRAST WHICH IS DEFINED AS THE PHOTOGRAPHIC DENSITY DIFFERENCES FROM ONE AREA TO ANOTHER ON THE RADIOGRAPH.

EVALUATION OF INSPECTION RESULTS

MINIMUM PHOTOGRAPHIC DENSITY VALUES ARE SPECIFIED IN ORDER TO OBTAIN A GRADIENT GREATER THAN 1.0 AS DETERMINED FROM A CHARACTERISTIC CURVE, WHERE PHOTOGRAPHIC DENSITY IS PLOTTED AGAINST THE LOGARITHM RELATIVE EXPOSURE, SOMETIMES REFERRED TO AS A H&D CURVE.

IF THE GRADIENT IS GREATER THAN 1.0, THE FILM ACTS AS A CONTRAST AMPLIFIER WHICH IS OF UTMOST PRACTICAL IMPORTANCE, SINCE OTHERWISE, MANY SMALL DIFFERENCES IN THE OBJECT BEING RADIOGRAPHED COULD NOT BE MADE VISIBLE ON THE RADIOGRAPH.

FOR THIS REASON, FILM DENSITY VALUES ARE SPECIFIED AS BEING 2.0 MINIMUM THROUGH THE WELD FOR SINGLE FILM VIEWING AND 2.6 FOR COMPOSITE VIEWING OF DOUBLE FILM EXPOSURES, WITH EACH FILM OF A COMPOSITE SET TO HAVE A MINIMUM DENSITY OF 1.0.

RADIOGRAPHIC SENSITIVITY IS A QUALITATIVE TERM REFERRING TO THE SIZE OF THE SMALLEST DETAIL THAT CAN BE SEEN IN A RADIOGRAPH OR THE EASE WITH WHICH THE IMAGES OF SMALL DETAILS CAN BE DETECTED AND IS DEPENDENT ON THE COMBINED EFFECTS OF TWO INDEPENDENT SETS OF FACTORS. ONE IS RADIOGRAPHIC CONTRAST AND THE OTHER IS DEFINITION, THE ABRUPTNESS OR SMOOTHNESS OF THE DENSITY TRANSITION.

EQUALLY IMPORTANT, IS THE GEOMETRY OF THE WELD. FOR EXAMPLE, A CIRCULAR SEAM HAVING A WELD PREPARATION WITH 7° SLOPE ON THE SIDE WALLS AND THE WIDE PART OF THE GROOVE ON THE OUTSIDE, WOULD BE MOST FAVORABLY RADIOGRAPHED FROM A GEOMETRIC STANDPOINT USING A SOURCE PLACED ON THE CENTERLINE OF THE VESSEL IN LINE WITH THE WELD SEAM. THE RADIATION IN THIS CASE WOULD BE PASSING THROUGH THE WELD RADIALLY WITH THE BEAM SPREAD

CLOSELY PARALLELING THE SLOPE OF THE WELD GROOVE SIDE WALLS. THIS IS IDEAL AS REGARDS GEOMETRY. THE SOURCE USED IN SUCH A SITUATION WOULD THEREFORE BE EITHER COBALT 60 OR IRIDIUM 192 BASED UPON THE THICKNESS INVOLVED. ON THE OTHER HAND, THE SAME WELD CAN BE MEANINGFULLY RADIOGRAPHED USING A LINAC WITH THE FILM PLACED INSIDE THE VESSEL. IN THIS CASE THE ADVANTAGES WOULD BE (1) SMALLER FOCAL SPOT AND (2) LARGE QUANTITY OF RADIATION ENABLING THE USE OF A LONG SOURCE TO FILM DISTANCE AND FINE GRAIN FILM.

THE ABILITY TO OBTAIN A REQUIRED LEVEL OF RADIOGRAPHIC SENSITIVITY IS BASED ON THE TOTAL RADIOGRAPHIC SYSTEM.

A. RADIOACTIVE ENERGY LEVELS
B. PHYSICAL SOURCE SIZE
C. TYPE FILM
D. SOURCE TO FILM DISTANCE
E. EXPOSURE TIME
F. GEOMETRY OF THE WELD
G. LOCATION OF THE SOURCE WITH RELATION TO THE WELD
H. TYPE OF SCREENS AND FILTERS USED
I. FILM DEVELOPMENT

RADIATION ENERGY LEVEL ALONE IS NOT THE CONTROLLING FACTOR IN OBTAINING A SPECIFIED SENSITIVITY LEVEL.

IN ADDITION, THE PERSONNEL WHO APPLY THE RADIOGRAPHIC PROCEDURES ARE TRAINED AND QUALIFIED IN ACCORDANCE WITH THE CODE SPECIFIED DOCUMENTS SNT-TC-1A.

EVALUATION OF INSPECTION RESULTS

THE RADIOGRAPHY PERFORMED USING COBALT 60 FOR THE CODE ACCEPTANCE EXAMINATIONS OF THE MK 5 TO MK 6 WELD COMPLIED WITH ALL OF THE CRITERIA SPECIFIED IN THE CODE AND SHOULD NOT BE CONSIDERED MARGINAL OR LIMITED FOR THIS APPLICATION.

ALL OF THE RADIOACTIVE SOURCES AND THE VARIOUS ENERGIES PERMITTED BY THE CODE ARE NEEDED TO EXAMINE THE VARIOUS WELDMENTS AND GEOMETRIES ENCOUNTERED IN THE FABRICATION OF PRESSURE VESSELS.

COLLATION OF ULTRASONIC TEST RESULTS WITH FABRICATION HISTORY

DRAWING MT. V 12071C SHEETS 1 THRU 4 SHOW THE LOCATION OF THE TWENTY THREE RECORDABLE INDICATIONS FOUND BY ULTRASONIC EXAMINATION AS RELATED TO THE RADIOGRAPHIC STATIONS. FURTHER, THE DRAWINGS ILLUSTRATE THE PERTINENT DIMENSIONS AND LOCATION OF REPAIRS MADE DURING FABRICATION EXCEPT THOSE MADE TO THE WELD PREPARATION OF THE MK 5. FROM THIS COLLATION, IT IS APPARENT THAT ALL INDICATIONS EVALUATED AS NON FUSION BY ULTRASONICS WERE ASSOCIATED WITH AREAS WHICH HAVE UNDERGONE REPAIR.

IT SHOULD BE NOTED THAT ALL RECORDED DATA WERE INITIALLY PLOTTED BASED ON THE ULTRASONIC DATA OBTAINED FROM THE CLAD SURFACE. VARIABLES SUCH AS LOWER HEAD CONTOUR, CLADDING INTERFACE WITH BASE METAL, IRREGULAR CLAD SURFACES AFFECT ACCURATE RADIAL LOCATION CAPABILITY. FURTHER ULTRASONIC VERIFICATION WAS PERFORMED FROM THE OUTSIDE SURFACE OF THE LOWER HEAD TO ESTABLISH THE RADIAL AND DEPTH LOCATION OF THE INDICATIONS. THE INSPECTION RESULTS FROM THE OUTSIDE SURFACES SHOWED THAT ALL INDICATIONS CLASSIFIED AS NON FUSION WERE ASSOCIATED WITH THE MK 5 FUSION BOUNDARY.

METALLURGICAL INVESTIGATION

THE INDICATIONS WERE EXPOSED FOR EXAMINATION BY THE ARC AIR PROCESS AND CHECKED FOR DEPTH; A COMPILATION OF RESULTS BEING SHOWN IN TABLE 1. THE RESULTANT CAVITIES AT RADIOGRAPHIC STATIONS 14-15 AND 20 WERE GROUND AND ETCHED WITH A 10% NITAL SOLUTION IN ORDER TO ESTABLISH THE POSITION WITH RESPECT TO THE WELD SEAM, HEAT AFFECTED ZONE AND BASE METAL. METALLOGRAPHIC SAMPLES WERE REMOVED FROM RADIOGRAPHIC STATIONS 14-15 AND 15-16 FOR DETERMINATION OF THE TYPE AND ORIGIN OF THE INDICATIONS.

PROBING TECHNIQUE

THE PROBE LOCATIONS WERE CAREFULLY LAID OUT ON THE OUTSIDE SURFACE OF THE VESSEL USING AN ULTRASONIC PLOT OF THE INDICATION POSITIONS. EACH AREA WAS THEN PROBED USING THE ARC AIR PROCESS, MAGNETIC PARTICLE INSPECTION BEING PERFORMED BETWEEN CUTS UPON REACHING 3/4" ABOVE THE REPORTED DEPTH POSITION. AFTER INITIAL EXPOSURE OF THE INDICATION IN THE CAVITY, THE TOTAL LENGTH WAS EXPLORED BY PROGRESSIVELY EXTENDING THE CAVITY. AT RADIOGRAPHIC STATION 20 A CAVITY WAS CUT BETWEEN AND BELOW TWO REPORTED NON FUSION AREAS, EXPOSING CROSS SECTIONS OF BOTH AREAS IN THE END WALLS OF THE CAVITY.

EXAMINATION OF CAVITY BETWEEN RADIOGRAPHIC STATIONS 14-15

FIGURE 1 SHOWS THE INDICATION AS REVEALED BY THE MAGNETIC PARTICLE TEST, THE INSPECTION BEING PERFORMED AFTER GRINDING AND LIGHTLY ETCHING THE

CAVITY. FIGURE 2 SHOWS A MORE DEEPLY ETCHED VIEW OF THE CAVITY, WHICH ILLUSTRATES THAT THE INDICATION IS ASSOCIATED WITH THE MK 5 FUSION BOUNDARY OF THE SUBMERGED ARC WELD. THE LIGHT ETCHING ZONE ADJACENT TO THE SUBMERGED ARC WELD BOUNDARY REVEALS THE PRESENCE OF ANOTHER WELD AREA, WHICH INDICATES THAT A REPAIR HAD BEEN MADE TO THE ORIGINAL WELD PREPARATION SURFACE AT THIS LOCATION.

SAMPLES FOR CHEMICAL ANALYSIS WERE TAKEN WITH A CHIPPING TOOL FROM BOTH THE SUBMERGED ARC DEPOSIT AND THE APPARENT WELD PREPARATION REPAIR. THE RECORDED ANALYSIS FOR THESE SAMPLES WAS AS FOLLOWS:

TABLE 2

SAMPLE	% C	% Mn	% Si	% Ni	%Mo	% S	% P	% Cu
SUBMERGED ARC	0.166	1.40	0.35	0.65	0.40	0.015	0.018	0.19
WELD REPAIR	0.194	1.40	0.29	0.58	0.40	0.013	0.008	0.11

EXAMINATION OF CAVITY AT RADIOGRAPHIC STATION 20

FIGURE 3 SHOWS A VIEW OF THE ETCHED CAVITY, THE INDICATIONS BEING HIGHLIGHTED BY MAGNETIC PARTICLE TESTING. BASE METAL LAMINAR TYPE INDICATIONS CAN BE SEEN AT BOTH ENDS OF THE CAVITY, THERE BEING NO EVIDENCE OF NON FUSION. SHORT CURVED CRACKS WERE OBSERVED AT THE ENDS OF THE LAMINAR TYPE CONDITION, RUNNING OBLIQUELY THROUGH THE HEAT AFFECTED ZONE TO THE FUSION BOUNDARY OF THE WELD.

METALLOGRAPHIC EXAMINATION

SAMPLE 1 - RADIOGRAPHIC STATION 14-15

SAMPLE 1 WAS REMOVED BY THE ARC AIR PROCESS AT A DISTANCE OF 2" FROM

RADIOGRAPHIC STATION 14. FIGURES 4A AND 4B (X 2 1/2) SHOW A SEGMENT OF THIS SAMPLE WITH THE INDICATION FACES IN THE PARTIALLY CLOSED AND OPEN POSITIONS. NO EVIDENCE WAS PRESENT TO SUPPORT A LOW TEMPERATURE CRACKING MECHANISM. THE SMOOTH OXIDIZED SURFACES ARE TYPICAL OF A NON FUSION CONDITION. FIGURES 5A AND 5B (X 2 1/2) SHOW CROSS SECTIONS FROM TWO POSITIONS WITHIN THE SAMPLE. THE ARC AIR CUTTING OPERATION HAS THERMALLY AFFECTED A WIDE ZONE ON BOTH SIDES OF THE SAMPLE, DESTROYING THE ORIGINAL METALLURGICAL STRUCTURE IN THE SUBJECT AREAS. WHILE THE REPAIR WELD CAN STILL BE DIFFERENTIATED FROM THE SUBMERGED ARC DEPOSIT, NO STRUCTURAL DETAILS REMAINED TO ALLOW POSITIVE IDENTIFICATION THAT IT WAS A MANUAL METAL ARC DEPOSIT.

SAMPLE 2 - RADIOGRAPHIC STATION 15-16

SAMPLE 2 WAS TAKEN AT A DISTANCE OF 2 5/8" FROM RADIOGRAPHIC STATION 16 USING BOTH A CUTTING DISC AND THE ARC AIR PROCESS. MACRO EXAMINATION REVEALED NON FUSION, AS SHOWN IN FIGURES 6 (X 2 1/2) AND 7 (X 11), LOCATED ON THE BOUNDARY BETWEEN SUBMERGED ARC WELD METAL AND TWO BEADS OF A MANUAL METAL ARC WELD DEPOSIT. THE MANUAL METAL ARC BEAD PROFILES WOULD SUGGEST THAT THEY WERE DEPOSITED IN A REPAIR CYCLE TO THE SUBMERGED ARC WELD.

EXAMINATION AT HIGHER MAGNIFICATION OF THE SUBMERGED ARC DEPOSIT INDICATED A SLIGHT INCREASE IN CARBON CONTENT IN THE IMMEDIATE VICINITY OF THE INDICATION, AS SHOWN IN FIGURES 8A AND 8B (X 300). A LIGHT GREY PHASE WAS PRESENT IN THE INDICATION AT INTERVALS ALONG ITS ENTIRE LENGTH.

METALLURGICAL INVESTIGATION

ELECTRON PROBE MICRO-ANALYSIS OF THIS PHASE AT B&W AGLLIANCE RESEARCH CENTER HAS DETECTED A HIGH CONCENTRATION OF IRON AND STRONG EMISSIONS FOR POTASSIUM, SILICON AND TITANIUM. IT IS CONCLUDED THAT THE PHASE CONSISTS OF A MIXTURE OF BOTH SLAG FROM MANUAL METAL ELECTRODES AND IRON OXIDES, WHICH FURTHER SUBSTANTIATES THAT THE DEFECT ORIGINATED DURING MANUAL METAL ARC REPAIR WELDING OF THE SUBMERGED ARC WELD JOINT.

DISCUSSION

METALLOGRAPHIC EVALUATION OF THE INDICATION LOCATED IN RADIOGRAPHIC STATION 14-15 WAS LIMITED BY THE STRUCTURAL CHANGES IN THE INDICATION REGION, INDUCED BY ARC AIR CUTTING. EXAMINATION OF THE SURFACES SHOWED NO EVIDENCE, HOWEVER, OF A LOW TEMPERATURE CRACK PROPAGATION MECHANISM, THE SURFACE CONDITION BEING TYPICAL OF NON FUSION INDICATIONS. MACRO ETCHING OF THE CAVITY SHOWS THAT THE INDICATION WAS LOCATED ON THE BOUNDARY OF THE SUBMERGED ARC WELD AND A REPAIR TO THE MK 5 WELD PREPARATION.

NON FUSION CAN OCCUR WITH ANY WELDING PROCESS. IN THE MAJORITY OF CASES WHEN USING THE TWO WELDING PROCESSES, SUBMERGED ARC OR MANUAL METAL ARC, NON FUSION RESULTS FROM INCORRECT BEAD PLACEMENT OR IRREGULARITIES IN THE SURFACE OF THE WELD PREPARATION. FROM THE AVAILABLE EVIDENCE IT WAS NOT POSSIBLE TO POSITIVELY IDENTIFY THE RESPONSIBLE WELDING PROCESS. THE CARBON VALUES REPORTED IN TABLE 2 COULD BE ATTRIBUTED TO EITHER OF THE FOLLOWING:

1. INADEQUATE GRINDING OF THE CAVITY AFTER ARC AIR GOUGING TO REMOVE ALL TRACES OF CARBON ENRICHED MATERIAL.
2. OVERLAPPING OF SAMPLING INTO ADJACENT BASE METAL WITH CONSEQUENT CARBON PICK UP, DUE TO THE NARROW REPAIR AREA.

EXAMINATION OF THE REPAIR RECORDS FOR RADIOGRAPHIC STATION 15-16, SHOWS THAT A CAVITY WAS CUT FROM THE INSIDE SURFACE OF THE SUBMERGED ARC WELD TO A DEPTH OF 4 9/16" AND REPAIRED BY THE MANUAL METAL ARC PROCESS. THE NON FUSION CONTAINED IN SAMPLE 2 WAS DISCOVERED AT A DEPTH OF 2 1/8" FROM THE

DISCUSSION

OUTSIDE SURFACE OF THE SUBMERGED ARC WELD, WHICH WOULD CORRESPOND WITH THE BOTTOM OF THE REPAIR CAVITY.

THE METALLORGRAPHIC DATA CONFIRMS THAT THE NON FUSION WAS IN FACT LOCATED ALONG THE BOUNDARY OF A MANUAL METAL ARC WELD, THE SLIGHT CARBON ENRICHMENT IN THE ADJACENT SUBMERGED ARC WELD INDICATING PRIOR ARC AIR GOUGING. THE PRESENCE OF ELECTRODE COATING ELEMENTS IN THE INDICATION FURTHER SUPPORTS THAT IT ORIGINATED DURING MANUAL METAL ARC REPAIR WELDING OF THE SEAM WHICH IS ATTRIBUTED TO INCORRECT BEAD PLACEMENT. ALTHOUGH THE INDICATIONS BETWEEN RADIOGRAPHIC STATIONS 19-21 WERE EVALUATED BY ULTRASONICS AS NO FUSION, SUBSEQUENT PROBING PROVED THEM TO BE LAMINAR INDICATIONS IN THE MK 5 BASE MATERIAL. THE CONCENTRATED EFFECTS OF THE WELDING STRESSES AT THE END OF THE CODE ACCEPTABLE LAMINAR INDICATIONS RESULTED IN PROPAGATION OF SHORT CRACKS TO THE WELD FUSION BOUNDARY. IT IS BELIEVED THAT THESE CRACKS ACTED AS A FOCUSSING REFLECTOR FOR THE ULTRASONIC WAVES, PRODUCING A HIGH AMPLITUDE SIGNAL CHARACTERISTIC OF A CRACK OR OF NON FUSION.

CORRECTIVE ACTION

REPAIR OF ZION I VESSEL

THE CAVITIES RESULTING FROM THE REMOVAL OF INDICATIONS EVALUATED AS NON FUSION ARE DEPICTED IN DRAWING MT V 12071 SHEET 5. THE REPAIR OF THESE CAVITIES IS OUTLINED IN ATTACHMENT D.

AFTER LOCAL STRESS RELIEF THE FOLLOWING NON DESTRUCTIVE TESTS WERE PERFORMED:

A MAGNETIC PARTICLE EXAMINATION (Y-5 YOKE) WAS MADE OF THE ENTIRE OUTSIDE SURFACE OF THE WR-16 WELD AND WAS ACCEPTED ON FEBRUARY 9, 1971. A 100% LIQUID PENETRANT INSPECTION (SOLVENT REMOVABLE) WAS PERFORMED ON THE WR-33 BACKCLADDING AND ACCEPTED ON FEBRUARY 11, 1971. THE WR-16 WELD WAS RE-INSPECTED FROM THE INSIDE SURFACE BY ULTRASONICS IN ACCORDANCE WITH B&W SPECIFICATION BLI-1. NO REJECTABLE INDICATIONS WERE FOUND, THE WELD BEING ACCEPTED ON FEBRUARY 10, 1971. A RADIOGRAPHIC EXAMINATION WAS PERFORMED OF THE SEGMENT OF THE SEAM WHICH HAD BEEN WELD REPAIRED, USING A 12 MEV LINAC UNIT AND KODAK AA FILM. EXPOSURES WERE TERMINATED AT ONE 12" RADIOGRAPHIC STATION PAST THE ENDS OF THE REPAIRED REGION. NO REJECTABLE INDICATIONS WERE FOUND, THE WELD BEING ACCEPTED ON FEBRUARY 11, 1971.

IMPROVEMENT OF WELD PERFORMANCE

THE SEAM BETWEEN THE MK 5/6 WAS WELDED IN THE PERIOD AUGUST, 1969 TO NOVEMBER, 1969. SINCE THAT TIME, B&W HAS ADDED EXPERIENCED AND SKILLED SUPERVISORY PERSONNEL IN THE WELDING GROUPS. FURTHER, ALL WELDING

CORRECTIVE ACTION

OPERATORS HAVE RECEIVED ADDITIONAL TRAINING WITH PARTICULAR EMPHASIS BEING PLACED ON THE IMPORTANCE OF BEAD SEQUENCE AND PREPARATION OF CAVITIES.

THE EFFECTIVENESS OF THE ABOVE ACTIONS IS DEMONSTRATED BY A PROGRESSIVE IMPROVEMENT IN THE WELD DEFECT RATINGS FOR BOTH MANUAL METAL ARC AND SUBMERGED ARC WELDING. THE IMPROVEMENT IN PERFORMANCE IS ILLUSTRATED AS FOLLOWS:

> THE MANUAL METAL ARC DEFECT RATING FOR THE MONTH OF AUGUST 1969 WAS 2.78%. THE RATING WAS REDUCED TO 0.32% BY DECEMBER 1970. SIMILARLY, THE SUBMERGED ARC DEFECT RATING OF 4.7% WAS REDUCED TO 0.25% FOR THE SAME PERIOD.

THE OVERALL IMPROVEMENT IN PERFORMANCE IS EVEN GREATER THAN INDICATED BY THE ABOVE PERCENTAGES. THE ORIGINAL CONCEPT OF THE WELD DEFECT RATING PROGRAM WAS BASED ON RADIOGRAPHIC INSPECTION RESULTS ONLY. THE PRESENT PROGRAM, WHICH WAS INITIATED IN APRIL, 1970 NOW INCLUDES THE RESULTS OF ALL NON DESTRUCTIVE TEST METHODS IN THE RATING EVALUATION.

AN AUDIT OF THE WELDING OPERATIONS HAS SHOWN THE PRACTICES OUTLINED ABOVE ARE BEING FOLLOWED. THIS MONITORING PROGRAM WILL BE CONTINUED AND AN EFFORT MADE TO FURTHER REDUCE THE INCIDENCE OF WELD DEFECTS.

ATTACHMENT A

LISTING OF WELDERS AND WELDING MATERIALS WHICH PERTAIN TO THE MK 5/6 WELD

LISTING OF WELDERS AND WELDING MATERIALS

DATE WELDED	WELDER'S NAME/NO.	WELD PROCESS	WELD DATA SHEET	WIRE HT. NO. OR P-ORDER NO.	FLUX LOT NO.
8-11-69	1174 CHANDLER	S/A	WR-16, ALT.1	299L44	8650
8-11-69	1149 EVANS		R-0		
8-12-69	1112 GARRETT				
	1559 BAILEY				
	1328 WHIPPLE				
	1305 SKIPWORTH				
	1459 EVANS				
8-12-69	1174 CHANDLER				
8-13-69	1112 GARRETT				
	1559 BAILEY				
	1387 DUVALL				
	1328 WHIPPLE				
	1184 CHANDLER				
8-13-69	1459 EVANS				
8-14-69	1112 GARRETT				
	1559 BAILEY				
	1387 DUVALL				
	1328 WHIPPLE				
	1174 CHANDLER				
8-14-69	309 NAAS				
8-15-69	1112 GARRETT				
	1559 BAILEY				
	1387 DUVALL				
	1328 WHIPPLE				
	1174 CHANDLER				
8-15-69	1459 EVANS				
8-16-69	1112 GARRETT				
	1559 BAILEY				
	1328 WHIPPLE				
	1333 WILSON				
	309 NAAS				
8-16-69	1041 CARRIER				
8-18-69	1559 BAILEY				
	1387 DUVALL				
	1328 WHIPPLE				
	1459 EVANS				
8-18-69	1174 CHANDLER				
8-19-69	1559 BAILEY				
9-19-69	ORIGINAL WELD	COMPLETE			

LISTING OF WELDERS AND WELDING MATERIALS

A5-144-1 C/S

DATE WELDED	WELDER'S NAME/NO.	WELD PROCESS	WELD DATA SHEET	WIRE HT. NO. OR P-ORDER NO.	FLUX LOT NO.
9-9-69	1299 TROUT	MMA	REPAIR TO WS 69-5	818-026349	NOT APPLICABLE
	1287 EARLY				
9-9-69	1192 COX				
9-10-69	1296 SINGER				
	1141 MEYER				
	1080 STALLINGS				
	1110 BAYLER				
	1202 DOUGLAS				
	1417 MINTON				
	1290 MULVEY				
9-10-69	1382 BOWLING				
9-11-69	1261 ROBISON				
	1188 BOULTINGHOOSE				
	1417 MINTON				
	1110 BAYLER				
	1290 MULVEY				
	1226 CARMON				
9-11-69	1436 FUGATE				
9-12-69	1024 ODLE				
	1081 ROMERHAUSEN				
9-12-69	1290 MULVEY			818-026349	
9-15-69	1081 ROMERHAUSEN			818-026224	
9-18-69	1250 INGRAM			818-026349	
9-18-69	1301 BRISCOE			818-026349	
10- 8-69	1192 COX			818-026222	
10- 8-69	1234 BROWNING				
10- 9-69	1024 ODLE			818-026222	
	1296 SINGER			495V3011	
	1110 BAYLER				
	1417 MINTON				
	1233 PATRICK				
10- 9-69	1234 BROWNING				
10-10-69	1024 ODLE				
	1447 CROWDUS			495V3011	
10-10-69	1110 BAYLER			818-026222	
10-22-69	1299 TROUT				
10-22-69	1409 FRITTS				
	1274 DONNER				
	1243 PHILLIPS				
10-22-69	1234 BROWNING			818-025187	
10-23-69	1409 FRITTS			818-025187	
10-23-69	1206 BRAUSER			818-026223	
10-29-69	1290 MULVEY			818-025187	
10-29-69	1223 PATRICK	MMA	WS 69-5	818-026223	N/A

LISTING OF WELDERS AND WELDING MATERIALS

A5-144-1 C/S

DATE WELDED	WELDER'S NAME/NO.	WELD PROCESS	WELD DATA SHEET	WIRE HT. NO. OR P-ORDER NO.	FLUX LOT NO.
10-30-69	1351 GRABERT	MMA	WS 69-5	(818-025187 (818-026223	N/A
10-31-69	1261 ROBISON			818-026223	
10-31-69	1234 BROWNING			818-026223	
11- 1-69	1198 BARTOK			818-025187	
11- 1-69	1441 McDOWELL			818-025187	
11- 2-69	1271 STANKOVICH			818-025187	
11- 2-69	1351 GRAGERT	MMA	WS 69-5	818-025187	N/A

LISTING OF WELDERS
AND WELDING MATERIALS

5-152-1 W/P

DATE WELDED	WELDER'S NAME/NO.	WELD PROCESS	WELD DATA SHEET	WIRE HT. NO. OR P-ORDER NO.	FLUX LOT NO
6-25-69	1228 PUCKETT	MMA	WS 69-5	818-021736	N/A
6-25-69	1268 LITTON	MMA	WS 69-5	818-021736	N/A
7-24-69	1296 SINGER	MMA	WS 69-5	818-026224	N/A

ATTACHMENT B

WELDING PROCEDURE QUALIFICATION

AND WELDING MATERIALS TEST REPORTS

WHICH PERTAIN TO THE MK 5/6 WELD

THE BABCOCK & WILCOX COMPANY
BARBERTON, OHIO

CONTRACT NO. 610-0112
SPECIFICATION NO. W-21; WS-30

RECORD OF PROCEDURE AND/OR OPERATOR QUALIFICATION TEST-QC 2E4-122

NAVSHIPS 250-1500-1	ASME SECTION 8	ASME SECTION 1 & 8	OTHER	PLATE	PIPE		SINGLE PASS	MULTIPLE PASS
[X]	[X]	[ ]	[ ]	[X]	[ ]	[ ]	[ ]	[X]

WELDING PROCESS	QUALIFICATION POSITION	METAL THICKNESS	SINGLE LAYER	MULTIPLE LAYER
Automatic Submerged Arc	Flat	6-1/4 IN.	[ ]	[X]

MATERIAL SPECIFICATION	SINGLE ARC	MULTIPLE ARC
Base material in quenched & tempered condition B-3 (SA-302-B)	[X]	[2]

HEAT TREATMENT	MACRO. EXAM.	MICRO. EXAM.
50 hours at 1100° - 1150°F	N.R.	N.R.

FLUX NAME OR COMPOSITION	PREHEAT TEMPERATURE	INTERPASS TEMPERATURE	WELDER SYMBOL CLOCK NO.
Linde #80 48XD	300 °F MIN.	500 °F MAX.	Nixon 1167

TYPE OF BACKUP (STRIP OR GAS) AND COMPOSITION	FILLER METAL GROUP NO.	SHIELDING GAS	CUP SIZE	TORCH GAS FLOW RATE
B-3 material	W-1-B	N.A.	N.A.	N.A. CFH

AMPS, VOLTS, CURRENT, POLARITY
Alternating current — Single arc 32-35 Volts 500-550 Amps; Tandem arc 33-36 Volts 500-600 Amps

SIZE OF ELECTRODE, IN. DIA. / SIZE OF FILLER WIRE, IN. DIA.	ELECTRODE	TRAVEL SPEED IPM *	WIRE FEED IPM	OSCILLATION	DWELL TIME SEC
1/8	spacing 3/4"	10-12 & 20	N.A.	N.A. CY/PM	N.A.

LIQUID PENETRANT	MAGNETIC PARTICLE	RADIOGRAPH	ULTRASONIC TEST
N.A.	Acceptable	Acceptable	N.R.

REMARKS: Procedure Qualification for welding B-3 material in the flat position by the auto Submerged Single & tandem arc process using 1/8" dia. (U.S.Steel) Mn-Mo-Ni filler wire with Linde #80 48XD flux. See reverse side for groove configuration.

CHEMICAL ANALYSIS - % E NO. 59923

LOCATION	C	MN	P	S	SI	CR	NI	MO	FE	CU	CB	CO	TA	TI	AL
Weld	.09	1.46	.024	.020	.50	.07	.53	.43		.19					
Weld	.09	1.42	.023	.019	.49	.07	.53	.43		.16					

REDUCED SECTION TENSILE (TRANSVERSE TO WELD)

SPECIMEN NO.	DIMENSIONS, INCHES WIDTH	THICKNESS	AREA SQ. IN.	ULTIMATE LOAD LBS.	ULTIMATE TENSILE STRENGTH PSI	FRACTURE LOCATION
PQ-1687	.997	3.000	2.991	254,000	89,920	Weld
PQ-1687	.996	2.765	2.754	232,500	84,420	Weld
PQ-1687	.996	2.627	2.616	224,000	85,640	Weld
PQ-1687	1.000	2.677	2.677	224,500	83,840	Weld

BEND TEST

Four side bends - Acceptable (No defects)

ALL WELD METAL TENSILE

SPECIMEN NO.	DIAMETER, IN.	AREA SQ. IN.	YIELD POINT PSI	TENSILE STR. PSI	ELONG % IN 2"	RED AREA %
			N.R.			

TYPE CHARPY V-NOTCH IMPACT TEST AT +10 °F 240 FT. LBS. ENERGY LOAD

BASE METAL	42.	43.	32	FT.LBS.	Weld Metal 1/4T	40.	70.	32	FT.LBS.
WELD METAL Surface	44.	44.	43	FT.LBS.	Weld Metal 1/4T	41.	38.	49	FT.LBS.
HEAT AFFECTED ZONE	27.	35.	39	FT.LBS.					FT.LBS.

WE CERTIFY THAT TO THE BEST OF OUR KNOWLEDGE THE STATEMENTS MADE IN THIS RECORD ARE CORRECT AND THAT THE TEST WELDS WERE PREPARED, WELDED AND TESTED IN ACCORDANCE WITH THE APPLICABLE SPECIFICATIONS.

N.R. = NOT REQUIRED N.A. = NOT APPLICABLE

* See reverse side

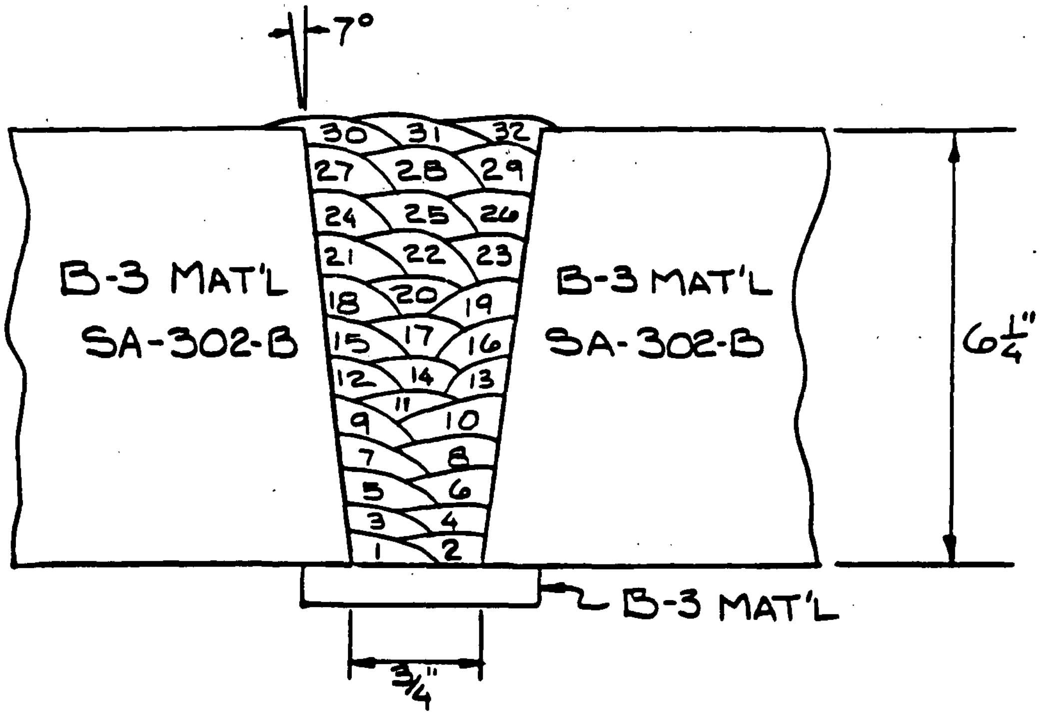
MTV-1
PQ 1687

WITNESSED R. Wagner D.C.A.S.O.

BY Paul E Campbell
BABCOCK & WILCOX COMPANY

DATE Sept. 18, 1967

AUTO SUBMERGED ARC (A.C.)



AUTO SUBMERGED ARC (A.C.)

1/8" DIA (U.S. STEEL) MN-MO-NI FILLER WIRE

SINGLE ARC 32-35 VOLTS, 500-550 AMPS 10"-12" IPM

TANDEM ARC 33-36 VOLTS, 500-600 AMPS 20" IPM

LINDE #80 48×D FLUX

PQ 1687

WELDED IN FLAT POSITION

DATE 10-6-67

THE BABCOCK & WILCOX COMPANY

Mt. Vernon, Ind.

CONTRACT NO. 610-0112

SPECIFICATION NO. W-20, W-50

90)-6

ORD OF PROCEDURE AND/OR OPERATOR QUALIFICATION TEST-QC 2E4-122

NAVSHIPS 250-1500-1 [X]	ASME SECTION 3 [X]	ASME SECTION 1 & 8 [X]	OTHER [ ]	PLATE [X] PIPE [ ] [ ]	SINGLE PASS [ ]	MULTIPLE PASS [X]
WELDING PROCESS: Manual Metal Arc		QUALIFICATION POSITION: Horizontal, Flat		METAL THICKNESS: 8 IN.	SINGLE LAYER [ ]	MULTIPLE LAYER [X]
MATERIAL SPECIFICATION: A-302 Gr. B Q & T (B-3)					SINGLE ARC [X]	MULTIPLE ARC [ ]
HEAT TREATMENT: 80 Hours @ 1100-1150°F					MACRO. EXAM.: N.R.	MICRO. EXAM.: N.R.
FLUX NAME OR COMPOSITION: N.A.	PREHEAT TEMPERATURE: 200 °F MIN.	INTERPASS TEMPERATURE: 500 °F MAX.	WELDER SYMBOL NO.: See reverse side			
TYPE OF BACKUP (STRIP OR GAS) AND COMPOSITION: (B-1) P-1 Strip		FILLER METAL GROUP NO.: W1-A (F-4)	SHIELDING GAS: N.A.	CUP SIZE: N.A.	TORCH GAS FLOW RATE: N.A.	
AMPS, VOLTS, CURRENT, POLARITY: 150-425 Amps, 22-26 Volts: DCRP						
SIZE OF ELECTRODE, IN. DIA. / SIZE OF FILLER WIRE, IN. DIA.: 5/32, 3/16, 1/4		ELECTRODE EXT. BEYOND CUP IN.: N.A.	TRAVEL SPEED IPM: N.A.	WIRE FEED IPM: NA	OSCILLATION: NA CY/PM	OSC. TIME SEC.: [illegible]
LIQUID PENETRANT: N.A.		VISUAL INSPECTION: Acceptable - fit up and final weld surfaces				
MAGNETIC PARTICLE: Acceptable, weld prep, & final weld surfaces		RADIOGRAPH: Acceptable, final weld surfaces		ULTRASONIC TEST: Acceptable, final weld surfaces		

REMARKS: Procedure Qualification for welding (B-3) P-3 material in the horizontal position by the manual metal arc process using 5/32", 3/16" & 1/4" Dia. electrodes.

CHEMICAL ANALYSIS - % E NO.

LOCATION	C	MN	P	S	SI	CR	NI	MO	FE	CU	CB	CO	TA	TI	AL
							NR								

REDUCED SECTION TENSILE (TRANSVERSE TO WELD)

SPECIMEN NO.	DIMENSIONS, INCHES WIDTH	THICKNESS	AREA SQ. IN.	ULTIMATE LOAD LBS.	ULTIMATE TENSILE STRENGTH PSI	FRACTURE LOCATION
PQ 107 A	0.553	3.817	2.1108	195,500	92,650	Weld Metal
PQ 107 B	0.674	3.945	2.6589	244,750	92,000	Weld Metal
PQ 107 C	0.882	3.491	3.4910	331,500	95,000	Weld Metal
PQ 107 D	0.899	3.386	3.3856	306,000	90,500	Weld Metal

BEND TEST

Four (4) Transverse Side Bends (split into 24) acceptable. See reverse side.

ALL WELD METAL TENSILE

SPECIMEN NO.	DIAMETER, IN.	AREA SQ. IN.	YIELD POINT PSI	TENSILE STR. PSI	ELONG % IN 2	RED AREA %
			N.R.			

TYPE CHARPY V-Notch IMPACT TEST AT 10 °F 240 FT. LBS. ENERGY LOAD

BASE METAL N.R.		FT. LBS.		FT. LBS.
WELD METAL Surface	86-76-98	FT. LBS.		FT. LBS.
HEAT AFFECTED ZONE 1/4T	69-39-117	FT. LBS.		FT. LBS.

WE CERTIFY THAT TO THE BEST OF OUR KNOWLEDGE THE STATEMENTS MADE IN THIS RECORD ARE CORRECT AND THAT THE TEST WELDS WERE PREPARED, WELDED AND TESTED IN ACCORDANCE WITH THE APPLICABLE SPECIFICATIONS.

N.R. = NOT REQUIRED N.A. = NOT APPLICABLE

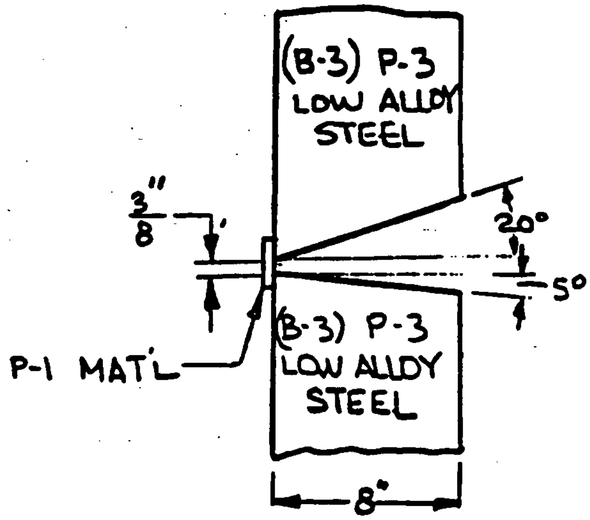
PQ MTV-107 2492

J. Young, NDT - DCAS

WITNESSED J. Walker, D.T. - NDT DCAS DATE 1-23-70

BY O.L. McLaughlin

BABCOCK & WILCOX



TRANSVERSE SIDE BENDS

SPECIMEN NO.	SIZE & NO. OF INDICATIONS	SPECIMEN NO.	SIZE & NO. OF INDICATIONS
1A	1-1/64" Indications	3A	No Indications
1B	1-1/32" Indications	3B	No Indications
1C	No Indications	3C	1-1/64" Indication
1D	No Indications	3D	No Indications
1E	1-1/32" Indications	3E	No Indications
1F	No Indications	3F	No Indications
2A	1-1/32" Indications	4A	1-1/32" Indication
2B	2-1 32" Indications	4B	No Indications
2C	No Indications	4C	1-1/64" Indication
2D	1-1/32" & 1-1/16" Indications	4D	No Indications
2E	1-1/32" Indications	4E	2-1/64" Indication
2F	No Indications	4F	No Indications

Welder Symbol	Number
L. Taylor	688
J. Chism	1182
J. Fisher	219
R. Kaiser	1185

Manual Metal Arc

MIL-E-8015 Electrodes

5/32" Dia., 150-200 Amps, 22-26 Volts
3/16" Dia., 190-270 Amps, 22-26 Volts
1/4" Dia., 325-425 Amps, 22-26 Volts

Welded in the Horizontal Position

Date: 1-23-70

P.Q. MTV-107
2492

THE BABCOCK & WILCOX COMPANY
POWER GENERATION DIVISION
BARBERTON, OHIO

RECORD OF FILLER WIRE QUALIFICATION TEST

TEST NO. WF-25

DIAMETER	ELECTRODE SPECIFICATION AND WELDING PROCESS	FILLER WIRE IDENTIFICATION AND/OR FLUX	CORE WIRE HEAT NO.
1/8"	Submerged Arc ASME III 2E4-142-6 & 156-6	Page Linde Div. Mn-Mo-Ni Linde 80 Lot 8650	299L44

P. ORDER NUMBER	TYPE OF CURRENT	AMPERES
	A.C.	500 Volts 34

WET BATCH EQUIVALENCY CHEMICAL ANALYSIS TESTS

BATCH	HEAT NO.	PAD	C.	MN.	P.	S.	SI.	CR.	NI.	MO.	OTHER
	5396		.088	1.50	.019	.010	.45	.11	.71	.33	
	TEST REPORT	ANALYSIS									

TENSILE PROPERTIES

TEST NO.	HEAT TREATMENT	ULT. TEN. STR. PSI	YIELD POINT PSI	E-LONG IN 2" %	RED OF AREA %
WF-25	48 Hours @ 1100-1150°F.	80,750	66,500	27.3	N.R.

CHARPY V-NOTCH IMPACT TEST @ 10 OF 240 FT/LB. ENERGY LOAD

HEAT TREATMENT	TEST NO.	FT/LB.	TEST NO.	FT. LB.
48 Hours @ 1100-1150°F.	WF-25 A	38		
	WF-25 B	28		
	WF-25 C	49		

GUIDED BEND TESTS

FACE	ROOT	SIDE
N.R.	N.R.	N.R.

MICRO OR MACRO FISSURE ANALYSIS.	N.R.

MATERIAL APPROVAL	APPROVED	REJECTED
NAVSHIPS - 250/1500-1	X	
ASME - COMM'L NUCLEAR	X	
STEAM GENERATORS	N/A	

GROOVE WELD TEST
RADIOGRAPHIC EXAMINATION

WE HEREBY CERTIFY THAT THE ABOVE MATERIAL HAS BEEN TESTED IN ACCORDANCE WITH THE ABOVE LISTED SPECIFICATION AND IS IN CONFORMANCE WITH ALL REQUIREMENTS.

WIRE FOLIO NO. 504-223
FLUX FOLIO NO. 499-063
WORKS Mt. Vernon
CONTRACT NO.

DATE 6/12/69 Revised 11/2/70
SIGNED BABCOCK & WILCOX COMPANY
INSPECTION AGENCY
INSPECTOR J.B. TOON B&W

THE BABCOCK & WILCOX COMPANY
POWER GENERATION DIVISION
BARBERTON, OHIO

RECORD OF FILLER WIRE QUALIFICATION TEST

TEST NO. WF 47

DIAMETER	ELECTRODE SPECIFICATION AND WELDING PROCESS	FILLER WIRE IDENTIFICATION AND/OR FLUX	CORE WIRE HEAT NO.
1/4"	Manual Metal Arc 2E4-142-2 ASME SECTION III	RACO E-8018C-3	495V3011

P. ORDER NUMBER	TYPE OF CURRENT	AMPERES

WET BATCH EQUIVALENCY CHEMICAL ANALYSIS TESTS

BATCH	LAB. NO.	PAD	C.	MN.	P.	S.	SI.	CR.	NI.	MO.	V
	6049		.055	1.10	.007	.013	.38	.03	.955	.31	ND
			ACCEPTABLE								
	TEST REPORT	ANALYSIS									

TENSILE PROPERTIES

TEST NO.	HEAT TREATMENT	ULT. TEN. STR. PSI	YIELD POINT PSI	E-LONG IN 2" %	RED OF AREA %
WF 47	48 Hours at 1100-1150°F.	84,750	75,000	32	74

CHARPY V-Notch IMPACT TEST @ +10 °F 240 FT/LB. ENERGY LOAD

HEAT TREATMENT	TEST NO.	FT/LB.	TEST NO.	FT/LB.
48 Hours at 1100-1150°F.	WF 47	127		
	WF 47	127		
	WF 47	143		

GUIDED BEND TESTS

FACE	ROOT	SIDE
N.R.	N.R.	N.R.

MICRO OR MACRO FISSURE ANALYSIS.	N.R.

MATERIAL APPROVAL	APPROVED	REJECTED
NAVSHIPS - 250/1500-1	X	
ASME - COMM'L NUCLEAR	N/A	
STEAM GENERATORS	N/A	

GROOVE WELD TEST

RADIOGRAPHIC EXAMINATION

WE HEREBY CERTIFY THAT THE ABOVE MATERIAL HAS BEEN TESTED IN ACCORDANCE WITH THE ABOVE LISTED SPECIFICATION AND IS IN CONFORMANCE WITH ALL REQUIREMENTS.

WIRE FOLIO NO. 503-772

FLUX FOLIO NO.

WORKS Mt. Vernon

CONTRACT NO.

DATE 8/20/69

SIGNED Babcock & Wilcox Company

INSPECTION AGENCY

INSPECTOR J.B. Toon B&W

THE BABCOCK & WILCOX COMPANY
POWER GENERATION DIVISION
BARBERTON, OHIO

RECORD OF FILLER WIRE QUALIFICATION TEST

TEST NO. ______

DIAMETER	ELECTRODE SPECIFICATION	FILLER WIRE IDENTIFICATION	CORE WIRE HEAT NO.
5/32"	2E4-142-1 ASME SECTION III	E-8015C-3	80E010

P. ORDER NUMBER	TYPE OF CURRENT	AMPERES

WET BATCH EQUIVALENCY CHEMICAL ANALYSIS TESTS

	LAB. NO.	PAD	C.	MN.	P.	S.	SI.	CR.	NI	MO.	V
	E 70183	2752	.05	.66	.011	.017	.23	.02	.87	.51	.01
	TEST REPORT	ANALYSIS									

TENSILE PROPERTIES

TEST NO.	HEAT TREATMENT	ULT. TEN. STR. PSI	YIELD POINT PSI	E-LONG IN 2" %	RED OF AREA %
	1100-1150 48 Hours	84,750	75,500	27.5	70.1

CHARPY V-Notch IMPACT TEST @ 10 OF 240 FT/LB. ENERGY LOAD

HEAT TREATMENT	TEST NO.	FT/LB.	TEST NO.	FT. LB.
48 Hours 1100-1150° F.	1	105		
	2	99		
	3	105		

GUIDED BEND TESTS

FACE	ROOT	SIDE
	N.R.	

MICRO OR MACRO FISSURE ANALYSIS.	N.R.

MATERIAL APPROVAL	APPROVED	
NAVSHIPS - 250/1500-1	X	
ASME - COMM'L NUCLEAR STEAM GENERATORS	X	

GROOVE WELD TEST

RADIOGRAPHIC EXAMINATION

WE HEREBY CERTIFY THAT THE ABOVE MATERIAL HAS BEEN TESTED IN ACCORDANCE WITH THE ABOVE LISTED SPECIFICATION AND IS IN CONFORMANCE WITH ALL REQUIREMENTS.

WIRE FOLIO NO. ______

FLUX FOLIO NO. ______

WORKS ______

CONTRACT NO. ______

DATE June 2, 1969

SIGNED G.K. Bryan

INSPECTION AGENCY ______

INSPECTOR ______
AL CREMERS

THE BABCOCK & WILCOX COMPANY
POWER GENERATION DIVISION
BARBERTON, OHIO

RECORD OF FILLER WIRE QUALIFICATION TEST

TEST NO. ____________

DIAMETER	ELECTRODE SPECIFICATION	FILLER WIRE IDENTIFICATION	CORE WIRE HEAT NO.
1/8"	2E4-142-1 ASME SECTION III	E 8015C-3	85D496

P. ORDER NUMBER	TYPE OF CURRENT	AMPERES
818-026222	D.C.	

WET BATCH EQUIVALENCY CHEMICAL ANALYSIS TESTS

BATCH	LAB. NO.	PAD	C.	MN.	P.	S.	SI.	CR.	NI.	MO	V
1	E-69592	2742	.05	.82	.013	.020	.25	.03	.84	.57	.01
	TEST REPORT	ANALYSIS									

TENSILE PROPERTIES

TEST NO.	HEAT TREATMENT	ULT. TEN. STR. PSI	YIELD POINT PSI	E-LONG IN 2" %	RED OF AREA %
	1100-1150 48 Hours	84,000	73,000	27.0	71.9

CHARPY V-Notch IMPACT TEST @ 10 OF 240 FT/LB. ENERGY LOAD

HEAT TREATMENT	TEST NO.	FT/LB.	TEST NO.	FT. LB.
1100-1150 48 Hours	1	100		
	2	81		
	3	113		

GUIDED BEND TESTS

FACE	ROOT	SIDE
	N.R.	

MICRO OR MACRO FISSURE ANALYSIS.	N.R.

MATERIAL APPROVAL	APPROVED	REJECTED
NAVSHIPS - 250/1500-1	X	
ASME - COMM'L NUCLEAR	X	
STEAM GENERATORS		

GROOVE WELD TEST

RADIOGRAPHIC EXAMINATION

WE HEREBY CERTIFY THAT THE ABOVE MATERIAL HAS BEEN TESTED IN ACCORDANCE WITH THE ABOVE LISTED SPECIFICATION AND IS IN CONFORMANCE WITH ALL REQUIREMENTS.

WIRE FOLIO NO. ____________

FLUX FOLIO NO. ____________

WORKS ____________

CONTRACT NO. ____________

DATE November 22, 1969

SIGNED G.K. Bryan

INSPECTION AGENCY ____________

INSPECTOR ____________
AL CREMERS

THE BABCOCK & WILCOX COMPANY
POWER GENERATION DIVISION
BARBERTON, OHIO

RECORD OF FILLER WIRE QUALIFICATION TEST

TEST NO. Pad 026223

DIAMETER	ELECTRODE SPECIFICATION	FILLER WIRE IDENTIFICATION	CORE WIRE HEAT NO.
3/16"	2E4-156-1 ASME MMA SECTION III	E 8015C-3	81D497

ORDER NUMBER	TYPE OF CURRENT	AMPERES
818-026223	D.C.	190-270

WET BATCH EQUIVALENCY CHEMICAL ANALYSIS TESTS

	LOT NO.	PAD	C.	MN.	P.	S.	SI.	CR.	NI.	MO.	V
	E 69461	2718	.05	.89	.011	.017	.28	.03	.85	.62	.01
		BARBERTON ANALYSIS									
	TEST REPORT	ANALYSIS									

TENSILE PROPERTIES

TEST NO.	HEAT TREATMENT	ULT. TEN. STR. PSI	YIELD POINT PSI	E-LONG IN 2" %	RED OF AREA
026223	48 Hours @ 1100-1150°F.	85,250	74,000	30.5	71.0

CHARPY V-Notch IMPACT TEST @10 OF 240 FT/LB. ENERGY LOAD

HEAT TREATMENT	TEST NO.	FT/LB.	TEST NO.	FT/LB.
48 Hours @ 1100-1150	026223	95		
	"	97		
	"	105		

GUIDED BEND TESTS

FACE	ROOT	SIDE
N.R.	N.R.	N.R.

MICRO OR MACRO FISSURE ANALYSIS.	

MATERIAL APPROVAL	APPROVED	REJECTED
NAVSHIPS - 250/1500-1		
ASME - COMM'L NUCLEAR STEAM GENERATORS	X	

GROOVE WELD TEST
RADIOGRAPHIC EXAMINATION

WE HEREBY CERTIFY THAT THE ABOVE MATERIAL HAS BEEN TESTED IN ACCORDANCE WITH THE ABOVE LISTED SPECIFICATION AND IS IN CONFORMANCE WITH ALL REQUIREMENTS.

WIRE FOLIO NO. ______
FLUX FOLIO NO. ______
WORKS ______
CONTRACT NO. ______

DATE August 22, 1969
SIGNED I. Barnes B&W
INSPECTION AGENCY ______
INSPECTOR ______

THE BABCOCK & WILCOX COMPANY
POWER GENERATION DIVISION
BARBERTON, OHIO

RECORD OF FILLER WIRE QUALIFICATION TEST

TEST NO. Pad 026224

DIAMETER	ELECTRODE SPECIFICATION	FILLER WIRE IDENTIFICATION AND/OR FLUX	CORE WIRE HEAT NO.
1/8"	2E4-156-1 ASME MMA SECTION III	E 8015C-3	84D798

P. ORDER NUMBER	TYPE OF CURRENT	AMPERES
818-026224	D.C.	110-150

WET BATCH EQUIVALENCY CHEMICAL ANALYSIS TESTS

BATCH	LAB. NO.	PAD	C.	MN.	P.	S.	SI.	CR.	NI.	MO.	V
	E 69256	2711	.04	.72	.009	.020	.32	.08	1.04	.55	.01
		BARBERTON ANALYSIS									
	TEST REPORT	ANALYSIS									

TENSILE PROPERTIES

TEST NO.	HEAT TREATMENT	ULT. TEN. STR. PSI	YIELD POINT PSI	E-LONG IN 2" %	RED OF AREA %
026224	48 Hours @ 1100-1150	87,000	76,750	30.5	69.2

CHARPY V-Notch IMPACT TEST @ 10 °F 240 FT/LB. ENERGY LOAD

HEAT TREATMENT	TEST NO.	FT/LB.	TEST NO.	FT. LB.
48 Hours @ 1100-1150	026224	105		
	"	110		
	"	119		

GUIDED BEND TESTS

FACE	ROOT	SIDE
N.R.	N.R.	N.R.

MICRO OR MACRO FISSURE ANALYSIS.	

MATERIAL APPROVAL	APPROVED	[illegible]
NAVSHIPS - 250/1500-1		
ASME - COMM'L NUCLEAR STEAM GENERATORS	X	

GROOVE WELD TEST

RADIOGRAPHIC EXAMINATION

WE HEREBY CERTIFY THAT THE ABOVE MATERIAL HAS BEEN TESTED IN ACCORDANCE WITH THE ABOVE LISTED SPECIFICATION AND IS IN CONFORMANCE WITH ALL REQUIREMENTS.

WIRE FOLIO NO. ____

FLUX FOLIO NO. ____

WORKS ____

CONTRACT NO. ____

DATE August 22, 1969

SIGNED I. Barnes B&W

INSPECTION AGENCY ____

INSPECTOR ____

THE BABCOCK & WILCOX COMPANY
POWER GENERATION DIVISION
BARBERTON, OHIO

RECORD OF FILLER WIRE QUALIFICATION TEST

TEST NO. Pad 2515

DIAMETER	ELECTRODE SPECIFICATION	FILLER WIRE CLASSIFICATION	CORE WIRE HEAT NO.
5/32"	2E4-142-1 NAVSHIPS 250-1500-1 SECTION III ASME	E 8015C-3 MIL 8015C-3	88D474

P. ORDER NUMBER	TYPE OF CURRENT	AMPERES
818-025187		

Radiography - Good

WET BATCH EQUIVALENCY CHEMICAL ANALYSIS TESTS

NO.	LAB. NO.	PAD	C.	MN.	P.	S.	SI.	CR.	NI.	MO.	V
1	E66210	2515	.05	.66	.009	.014	.26	.01	.87	.43	.01
2			.07	.75	.012	.013	.30	.01	.87	.42	.01
3			.07	.69	.011	.013	.27	.01	.87	.43	.01
4			.06	.64	.011	.012	.24	.01	.84	.42	.01
5			.06	.70	.011	.012	.26	.01	.84	.43	.01
6			.06	.64	.010	.014	.24	.01	.90	.45	.01
7			.06	.74	.011	.013	.30	.01	.89	.43	.01
8			.05	.63	.010	.013	.25	.01	.85	.46	.01
9			.06	.70	.011	.014	.27	.01	.87	.43	.01
10			.05	.71	.009	.014	.27	.01	.81	.44	.01
	TEST REPORT	ANALYSIS	GROOVE WELD TEST - GOOD								

TENSILE PROPERTIES

TEST NO.	HEAT TREATMENT	ULT. TEN. STR. PSI	YIELD POINT PSI	E-LONG IN 2" %	RED OF AREA %
AW	NO	84,000	75,000	27.0	72.3
SR	YES	80,500	71,500	29.0	73.5

CHARPY V-NOTCH IMPACT TEST @ 0 OF 240 FT/LB. ENERGY LOAD

HEAT TREATMENT	TEST NO.	FT/LB.	TEST NO.	FT. LB.
1100-1150°F @ 48 Hours	2515	119	120	105
90 Hours	2515	110	109	106
140 Hours	2515	118	168	140

GUIDED BEND TESTS

FACE	ROOT	SIDE

MICRO OR MACRO FISSURE ANALYSIS.

MATERIAL APPROVAL	APPROVED	REJECTED
NAVSHIPS - 250/1500-1	X	
ASME - COMM'L NUCLEAR STEAM GENERATORS	X	

GROOVE WELD TEST
RADIOGRAPHIC EXAMINATION

WE HEREBY CERTIFY THAT THE ABOVE MATERIAL HAS BEEN TESTED IN ACCORDANCE WITH THE ABOVE LISTED SPECIFICATION AND IS IN CONFORMANCE WITH ALL REQUIREMENTS.

WIRE FOLIO NO. ______
FLUX FOLIO NO. ______
WORKS ______
CONTRACT NO. ______

DATE November 22, 1968
SIGNED P.E. Campbell
INSPECTION AGENCY ______
INSPECTOR W. Ruble/DCASO

THE BABCOCK & WILCOX COMPANY
POWER GENERATION DIVISION
BARBERTON, OHIO

RECORD OF FILLER WIRE QUALIFICATION TEST

TEST NO. Pad 1779

DIAMETER	[illegible] SPECIFICATION	FILLER [illegible] IDENTIFICATION	CORE WIRE HEAT NO.
3/32"	MIL-E2220016 ASME NAVSHIPS 250-1500-1	E 8015C-3 MIL 8015C-3	5861
P. ORDER NUMBER	SECTION III 818-021736	TYPE OF CURRENT	AMPERES

Radiograph - Good

WET BATCH EQUIVALENCY CHEMICAL ANALYSIS TESTS

[illegible]	[illegible] NO.	PAD	C.	MN.	P	S.	SI.	CR.	NI.	MO.	V
1	E 56231	1779	.04	.86	.012	.017	.28	.03	1.06	.55	ND
2	"	"	.05	.99	.006	.021	.34	.04	1.01	.55	ND
3	"	"	.04	.90	.006	.022	.29	.03	1.06	.54	ND
4	E 56257	"	.04	.68	.011	.024	.20	.04	.98	.54	ND
			GROOVE WELD TEST - 4 GOOD								
	TEST REPORT	ANALYSIS									

TENSILE PROPERTIES

TEST NO.	HEAT TREATMENT	ULT. TEN. STR. PSI	YIELD POINT PSI	E-LONG IN 2" %	RED OF AREA %
1779AW	NO	101,000	94,500	26.0	66.0
1779-2	YES	98,750	94,000	25.0	63.9

CHARPY V-NOTCH IMPACT TEST @ 0&10 °F 240 FT/LB. ENERGY LOAD

HEAT TREATMENT	TEST NO.	FT/LB.	TEST NO.	FT/LB. + 10
48 Hours at 1100 to 1150° F	1	70	1	70
	2	72	2	76
	3	79	3	95

GUIDED BEND TESTS

FACE	ROOT	SIDE

MICRO OR MACRO FISSURE ANALYSIS.	

MATERIAL APPROVAL	APPROVED	REJECTED
NAVSHIPS - 250/1500-1	X	
ASME - COMM'L NUCLEAR STEAM GENERATORS	X	
GROOVE WELD TEST		
RADIOGRAPHIC EXAMINATION		

WE HEREBY CERTIFY THAT THE ABOVE MATERIAL HAS BEEN TESTED IN ACCORDANCE WITH THE ABOVE LISTED SPECIFICATION AND IS IN CONFORMANCE WITH ALL REQUIREMENTS.

WIRE FOLIO NO. ____

FLUX FOLIO NO. ____

WORKS ____

CONTRACT NO. ____

DATE May 1, 1967

SIGNED P.E. Campbell

INSPECTION AGENCY ____

INSPECTOR ____

THE BABCOCK & WILCOX COMPANY
BOILER DIVISION

QUALITY CONTROL SPECIFICATION

ISSUED	SUBJECT	SPEC. NO.
1-10-66	SUBMERGED ARC WELDING GENERAL SPECIFICATION FOR SPECIAL PRODUCTS OR NUCLEAR APPLICATION	WS-48

1. SCOPE: This general welding specification shall govern the submerged arc welding of carbon steel (P-Number 1), low alloy steel (P-Number 3), stainless steel (P-Number 8), and nickel-chromium-iron (P-Number 43) materials in accordance with the ASME Boiler and Pressure Vessel Code and all applicable contract requirements. This specification shall be used for special products and nuclear application. First Contract 610-0110.

   The details of each specific welding operation shall be listed on data sheets which supplement this specification.

2. BASE MATERIALS: The base materials shall be those listed under P-Numbers 1, 3, and 8 of Table Q-11.1 and P-Number 43 of Table QN-11.1 of Section IX of the ASME Boiler and Pressure Vessel Code.

   Welds may involve P-Numbers 1, 3, 8, and 43 materials singly or in combination with each other.

3. FILLER MATERIALS: The specific type of electrode to be used for each application shall be designated on the applicable data sheet. These electrodes shall consist of the following chemical compositions:

3.1 Carbon and Low Alloy (F-1) Electrodes:

3.1.1 High Manganese-Molybdenum

Carbon	0.010-0.17
Manganese	1.70 -2.10
Silicon	0.05 Maximum
Sulphur	0.035 Maximum
Aluminum	0.020 Maximum
Molybdenum	0.45 -0.60

3.1.2 High Manganese-Molybdenum-Nickel

Carbon	0.10-0.14
Manganese	1.75-2.25
Sulphur	0.020 Maximum
Phosphorus	0.020 Maximum
Silicon	0.10 Maximum
Nickel	0.50-0.70
Molybdenum	0.35- .55

3.2 Stainless Steel (F-7) Electrodes:

Carbon	0.030 Maximum
Chromium	19.5 -22.0
Nickel	9.0 -11.0
Manganese	1.0 - 2.5
Silicon	0.25-0.60

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3	HLH:AEK	Revised Paragraph 7.5	WS-48
REVISION DATE 5-24-67			PAGE NO. 1 of 6

DISTRIBUTION

Phosphorus	0.030 Maximum
Sulphur	0.030 Maximum

The minimum Chromium to Nickel ratio shall be 1.9/1.0.

3.3 Nickel-Chrimium-Iron (F-45) Electrodes:

Carbon	0.10 Maximum
Manganese	2.5-3.5
Iron	3.00 Maximum
Sulphur	0.015 Maximum
Silicon	0.50 Maximum
Copper	0.50 Maximum
Nickel plus Cobalt	67.0 Minimum
Titanium	0.75 Maximum
Chromium	18.0-22.0
Columbium plus Tantalum	2.0-3.0
Other Elements Total	0.50 Maximum

3.4 The flux type and particle size, where applicable, shall be specified on the applicable data sheet.

4. PREPARATION OF WELDING GROOVE:

4.1 General: The weld preparation and adjoining base metal for a minimum distance of 1" from the welding groove shall be free of oil, grease, dirt, scale, and other materials which would affect the integrity of the weld.

4.2 P-Number 1 and 3 Materials:

4.2.1 The base material preparation for surfaces to be joined by welding may be accomplished by thermal cutting processes or by the mechanical removal of material. If thermal cutting processes are used, a minimum preheat equal to that shown on the supplementary data sheet shall be applied to the material and maintained during thermal cutting operations. After thermal cutting operations for P-Number 3 materials, the cut surfaces shall be removed by a mechanical method to clean, smooth, bright metal. Not less than 1/32" of base metal shall be removed by mechanical means from any cut surfaces prepared by the air-arc process.

4.2.2 The surfaces upon which weld metal will be deposited shall be examined by the magnetic particle method in accordance with Quality Control Specification S-102B prior to welding.

4.3 P-Number 8 and 43 Materials:

4.3.1 The base material preparation for surfaces to be joined by welding shall be accomplished by thermal cutting processes or by the mechanical removal of material to clean, smooth, bright metal.

4.3.2 Not less than 1/32" of metal shall be removed by mechanical means from surfaces from which metal has been removed by air-arc, inert gas arc, or carbon arc cutting. Not less than 1/8" of metal shall be removed where oxygen arc or oxygen cutting (including both powder and flux cutting) are used.

4.3.3 The surfaces upon which weld metal will be deposited shall be examined by the liquid penetrant method in accordance with Quality Control Specification S-102C.

BACKING STRIPS, BACKING RINGS, AND SPACER BLOCKS:

5.1 Similar Metal Welds:

5.1.1 The materials of the backing strips, backing rings, and spacer blocks shall be of the same P-Number as the base material except that carbon steel (P-Number 1) may be used for low alloy (P-Number 3) base material.

5.1.2 Spacer blocks and fitting bars for stainless steel and Nickel-Chromium-Iron base materials may be made of equivalent material or of carbon steel which is clad with weld metal of the same P-Number as the base material being welded.

5.1.3 After completion of welding, all backing materials shall be removed and the back side of the root shall be inspected by the magnetic particle or liquid penetrant methods.

5.2 Dissimilar Metal Welds:

5.2.1 For welding P-Number 8 to P-Number 1 or 3 materials, the backing ring, backing strip, or spacer block shall be made of P-Number 1 or 3 materials except that when the carbon or low alloy steel material has a corrosion resistant liner, the material shall be P-Number 8.

5.2.2 For welding P-Number 43 to P-Number 1 or 3 materials, the backing ring, backing strip, or spacer block shall be made of P-Number 43 material when the carbon or low alloy steel material has a corrosion-resistant liner. For welding P-Number 43 to P-Number 8 materials, the backing ring, backing strip, or spacer block shall be made of P-Number 43 or P-Number 8 material.

6. WELDING CURRENT CHARACTERISTICS: Welding current characteristics shall be listed on the supplementary data sheet for each weld.

7. WELDING TECHNIQUE:

7.1 Submerged arc welding shall be performed by the single and multiple arc technique.

7.2 Start and stop areas shall be ground or chipped prior to depositing succeeding passes. As an alternate, run-out tabs may be used which shall be removed upon completion of welding.

7.3 When welding in a groove, weld metal shall be deposited to a height of not less than 1/16" and not greater than 1/8" above the base material surface.

7.4 Arc initiation shall be through a small steel wool ball, approximately 1/4" in diameter, placed between the electrode tip and work piece.

7.5 Overlay Cladding:

7.5.1 When stainless steel (F-7) weld metal is clad on P-Number 1 or 3 materials, the initial starting bead shall be deposited by either manual metal arc welding with E-309-15 electrodes on the surfaces to be clad, or by the automatic submerged arc process on a flux ring extension of the surface to be clad. The flux ring, if used, shall be removed upon the completion of welding.

7.5.1.1 Weld beads of the first layer of cladding shall be deposited with the electrode wire positioned one wire diameter in from the leading edge of the previously deposited bead to insure that excessive dilution from the base material does not occur. Beads of subsequent layers shall be deposited with the electrode wire positioned off the previous bead a distance equal to the electrode wire diameter.

7.5.1.2 The ferrite content shall be periodically checked. Such checks shall be made with a magnagage once each layer or every four hours of welding, whichever occurs first, or when an in-process change is made in the heat of filler wire or lot of flux. The deposited weld metal shall have a ferrite content of 5 to 15 per cent. If these limits are exceeded, the welding shall be stopped and the problem referred to Quality Control.

7.5.2 When stainless steel (F-7) weld metal is clad on P-Number 1 or 3 material using the oscillating arc technique, the following procedure shall be employed:

7.5.2.1 The welding shoe shall be oscillated at a rate of 72 cycles per minute with an oscillation width of 7/8" to 1".

7.5.2.2 Each weld bead shall be deposited with the electrode wire positioned to overlap the previous bead a distance of 1/4" at the extremity of the oscillation adjacent to the previous bead.

7.5.3 When Nickel-Chromium-Iron (F-45) weld metal is clad on P-Number 1 or 3 materials, the welding shoe shall be oscillated at 72 cycles per minute with an oscillation width of 7/8" to 1". Each weld bead shall be deposited with the electrode wire positioned to overlap the previous bead a distance of 1/4" at the extremity of the oscillation adjacent to the previous bead.

7.6 After each bead of weld is deposited, care shall be taken to:

7.6.1 Remove all slag or flux remaining on the surface.

7.6.2 Remove and repair, if necessary, any defect that appears on the surface of any weld bead.

7.6.3 Remove excessive overlap or crown on beads by chipping or grinding.

7.6.4 Remove all weld spatter, scale, or welding slag on the sidewalls of the weld joint.

7.7 Where wire brushes are used on stainless steel and nickel-chromium-iron alloys they shall be stainless steel and not previously used on carbon or low alloy steels.

7.8 Fused flux shall not be reclaimed by grinding and subsequently reused.

8. THERMAL TREATMENT:

8.1 Preheat and Interpass Temperatures:

8.1.1 The minimum preheat and maximum interpass temperatures shall be as specified in Quality Control specification S-170 "Preheat and Interpass Temperature Control" and the applicable data sheet. The preheat and interpass temperatures shall be measured by lead-free "Tempilstiks".

8.1.2 P-Number 1 Materials:

8.1.2.1 Preheat may be reduced to room temperature prior to postweld heat treatment for the following applications:

A. Longitudinal weld seams in vessels that have been completely welded and are 3" or less in thickness.

B. Circumferential weld seams in vessels and butt welds in structural members and flat plate provided 1" or 1/3 of the finished weld thickness, whichever is greater, has been deposited. Abrupt changes in contour between the edges of uncompleted welds and the groove face of the weld joint shall be eliminated prior to dropping preheat.

8.1.2.2 For longitudinal seams exceeding 3" in thickness and circumferential weld seams exceeding 8" in thickness, the preheat temperature may be lowered to 200F after the completion of welding prior to postweld heat treatment. During the holding period, the temperature shall not exceed 800F.

8.1.3 P-Number 3 Materials:

8.1.3.1 The preheat temperature shall be held at a temperature not less than the minimum specified on the data sheet for the specific weld and not more than 800F until a postweld heat treatment is performed.

8.2 Postweld Heat Treatment:

8.2.1 Postweld heat treatment temperature shall be as specified on the applicable data sheet. All postweld heat treating shall be done in accordance with the applicable approved contract specification.

8.2.2 The final heat treatment shall be the final scheduled heat treatment for a weld deposit which requires a heat treatment of the component in accordance with contract requirements and manufacturing schedules.

This heat treatment shall be performed with attached thermocouples used to record temperature and time at temperature. Attached thermocouples shall also be used when the last heat treatment is performed for weld repairs.

9. QUALIFICATION OF PROCEDURES: All welders and procedures shall be qualified in accordance with ASME Boiler and Pressure Vessel Code, Section IX and contract requirements. Previous qualifications shall be extended as permitted by the contract.

THE BABCOCK & WILCOX COMPANY
BOILER DIVISION

QUALITY CONTROL SPECIFICATION

ISSUED	SUBJECT	SPEC. NO.
12-2-63	RADIOGRAPHIC METHODS AND ACCEPTANCE STANDARDS	S-102A

1. SCOPE: This specification shall outline the general requirements for radiographic inspection of welds and base materials.

1.1 For more specific and detailed requirements and information for radiography performed on components fabricated in accordance with Sections I and VIII of the ASME Boiler and Pressure Vessel Code, refer to Supplement I of S-102A.

1.2 For more specific and detailed requirements of radiography performed on components fabricated in accordance with Section III of the ASME Boiler and Pressure Vessel Code or MIL-STD-271, refer to radiographic outline drawings prepared for each specific contract by the Engineering Department and the NDT Section of Quality Control.

2. EQUIPMENT: The radiographic equipment shall consist of X-ray machines and gamma-ray emitters, Cobalt 60 and Iridium 192, with suitable intensities to provide the desired radiographs in accordance with the limitations as specified in the applicable ASME Code or MIL-STD-271 documents.

3. EXTENT OF RADIOGRAPHIC INSPECTION: Parts or assemblies shall be radiographed as stipulated on contractual drawings and documents.

4. FILM: Radiographs shall be made using fine grain, extra fine grain, or equivalent, high contrast, safety base film unless otherwise specified by contractual specifications or documents.

4.1 Film Density:

4.1.1 The film density in the area being examined shall be as specified below:

ASME Code - Single Film viewing 1.3 minimum.
Double Film viewing (film superimposed) 1.8 minimum.

MIL-STD-271 - Single Film viewing shall be between 1.5 and 3.5.
Double Film viewing (film superimposed) shall be between 2.0 and 3.5.

4.1.2 Densities shall be measured using a densitometer.

4.2 Film Quality: All radiographs shall be free of mechanical, chemical, processing, or other defects which could interfere with interpretation. For all ASME Code Section III and MIL-STD-271 contracts, the double film technique shall be used.

4.3 Film Processing: Film processing shall be accomplished by either hand processing or automatic processors in accordance with the film manufacturers recommendations.

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2	AEK:clp	Rev. Par. 6.2.2, 8 and 9 Added Par. 12	S-102A
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THE BABCOCK & WILCOX COMPANY
BOILER DIVISION
QUALITY CONTROL SPECIFICATION

SPEC. NO.	SUBJECT	ISSUED
S-102A	RADIOGRAPHIC METHODS AND ACCEPTANCE STANDARDS	12-2-63

4.4 Film Identification: Identification on the radiographs shall include contract number, name of part being examined, weld seam identification, and date. If film size limitations restrict the inclusion of such information a radiographic number shall be assigned to the part or weld seam.

The radiographic number and date shall appear on all radiographs when this method is used. The identifying information shall be cross referenced to the radiograph numbers and shall - a part of the permanent records kept on file for that contract.

4.5 Film Viewing Facilities: Film viewing facilities shall consist of a darkened room exclusive of background lighting. An illuminator with a cooling device and sufficient intensity to evaluate radiographs with densities as outlined herein shall be provided.

4.6 Film Storage and Records: Radiographs shall be retained for 10 years or for a longer period as agreed upon by the customer, bureau, or agency concerned.

5. IDENTIFICATION MARKING: Location markers or markings shall be placed on the part to provide proper orientation of the radiograph. When permitted, location markers shall be permanently located by light center punch marks. The first and last positions shall be steel stamped using round bottom low stress stamps. When this type of permanent marking is prohibited, other methods shall be used, such as sketches of the layouts which shall be kept with the radiographs.

6. PENETRAMETERS:

6.1 Penetrameters shall be in accordance with Figure 1 or 2.

6.2 Penetrameter Placement:

6.2.1 Two penetrameters shall be placed on the source side, at each end of the area to be interpreted when physically possible. When radiographing welds, penetrameters shall be placed on the base metal, 1/8" to 3/4" from the edge of the weld.

6.2.2 When the source is placed on the axis of a weld joint and the complete circumference is radiographed with a single exposure, at least four penetrameters shall be uniformly spaced at 90 degrees intervals around the circumference.

6.2.3 When placement of the penetrameter on the source side is physically impossible, the penetrameter may be placed on the film side in accordance with procedures described in the applicable ASME Code or MIL-STD-271 documents.

7. SHIMS: When a weld reinforcement, backing ring, or backing strip is not removed, a shim of material which is radiographically similar to the backing ring or backing strip shall be placed under the penetrameter. The thickness of the shim shall be sufficient to insure the same thickness of material under the penetrameter as the average combined thickness of the weld reinforcement and the backing ring or strip. The shim shall exceed the penetrameter dimensions by at least 1/8" on at least three sides, and shall be placed so as not to over lap the backing ring or strip. The film density of the shim image shall be not greater than 15 per cent more than the film density in the area of interest.

8. SENSITIVITY OF RADIOGRAPHS OR IMAGES: Radiography shall be performed with a technique that will have sufficient sensitivity to show the outline of each penetrameter, the identifying numbers of the penetrameter, and the holes or slits in the penetrameter that are required to be shown by the codes and specifications applicable to the contract. If these are not visible the density of the film shall be increased sufficiently above the minimum densities listed in Par. 4.1.1 without exceeding the maximum limits permitted to insure that the features of the penetrameter are visible.

9. INTENSIFYING SCREENS: Radiographs shall employ both front and back lead screens in contact with the film as follows:

9.1 Intensifying screens shall not be used for an X-ray source of 120KV and less or material thicknesses up to and including 1/4".

9.2 Intensifying screens of .005" thickness shall be used for material thicknesses over 1/4" to and including 3/4" regardless of the X-ray source.

9.3 Intensifying screens of .010" thickness shall be used for material thicknesses over 3/4" regardless of the X-ray source.

9.4 Intensifying screens of .010" thickness shall be used for all thicknesses of material rediographed by gamma radiation.

10. FILTERS:

10.1 Front Filters: When using X-ray machines with energies above 0.7 MEV and the kilo curie Co-60 source, a filter 0.080" thick shall be placed in front of the film holder.

10.2 Back Filters: A lead filter of sufficient thickness shall be placed in back of the film holder to prevent scattered radiation from fogging the film. Each film holder shall have a lead letter "B" 1/2" high and 1/16" thick place on the back of the film holder, outside the area to be read for acceptance. If the image of this letter appears on the film, additional back filter thickness shall be required.

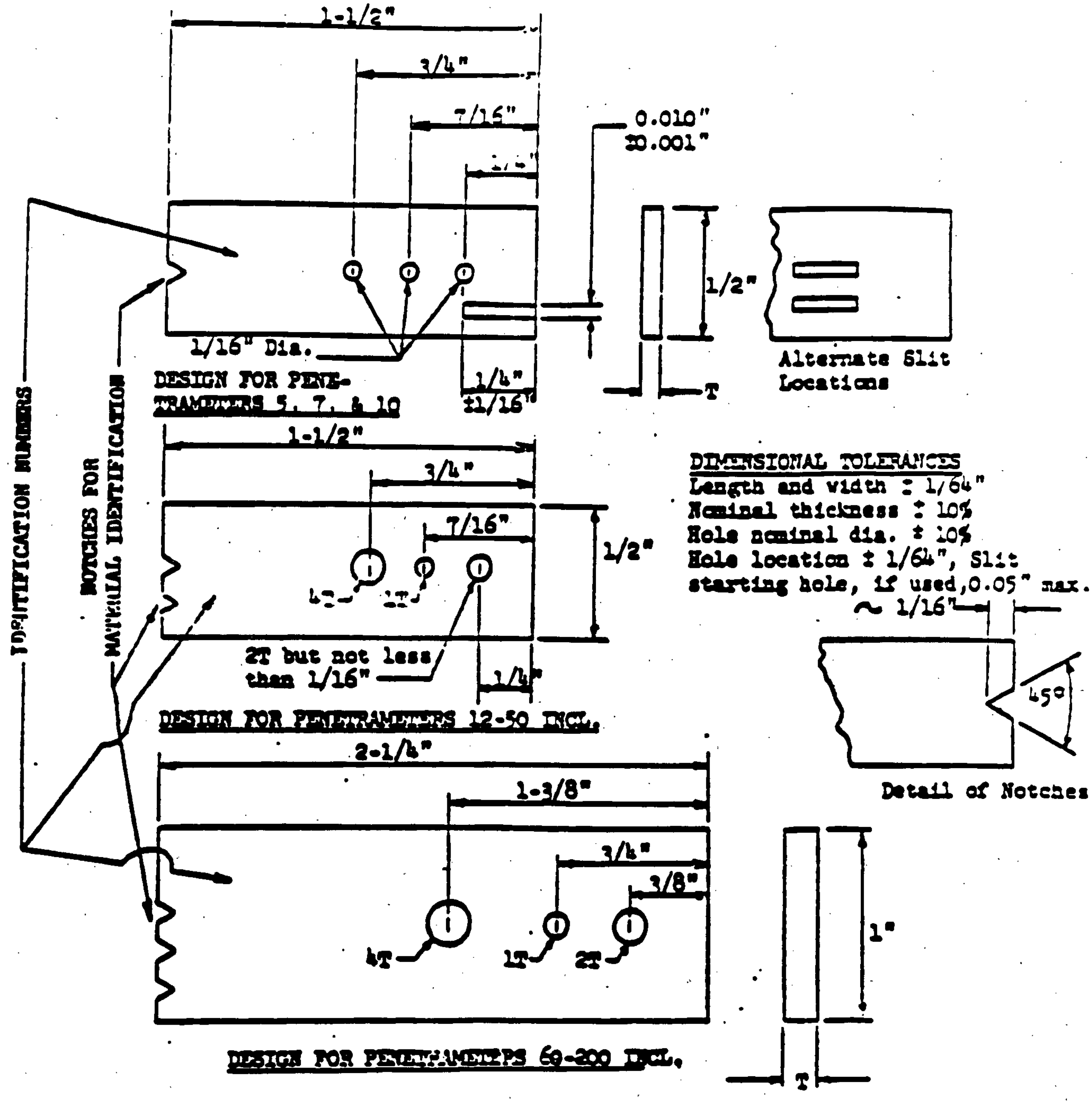
11. ACCEPTANCE STANDARDS: Acceptance standards shall be in accordance with the ASME Boiler and Pressure Vessel Code, or other contractual specifications and documents.

12. SURFACE PREPARATION: Surfaces to be radiographed shall be free of scale, slag, undercuts, adherring, or imbedded sand that may interfere with proper interpretation of radiographs. Surfaces of welds shall have valleys between beads, weld nipples, or other surface irregularities blended so that they are not detectable on the radiograph.

ISSUED	SUBJECT	SPEC. NO.
12-2-63	RADIOGRAPHIC METHODS AND ACCEPTANCE STANDARDS	S-102A

FIGURE 1

ASME PENETRAMETERS



REV. BY	REV. NO.	REVISION	SPEC. NO.
AEK:clp	2		S-102A
REVISION DATE 6-27-66		See Page 1	PAGE NO. 5 of 8

DISTRIBUTION

NOTES TO FIGURE 1

PENETRAMETER IDENTIFICATION

Material Thickness Range (TM)	Penetrameter Thickness	Penetrameter Identification
Up to and including 1/4"	0.005"	5
Over 1/4" through 3/8"	0.0075"	7
Over 3/8" through 1/2"	0.010"	10
Over 1/2" through 5/8"	0.0125"	12
Over 5/8" through 3/4"	0.015"	15
Over 3/4" through 7/8"	0.0175"	17
Over 7/8" through 1"	0.020"	[illegible]
Over 1" through 1-1/4"	0.025"	25
Over 1-1/4" through 1-1/2"	0.030"	30
Over 1-1/2" through 2"	0.035"	35
Over 2" through 2-1/2"	0.040"	40
Over 2-1/2" through 3"	0.045"	45
Over 3" through 4"	0.050"	50
Over 4" through 6"	0.060"	60
Over 6" through 8"	0.080"	80
Over 8" through 10"	0.100"	100
Over 10" through 12"	0.120"	120
Over 12" through 16"	0.160"	160
Over 16" through 20"	0.200"	200

NOTE: Each penetrameter shall be identified by notches and scribing or vibro tooling the material identification as stated below:

	Material Identification	No. of Notches
1. Steels	Fe	1
2. Copper-Nickel Alloy	CuNi	2
3. Nickel-Copper Alloy	NiCu	3
4. Nickel-Chromium-Iron Alloy	NiCr	4

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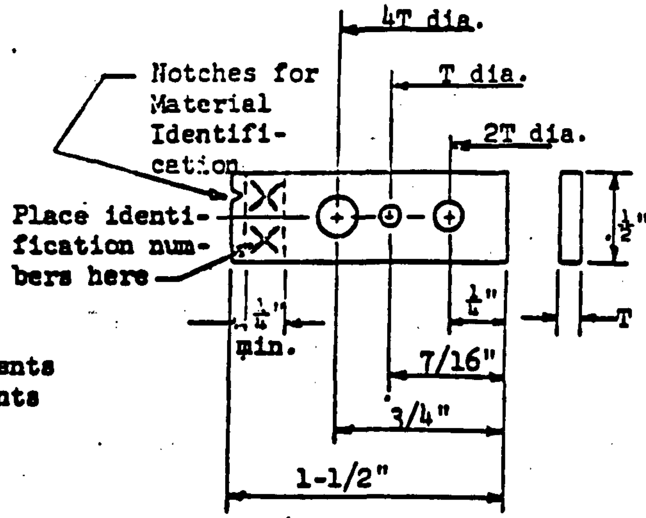
FIGURE 2
MIL STD 271 PENETRAMETERS

A. Thickness of penetrameter - 0.005" to and including 0.050".

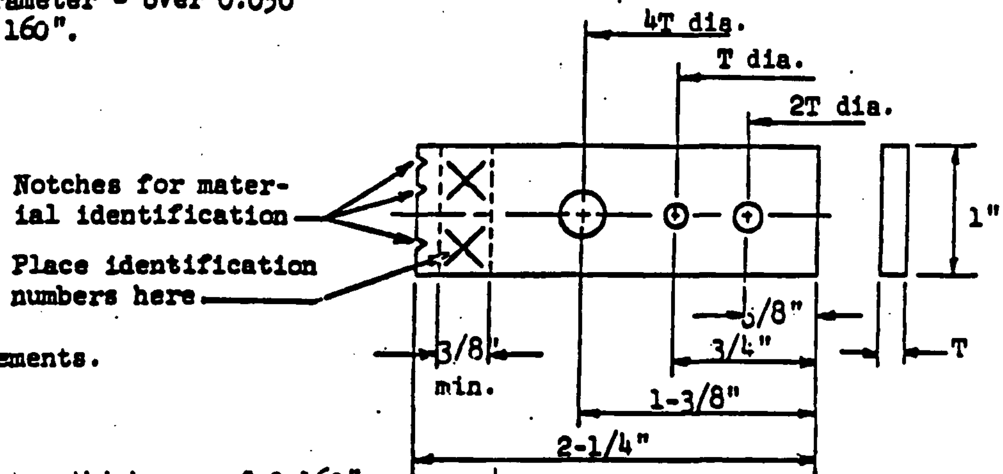
Min. penetrameter thickness (T)...0.005"
Min. dia. for 1T hole............0.010"
Min. dia. for 2T hole............0.020"
Min. dia. for 4T hole............0.040"

Holes shall be true and normal to the surface of the penetrameter. Do not chamfer.

From 0.005" to 0.020" made in 0.0025" increments
From 0.025" to 0.050" made in 0.005" increments

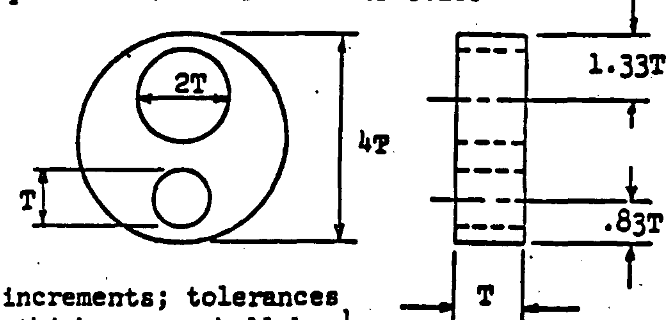


B. Thickness of penetrameter - over 0.050" to and including 0.160".



Made in 0.010" increments.

C. Design for penetrameter thickness of 0.160" and over.



made in 0.020" increments; tolerances of penetrameter thicknesses shall be ½ of the thickness increments between penetrameter sizes.

NOTES TO FIGURE 2

NOTE 1 - Tolerances on hole diameter shall be ± 10 percent.

NOTE 2 - Tolerances on penetrameter thickness shall be ± 10 percent or one half of the thickness increment between penetrameter sizes, whichever is smaller.

NOTE 3 - Penetrameters shall be identified by the minimum material thickness (TM) which they represent when radiographing at a 2 percent quality level. For example: A penetrameter identified with a 50 represents a minimum (TM) of 0.500".

NOTE 4 - Unless otherwise specified, the 0.005" penetrameter shall be used for material thicknesses less than 0.250".

NOTE 5 - Each penetrameter shall be identified by scribing or vibro tooling the material identification as stated below.

Material Notch Identification	No. of Notches
1. Steels - Fe	1
2. Copper-Nickel Alloy - CuNi	2
3. Nickel-Copper Alloy - NiCu	3
4. Nickel-Chromium-Iron Alloy - NiCr	4

THE BABCOCK & WILCOX CO...
BOILER DIVISION

QUALITY CONTROL SPECIFICATION

ISSUED	SUBJECT	SPEC. NO.
12-2-63	MAGNETIC PARTICLE INSPECTION AND ACCEPTANCE STANDARDS - WELDS	S-102B

1. SCOPE: This specification shall govern the dry and wet methods of magnetic particle inspection of welds, adjacent base materials for a minimum distance of 1/2" on each side of the welds when possible, and weld joint groove preparations for the detection of surface defects in magnetic materials. This procedure is in accordance with the ASME Boiler and Pressure Vessel Code and MIL-STD-271 and shall be used for contracts in which the following codes and regulations are applicable:

   A) ASME Boiler and Pressure Vessel Code
   B) MIL-STD-278 and 278A
   C) Coast Guard Regulations - CG115
   D) American Welding Society Codes

2. PREPARATION OF SURFACES:

   2.1 The surfaces to be inspected including 1" on each side of the weld or weld groove shall be free of slag, dirt, oil, grease, and loose scale.

   2.2 The surface finish of the welds or the weld grooves shall be such that proper interpretation can be accomplished. As-welded surfaces, following the removal of slag, shall be considered suitable without grinding if this does not interfere with interpretation of the test results and if the weld contour blends into the base metal without undercutting. However, when a weld is to be inspected in the final surface condition, the weld shall be free of sharp surface irregularities such as valleys between stringers. A surface finish of 1,000 micro inches shall be considered to meet this requirement for the dry method unless otherwise specified on the drawing. Where practical, a surface finish of 250 micro inches shall be considered acceptable for the wet method. All openings shall be plugged with a non-abrasive material that is easily removed to prevent accumulation of magnetic particles or other matter that cannot be completely or easily removed by washing and air blasting.

3. PROCEDURES FOR THE DRY METHOD: Finely divided magnetic particles shall be applied directly to the part being tested.

   3.1 Method of Magnetization:

   3.1.1 The direct method of magnetization using electrical contact electrodes (prods) to pass direct or rectified alternating current through the part under test shall be used when inspecting welds, partially completed welds, and weld groove preparations except as outlined in Paragraphs 3.1.2 and 3.1.3.

   3.1.2 The indirect method of magnetization using the electromagnetic yoke with alternating current or direct current passing through the yoke shall be used to inspect completed welds that have received a final heat treatment and were previously inspected by the direct magnetization method.

3.1.3 The direct method of magnetization using alternating current with the contact prod method may be used to inspect the root passes of fillet welds to evaluate indications obtained when inspecting with direct or rectified alternating current. This will insure that indications from the natural crevice at the base of the fillet welds are not treated as defects.

3.2 Equipment:

3.2.1 The magnetizing apparatus shall be capable of inducing in the item under test, a magnetic flux of suitable intensity in the desired direction. Contact electrodes (prods) and electromagnetic yokes shall be used. The size and type of contacts used and the time of application of current shall be such that overheating the part under test shall not occur either locally or generally.

3.2.2 The magnetic particles used for obtaining patterns of discontinuities shall be of non-toxic, finely divided ferromagnetic material of high permeability and low retentivity, free from deleterious rust, grease, paint, dirt, or other material that may interfere with proper functioning. Particles shall be of such size, shape, and color as to provide adequate sensitivity and contrast for the intended use.

3.2.3 De-magnetizing equipment shall consist of units such as open coil or box type demagnetizers or other means having the necessary capacity for demagnetizing.

3.3 Sequence of Operation:

3.3.1 Inspection shall be carried out by the continuous method; that is, the magnetizing current shall remain on during the period the magnetic particles are being applied and also while excess particles are being removed. The current shall not be turned on until after the electrodes have been properly positioned in contact with the surface and shall be turned off before the electrodes are removed. Current shall be applied for a minimum of 1/5 of a second.

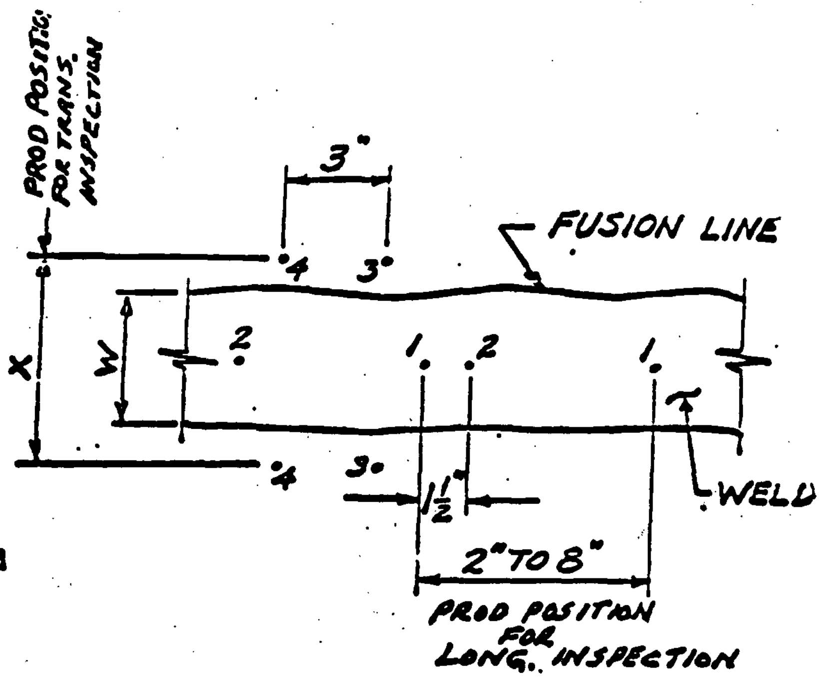
3.3.2 When the electrical contact prod method of magnetization is used, the magnetic field shall be induced with the prods placed parallel to the longitudinal axis of the weld for the detection of longitudinal defects. When the shape and configuration of the base material adjacent to the weld permit, the prods shall be rotated 90 degrees with respect to the original prod location in order to detect transverse defects. As a general rule, the effective width of the field is approximately half the distance between the prods. The prod spacings shall be a minimum of 2" and a maximum of 8". The preferred prod spacing to be used for the detection of longitudinal defects is five inches. The prod locations shall overlap approximately 1" to 1-1/2". See Figure 1.

FIGURE 1

Typical Example of
Prod Locations

1-1 } Prod locations to
2-2 } detect longitudinal defects

3-3 } Prod locations to
4-4 } detect transverse defects



X = W + 3
W = Width of weld
X = Prod spacing

The preferred distance between successive prod locations when inspecting for transverse defects is three inches. The magnetizing current whether direct, rectified alternating, or alternating current shall be 100 to 125 amperes per inch of spacing between the prods. Care shall be taken to prevent local overheating, arcing, or burning of the surface being inspected, particularly on steels subject to air hardening where hard spots or cracks could be produced by arc-burns caused by improper magnetizing techniques.

3.3.3 When the yoke method is used, the magnetic field shall be induced with the prods parallel to the longitudinal axis of the weld for the detection of transverse defects. When possible, the yoke shall be rotated 90 degrees with respect to the original location for the detection of longitudinal defects. The preferred distance between successive prod locations when inspecting for longitudinal defects is 3". When inspecting for transverse defects, the yoke prod locations shall overlap by approximately 1". Electromagnetic yokes using alternating or direct current shall have been qualified to be at least equivalent to that of the direct magnetization method when a minimum current of 25 amperes per inch of contact electrode spacing is used with a spacing of 3 to 6 inches. See Figure 2.

THE BABCOCK & WILCOX COMPANY
BOILER DIVISION
QUALITY CONTROL SPECIFICATION

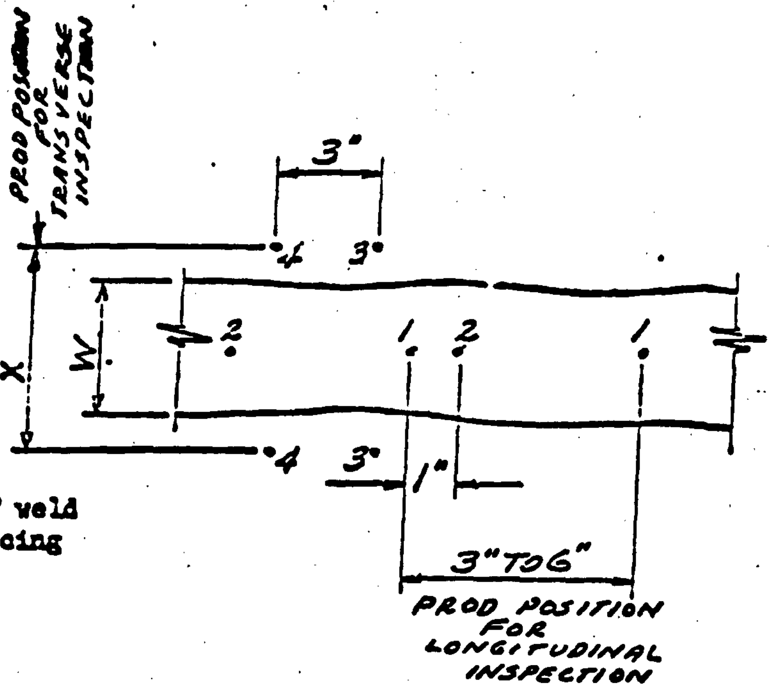
SPEC. NO.	SUBJECT	ISSUED
S-102B	MAGNETIC PARTICLE INSPECTION AND ACCEPTANCE STANDARDS - WELDS	12-2-63

FIGURE 2

Typical Example of
Yoke Prod Locations

1-1 } Prod locations to
2-2 } detect transverse defects

3-3 } Prod locations to
4-4 } detect longitudinal defects



X = W + 3
W = Width of weld
X = Prod spacing

3.3.4 The magnetic particles shall be applied in such a manner that a light uniform, dust-like coating settles on the surface under test. Excess particles shall be removed by means of a dry-air current of sufficient force to remove the excess without disturbing those that are indicative of discontinuities. Excess particles may be removed using an appropriate air stream generated by a hand-actuated rubber bulb or a hose with an inlet pressure as indicated in the table below. Nozzle size and air pressure shall result in a pressure of 1.0 to 1.5 and 0.6 to 0.3 inches of water at an axial distance from the nozzle of 1" to 2" respectively.

HOSE PRESSURE FOR REMOVING EXCESS DRY PARTICLES

Length of 5/16" ID Hose in Feet	Regulator Gage Reading in psi at Inlet End of Hose
25	1.25
50	1.75
75	2.75
100	3.50

3.3.5 When indications are detected, the prods shall be oriented in the best possible position to obtain the maximum sensitivity.

4. PROCEDURES FOR WET METHOD: Finely divided magnetic particles shall be suspended in a liquid vehicle for application to the part being tested.

4.1 Method of Magnetization:

4.1.1 With the circular method of magnetization, the item under test may be used as a conductor with the current passing through the item; or a central conductor may be placed inside the item with the current passing through the central conductor. When the item is used as a conductor, it is generally mounted horizontally between contact plates with suitable pressure to insure uniform magnetization.

4.1.2 With the longitudinal method of magnetization, the item under test is surrounded by a coil or solenoid and the current is passed through the coil or solenoid.

4.2 Equipment:

4.2.1 The magnetizing apparatus shall be capable of inducing in the item under test, a magnetic flux of suitable intensity in the desired direction by either the circular or longitudinal method. Direct current obtained from D. C. Generators, storage batteries, or rectifiers shall be used to induce the flux.

4.2.1.1 For the circular method of magnetization, a low voltage, high amperage current shall be passed through the item being tested or through a conductor that is inserted through the item being tested in order to induce the magnetic flux.

4.2.1.2 For the longitudinal method of magnetization, a solenoid, coil or magnet shall be used to induce the magnetic flux.

4.2.2 The magnetic particles used for obtaining patterns of discontinuities shall be non-toxic and shall be capable of exhibiting good visual contrast. The particles may be fluorescent when exposed to a filtered black light or non-fluorescent and shall be suspended in a liquid vehicle. Both type of particles shall not be used simultaneously.

4.2.2.1 The viscosity of the suspension vehicle shall be a maximum of 50 centistokes at any bath temperature used. Fluorescent particles shall be limited to 0.10 to 0.40 ounces of solid per gallon of the liquid vehicle. Non-fluorescent particles shall be limited to 1.0 to 1.4 ounces of solid per gallon of the liquid vehicle.

4.2.3 The liquid used as a vehicle for both the fluorescent and non-fluorescent magnetic particles shall comply with the following:

A) Petroleum distillate shall conform to either of the following:

1) P-S-661 - Commonly called "dry cleaning solvent"

2) VV-K-211 - Kerosene

B) Tap water with suitable rust inhibitors, wetting, and anti-foaming agents may be substituted for the petroleum distillate. The composition shall be approximately 0.3 per cent anti-foam agent, 3.9 per cent rust inhibitor, 12.8 per cent wetting agent, 83.0 per cent tap water.

C) Liquid vehicles used with fluorescent particles shall be non-fluorescent.

4.2.4 A darkened area or booth with a properly filtered black light source shall be provided for the fluorescent magnetic particle test.

4.2.5 De-magnetizing equipment shall consist of units such as open-coil or box-type demagnetizer or other means having the necessary capacity for demagnetizing.

4.3 Sequence of Operation:

4.3.1 Inspection shall be carried out by either the continuous or residual method after applying the suspensions to the item being tested by hosing or immersion to insure thorough coverage.

4.3.1.1 For the continuous method, the magnetizing circuit shall be closed just before diverting the suspension or just before removing the item from the suspension when immersion is used. The circuit shall remain closed for 1/5 to 1/2 second.

4.3.1.2 For the residual method, the item shall be magnetized by the application of current for at least 1/5 second after which the current shall be turned off and the suspension applied. For immersion application, care shall be exercised to avoid washing off the indications.

4.3.2 The magnetizing current for circular magnetization shall be 500 to 700 amperes per inch of diameter of the surface being tested. If both the inside and outside diameters of cylindrical parts are to be inspected, the larger diameter shall be used in establishing the current.

4.3.3 The magnetizing force for longitudinal magnetization shall be 3,000 to 4,000 ampere turns per inch of diameter of the surface being tested.

If both the inside and outside diameters of cylindrical parts are to be inspected, the larger diameter shall be used in establishing the current.

4.3.4 Suspensions shall be tested daily or when it appears that the suspension has become discolored by oil or contaminated by lint. The test shall be as follows:

A) Let pump motor run for several minutes to agitate a normal mixture of particles and distillate.

B) Flow the bath mixture through the hose and nozzle for a few minutes to clear hose.

C) Fill the centrifuge tube to the 100 cc line.

D) Place centrifuge tube and stand in location free from vibration.

E) Let tube stand for 30 minutes for particles to settle out.

F) After 30 minutes, readings for settled particles should be 1.7 to 2.4 cc for non-fluorescent pastes or .3 to 1.3 cc for fluorescent pastes. If reading is higher, add distillate; if lower, add paste.

5. DEMAGNETIZATION: Small parts shall be demagnetized between successive magnetizing operations to obtain satisfactory indications of discontinuities. All items shall be demagnetized prior to testing if the material contains strong residual magnetic fields from some previous operation. All items shall be demagnetized after testing if the residual field interferes with the removal of magnetic particles during cleaning.

6. INTERPRETATION OF RESULTS:

6.1 All indications revealed by magnetic particle inspections are not necessarily defects since irrelevant indications are sometimes encountered. Irrelevant or "false" indications are quite common, but may be easily identified. Examples of such indications are as follows:

Magnetic Writing: The indication is fuzzy and will be destroyed by demagnetization. These indications are caused by contact with other steel or magnets while magnetized.

Change in Section: The distribution of the magnetic field in an area of change in section of the piece being tested is such that the test pattern is broad and fuzzy.

Flow Lines: These are large groups of parallel indications that occur in some forgings when magnetized with high currents.

6.2 All indications believed to be non-relevant shall be evaluated by removing the surface roughness, or shall be reinspected by other nondestructive test methods. If reinspection reveals any indications, these indications shall be considered as relevant and shall be treated as defects and removed.

6.3 Linear defects are indications in which the length is greater than three times the width.

6.4 Rounded defects are indications which are circular or elliptical with the length less than three times the width.

7. ACCEPTANCE STANDARDS:

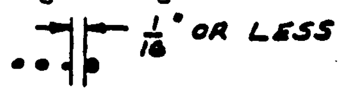
7.1 For Weldments tested contractually to ASME Boiler and Pressure Vessel Code Requirements:

7.1.1 Welds examined in accordance with this standard shall be free of the following defects:

(a) All cracks and linear defects.

(b) All rounded defects with a dimension greater than 3/16 inch.

(c) Four or more rounded indications in line separated by 1/16" or less, as measured from edge to edge.



(d) Ten or more rounded indications regardless of size located in any six square inch area whose minor dimension is no less than one inch, with these dimensions taken in the most unfavorable location relative to the defects being evaluated.

7.1.2 Base material defects detected during the inspection of welds shall be evaluated in accordance with the applicable base material specification. If the base material specification contains no acceptance standards, the acceptance standards for the welds shall apply.

Cracks and non-laminar defects on the surfaces of weld joint preparations shall be repaired. Linear indications that are parallel to the surface of the plate and are caused by laminations in plate material need not be repaired if they are:

(1) Equal to or less than 1" in length and are separated from other laminar type indications in the same line by at least 1/4".

(2) A group of aligned indications in a 1" length that are separated from another group of indications or a single indication in the same line by at least 1/4".

The depth of the repair of such defects in plate material shall be 3/8" from the surface of the weld preparation or the expected extent of the heat affected zone, whichever is greater.

7.2 For Weldments tested contractually to MIL-STD-271:

7.2.1 Welds examined in accordance with this standard shall be free of the following defects.

(a) All cracks, linear defects and other relevant defects.

7.2.2 Cracks and non-laminar defects on the surfaces of weld joint preparations shall be repaired. Linear indications that are parallel to the surface of the plate and are caused by laminations in plate material need not be repaired if they are:

1) Equal to or less than 1" in length and are separated from other laminar type indications in the same line by at least 1/4".

2) A group of aligned indications in a 1" length that are separated from another group of indications or a single indication in the same line by at least 1/4".

The depth of the repair of such defects in plate material shall be 3/8" from the surface of the weld preparation or the expected extent of the heat affected zone, whichever is greater.

8. ARC BURNS: When it is necessary to inspect areas that have been ground to remove arc burns on the welds or adjacent base materials, the dry powder method using the Type Y-5 yoke shall be used to inspect such areas. Arc burns made after final heat treatment or those that were not known to have been made prior to final heat treatment shall be etched with a 20 per cent solution of ammonium persulphate in water and all evidence of heat affected zone shall be removed. If removal reduces the wall thickness below the minimum requirements, a repair procedure shall be prepared and submitted to the customer for approval.

9. FINAL CLEANING: Magnetic particles shall be removed from all surfaces after the test. All temporary plugs shall be removed from holes and cavities.

10. PRECAUTIONS: To prevent arc flashes, electrodes shall be placed firmly on the surface to be inspected prior to turning on the current. For the same reason, the current shall be turned off prior to removing the electrodes.

ATTACHMENT C

PREOPERATIONAL BASELINE

INSPECTION SPECIFICATION

BLI - 1

6.1.5.1 Examination Schedule - This ultrasonic examination shall be performed after the hydrostatic test. Preparation of the clad surface may be done at any time during the fabrication sequence.

6.1.5.2 Areas to be Examined - An ultrasonic examination of the following areas of the reactor vessel and closure head shall be performed in accordance with the ASME Code for Inservice Inspection of Nuclear Reactor Coolant Systems. All vessel areas shall be examined through the I.D. clad surface and all closure head areas shall be examined through the outside surface.

a. All longitudinal shell and transition ring welds and at least one (1) plate thickness of adjacent base metal on both sides of the weld.

b. All circumferential shell, flange and transition ring welds and at least one (1) plate thickness of adjacent base metal on both sides of the weld.

c. All primary and safety injection nozzle to shell welds and all integral nozzle extensions inside the vessel.

d. All primary and safety injection nozzle to safe end welds.

e. Closure studs and nuts.

f. Ligaments between threaded closure stud holes.

g. Integrally welded vessel supports.

6.1.5.3 Surface Finish Requirements

a. The clad surface shall be prepared to the extent possible for a distance equal to 2T (T - wall thickness) on both sides of the welds to be examined, so that a meaningful ultrasonic examination can be performed. The crown of the weld overlay beads shall be flattened to the extent that visually discernible "valleys" are left between the beads. The adequacy of clad surface preparation for meaningful UT shall be evaluated in the following manner: place longitudinal mode, 2.0 (minimum) Mhz, search unit, either wheel type or contact type properly coupled on representative area of clad surface (ID) at a point 1/4T, minimum distance, from centerline of weld and mark spot for reference purposes. With sound beam directed radially toward OD surface, set reflection from OD surface (back reflection, BR) to 90% to 100% of full linear screen (Cathode Ray Tube screen, CRT) height. Scan along surface in normal fashion, starting at the marked reference spot and moving the search unit a distance equal to at least 1T away from, i.e., at 90° to, the weld. Scan again in the same manner a distance equal to at least 1T at 45° to the weld and then again, parallel to the weld, returning each time to the reference point to check the amplitude of the BR and to start the required scan. In each of the three scans, note the maximum reduction in BR amplitude in increments of 10%.

Surface conditions causing 30% and greater reductions in the calibration BR amplitude shall be considered inadequate for meaningful UT.

b. The supplier shall submit with their quotation a sample of the clad surface finish which he considers as acceptable for performing a meaningful ultrasonic examination of vessel welds and 1T of base metal as well as for cladding bond. The sample shall be at least 12" x 12" x 2" thick.

6.1.5.4 Examination requirements

a. Longitudinal wave testing based on discontinuity indication amplitude and depth (Distance - Amplitude - Correction) shall be performed on all areas defined in Paragraph 6.1.5.2. Discontinuity indication amplitudes shall be based on side drilled holes, as specified in Appendix IX-340 'Ultrasonic Examination of Welds' ASME Boiler and Pressure Vessel Code, in the representative, i.e., similar construction, configuration, surface finish, materials (clad, base and weld metal) and heat treat condition, reference blocks to be supplied.

   Longitudinal wave testing for area 6.1.5.2c will consist of two (2) perpendicular wave testing scans (from the inside surface of the nozzle and from the clad surface of the weld).

b. Shear wave testing based on discontinuity indication amplitude shall be performed for all areas indicated in Paragraph 6.1.5.2, except for area 6.1.5.2c.

6.1.5.5 Reportable Indications

The Supplier shall submit ultrasonic examination reports showing "reportable" indications. The locations of the "reportable" indications shall be shown on a chart or plan drawing of the material. For the reactor vessel examination (Paragraph 6.1.5), the "reportable" indications shall be located relative to vessel assembly reference surfaces and axes. Information relative to the magnitude and depth of the "reportable" indications shall be included. The definition of the "reportable" indications is given below.

6.1.5.6 Plate Material

A. Longitudinal Wave Examination

   1. Laminer defects "LT" which are defined as defects which cause a total loss of initial back reflection and which provide a reflection.

   2. Laminer defects "LP" which are defined as defects which lower the initial back reflection by more than fifty percent (50%), but less than one hundred percent (100%) and which provide a reflection.

3. Inclusion type defects "IT" which are defined as defects which cause a total loss of initial back reflection, but which do not provide a reflection.

4. Discontinuities which cause traveling indications. These shall be described with explanatory notes.

B. Shear Wave Examination

Indications whose amplitude equals or exceeds fifty percent (50%) of the calibration standard.

6.1.5.7 Forgings

The indications in Article 3 of Section III and ASTM-A388, Sec. 7, 1969 edition as "Reported for Information" and "Recording" respectively.

6.1.5.8 Cladding

Indications whose amplitude equals or exceeds fifty percent (50%) of that from the 3/4" diameter reference hole.

6.1.5.9 Reactor Vessel or Reactor Vessel Sub-Assemblies.

A. Longitudinal Wave Testing (Back Reflection)

When the back reflection is reduced to fifty percent (50%) or less of the initial calibrated back reflection due to the presence of discontinuity indications. The amplitude of the discontinuity indication as a percent of the initial calibrated back reflection shall be recorded in ten percent (10%) increments.

B. Longitudinal Wave Testing (Discontinuity Indication)

When the discontinuity indication equals or exceeds that from the reference discontinuity indication on the Distance Amplitude Curve.

C. Shear Wave Testing

Discontinuity indications whose amplitude equals or exceeds fifty percent (50%) of the calibration standard. The amplitude of the discontinuity indication as a percentage of the established calibration amplitude shall be recorded in ten percent (10%) increments.

PDS 9179

ISSUED	SUBJECT		SPEC. NO.
6-17-70		ULTRASONIC INSPECTION OF SIMILAR AND DISSIMILAR WELD SEAMS AND ATTACHMENT WELDS FOR THE PURPOSE OF MAPPING	BLI-1

BASE LINE INSPECTION

1. SCOPE: This specification shall govern the Ultrasonic inspection of similar and dissimilar weld seams and attachment welds for the purpose of mapping in accordance with ASME Boiler and Pressure Vessel Code, Sections III and XI.

2. EQUIPMENT: Ultrasonic inspection equipment shall consist of an electronic apparatus capable of producing, receiving, and displaying high frequency electrical pulses at the required frequencies and energy levels.

3. OPERATOR QUALIFICATION: The operator performing the inspection shall be qualified to Level II in accordance with SNT-TC-1A. The assistant shall be qualified to at least Level I in accordance with SNT-TC-1A.

4. SURFACE PREPARATION: The test surface shall be free of dirt, loose scale, machining or grinding particles, weld spatter, or other loose foreign matter. The surface finish shall be sufficiently smooth to maintain acoustical bond and minimize surface noise. A mill finish may be adequate for testing. Whenever necessary, surface conditioning shall be accomplished by available mechanical process, such as machining, grinding, sand blasting, or belt sanding to provide a suitable surface finish. Surface preparation shall consist of an area which includes the weld seam and two plate thicknesses on either side of weld edge. Both the inside and outside surfaces shall be prepared for testing where possible. The Ultrasonic Testing operator shall inspect the surface for suitability for testing.

5. COUPLANT: A suitable, liquid, semi-liquid, or paste couplant medium, such as water, oil, glycerin, or grease, shall be applied to the test surface.

6. AREA OF INTEREST: The inspection shall include the weld, weld fusion line, and one plate thickness beyond the fusion line of the weld.

7. LONGITUDINAL WAVE:

7.1 Base Material Test:

7.1.1 Test Procedure: An area one time the plate thickness of the base material along either side of the weld, shall be scanned with a normal beam to detect discontinuities that might affect the interpretation of angle beam results.

The instrument shall be adjusted to produce a back reflection from the opposite side of an indication free area of the base material, that is at least 75 percent but not greater than 90 percent of full scale amplitude.

7.1.2 If a base material condition exists which is indicated by a signal on the cathode ray tube from a discontinuity accompanied by a loss of back reflection that is 50% or less, the shear wave shall be conducted from both inside and outside surfaces where possible. If the opposite side is not accessable, a chart of this area shall be made and the shear wave inspection shall be performed on a best effort basis.

7.2 Longitudinal Wave Weld Test:

7.2.1 Calibration:

7.2.1.1 The calibration test block as shown in Figure 1 shall be used. The calibration block material shall be of an equivalent thickness and P-Number. P-Numbers 1, 3, 4 and 5 materials as listed in Table Q-11.1 of Section IX of the ASME Boiler and Pressure Vessel Code are considered to be equivalent.

7.2.2 Search Unit: The nominal frequency shall be 2.25 MHZ unless variables, such as production material grain structure, necessitate the use of other frequencies in order to assure adequate penetration.

7.2.3 Instrument Calibration for Thickness Greater than 1": The search unit shall be positioned to display the maximum amplitude from the calibration hole located at 1/4T test metal distance (T.M.D.) See Fig. 1 for calibration hole diameter.

The instrument sensitivity shall be adjusted to display an amplitude obtained from the calibration hole to 50% full screen height. The search unit shall be positioned to display the maximum amplitude from the calibration hole located at 3/4T without changing the instrument sensitivity. A reference curve shall be marked on the face of the C.R.T. by drawing a line between the peaks of the amplitude obtained as specified in Figure 2.

7.3 Instrument Calibration for Thicknesses up to 1" Inclusive:

The reference level shall be established from the calibration block shown in figure 3A. The instrument sensitivity shall be adjusted to display an amplitude of 75% of full screen height. A line shall be drawn across the face of the CRT Screen to cover the thickness being inspected.

7.3.1 Test Procedure: Shall be in accordance Section 7.4.

7.4 Test Procedure: The entire weld seam shall be inspected in accordance with the following paragraphs:

7.4.1 The entire weld seam shall be scanned in parallel paths.

7.4.2 To assure complete coverage of the inspection area, the search unit shall be indexed with an overlap of at least 10% between each pass.

7.4.3 The scanning rate shall not exceed 6" per second unless it can be demonstrated that proper screen interpretation can be obtained at greater speeds.

7.4.4 When possible, scanning shall be performed at a minimum sensitivity level of two (2) times the calibration level. The sensitivity level for evaluation of indications shall be performed in accordance with Paragraph 7.2.3.

7.4.5 When an indication is observed, the search unit shall be positioned to obtain the maximum amplitude possible. The size and location of discontinuities shall be recorded on form shown in Figure 4 as the test progresses.

7.4.6 All conditions such as scanning speed, search unit and couplant used during calibration shall be recorded and duplicated during inspection and evaluation.

7.4.7 Verification of sound penetration through the weldment shall be accomplished by observing the back reflection from the opposite parallel surface or obtaining a back reflection on acoustically similar material using a test metal distance within $\pm$ 10% of the weldment being inspected.

7.4.8 Where the configuration of the part being tested will not permit inspection as described in Section 7, the inspection shall be performed to the maximum extent that geometric configuration will permit. A chart shall be maintained which documents inaccessible areas.

8. SHEAR WAVE ANGLE BEAM:

8.1 Calibration:

8.1.1 The calibration test block as specified in Paragraph 7.2.1.1 and Figure 1 shall be used.

8.2 Search Unit: A 2.25 MHZ frequency angle beam search unit not exceeding one square inch maximum effective area shall be used. The search unit beam angle shall provide an angle in the production material in the range of 40 degrees to 75 degrees. If sound beam penetration cannot be obtained when using 2.25 MHZ, then a test frequency of 1.0 MHZ shall be used. In the event the 1.0 MHZ is used, recalibration shall be performed.

8.3 Calibration:

8.3.1 Instrument calibration for thicknesses up to 1" inclusive. A distance amplitude correction curve shall be established by using the calibration block as shown in Figure 3A. The first point on the curve shall be established by placing the search unit in the 6/8 node position as shown in Figure 3A. The search unit shall be positioned to obtain the maximum response from the calibration hole and the instrument sensitivity shall be adjusted to display an amplitude of 75% full screen height. At this sensitivity level, the search unit shall be placed at the respective 10/8 and 14/8 node positions and the peaks of the amplitude obtained marked on the face of the cathode ray tube (C.R.T.). A line shall be drawn between the peaks of the amplitude signals obtained. The length of the line shall cover the thickness being examined.

8.3.2 Instrument calibration for thickness over 1". A distance amplitude correction curve shall be established by using the calibration block as shown in Figure 3. The first point on the curve shall be established by placing the search unit in the 3/8 node position for materials 1" through 6" as shown in Figure 3. For thicknesses over 6" the first point on the curve shall be established by placing the search unit in the 1/8 node position as shown in Figure 3. The search unit shall be positioned to obtain the maximum response from the calibration hole and the instrument sensitivity shall be adjusted to display an amplitude of 75% full screen height. At this sensitivity level the search unit shall be placed at the respective 1/8 node if applicable, the 3/8, 5/8 and 7/8 node positions and the amplitude obtained marked on the face of the C.R.T. A line shall be drawn between the peaks of the amplitudes obtained. The length of the line shall cover the thickness being examined.

8.3.3 Instrument calibration for areas where sound will be reflected from a clad surface shall be: Obtain signal response of 20% from 1/2 node cladded interface as shown in Figure 3. At this sensitivity level, the search unit shall then be placed at the 1/8, 3/8 and 7/8 node positions and the amplitude obtained marked on the face of the C.R.T. Any indication at this sensitivity level shall be evaluated by readjusting the sensitivity level to contain the DAC evaluated within the dynamic range of the instrument. Example: Indication seen before 1/2 node position re-calibration shall be performed as described in 8.3.2. Any indication noted beyond 1/2 node position shall be evaluated as described in 8.3.3.

8.4 Test Procedure:

8.4.1 Testing of the weld shall consist of indexing the search unit to obtain a 10% minimum overlap for each scan to include the weld fusion zone. The speed of scanning shall be at a uniform rate commensurate with the ability to accurately interpret and evaluate all indications resulting from discontinuities. The following scanning patterns shall be used:

8.4.1.1 The transducer shall be moved in an oscillating fashion, backward and forward at a 90° angle from the longitudinal axis of the weld, a sufficient distance over the test surface to project sound through the entire volume of weld metal as established on the referenced standard. The sound beam shall be directed toward the longitudinal axis of the weld. The search unit shall be moved in this progression along the length of the weld until the entire weld is traversed.

8.4.1.2 The total length of the weld seam shall be scanned in two opposite directions by moving the search unit in an oscillating fashion along the centerline of the weld. The oscillation shall consist of rotating the transducer so that the sound beam is scanning a fan shaped area. The included angle of oscillation shall be approximately 30°.

8.4.1.3 When possible, scanning shall be performed at a minimum sensitivity (gain) setting of two times the reference level sensitivity.

8.4.1.4 The sensitivity level shall be verified by comparing D.A.C. to calibration standard at least once each hour of operation and every time the machine is moved or a power failure results.

8.4.1.5 Where the configuration of the piece being tested will not permit inspection as noted in Section 8, the inspection shall be performed to give the best possible inspection and documentation.

9. RECORDING:

9.1 All discontinuities located within the weld, or weld fusion zone which produce an amplitude greater than 20 percent of the calibration level (DAC) and judged acceptable in accordance with Paragraphs 7.2 and 8 of this specification shall be recorded and charted.

9.1.1 The chart shall show location and size of discontinuities.

9.1.2 Amplitudes of discontinuity indications shall be recorded in 10% increments.

9.1.3 The depth of discontinuities shall be recorded.

9.1.4 Any area where best effort inspection was performed due to configuration shall be recorded and reported. Method of inspection in these areas shall be documented.

10. ACCEPTANCE STANDARDS:

10.1 All indications which produce a response greater than 20 percent of the DAC reference level shall be investigated to the extent that the operator can determine the shape, identity, and location of all such reflectors and evaluate them in accordance with the following:

The search unit shall be placed in a position which produces the maximum amplitude. If the maximum amplitude equals or exceeds 75% of the DAC reference calibration curve, it shall be reported.

If the maximum amplitude exceeds 20% but is less than 75% of the DAC reference calibration curve, it shall be noted as supplemental information, and supplied to the owner in a separate supplemental report.

10.2 Any discontinuity detected in the weld or weld fusion zone which produces a signal amplitude greater than the DAC reference calibration curve and that has a linear dimension as follows shall be reported and an evaluation made to the acceptance standards involved in the original construction. Thus, a discontinuity in a weld would be evaluated using the fabrication radiographs whereas a discontinuity in a plate would be evaluated to the U.T. acceptance standards for plate, etc.

(A) 1/4" for thicknesses up to and including 3/4".

(B) 1/3" of the thickness for thicknesses over 3/4" to and including 2 1/4".

(C) 3/4" for thicknesses over 2 1/4".

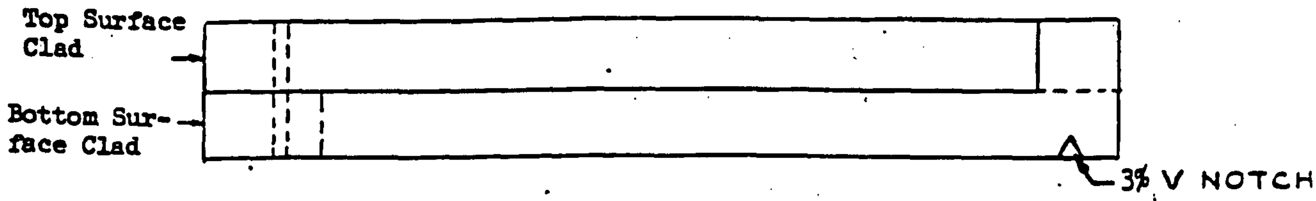
If there is any doubt regarding the proper interpretation of Ultrasonic result of the rejected areas, such doubt may be resolved by radiography.

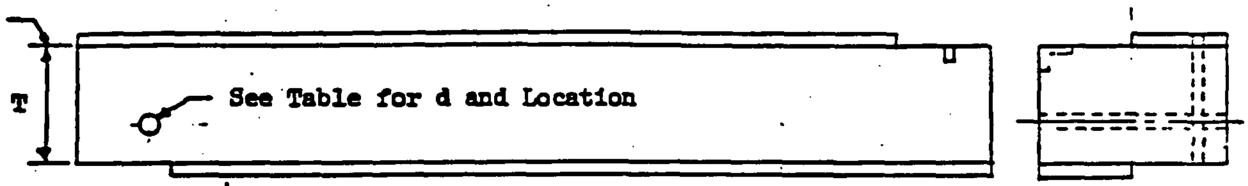
11. RECORD OF TEST RESULTS: A copy of the test results shall be made available to the Customer with the following information: (Figure 4)

A. Contract Number
B. Operator & Assistant
C. Instrument
D. Method of Test
E. Couplant
F. Type & Size of Crystal
G. Block Number
H. Chart of results obtained in para. 7 & 10.
I. Specification
J. Date

NOTE: It is recognized that Ultrasonic examination cannot define the finite size of discontinuities in the path of the beam but rather indicates the sum of the total reflectors in the beam path at a given interval. It is also recognized that, due to beam spread, the recorded signal will always be larger than the reflector causing the signal. No attempt shall be made to correct for this exaggerated signal since it will simplify subsequent In-Service examinations if the received signal is recorded uncorrected.

ISSUED 6/17/71	SUBJECT ULTRASONIC INSPECTION OF SIMILAR AND DISSIMILAR WELD SEAMS AND ATTACHMENT WELDS FOR THE PURPOSE OF MAPPING	SPEC. NO. BLI-1





BASIC CALIBRATION BLOCK

L = Length of block determined by the angle of search unit and the node used for examination of Production Materials.
T = Thickness of basic calibration block (See Table below).
D = Depth of Basic Calibration hole (See Figure)
d = Diameter of Basic Calibration hole (See Table below).
t = Nominal production material thickness.

Production Material Thickness (t)	Basic Calibration Block Thickness (T)	Hole Location	Hole Diameter (d)
Up to 1" Incl.	3/4" Or t	1/2 T	3/32"
Over 1" thru 2"	1/2" or t	1/4 T	1/8"
Over 2" thru 4"	3" or t	1/4 T	3/16"
Over 4" thru 6"	5" or t	1/4 T	1/4"
Over 6" thru 8"	7" or t	1/4 T	5/16"
Over 8" thru 10"	9" or t	1/4 T	3/8"
Over 10"	t	1/4 T	See Note 1

[1] For each increase in thickness of 2 inches or fraction thereof, the hole diameter shall increase 1/16 inch.

Figure 1

ISSUED	SUBJECT		SPEC. NO.
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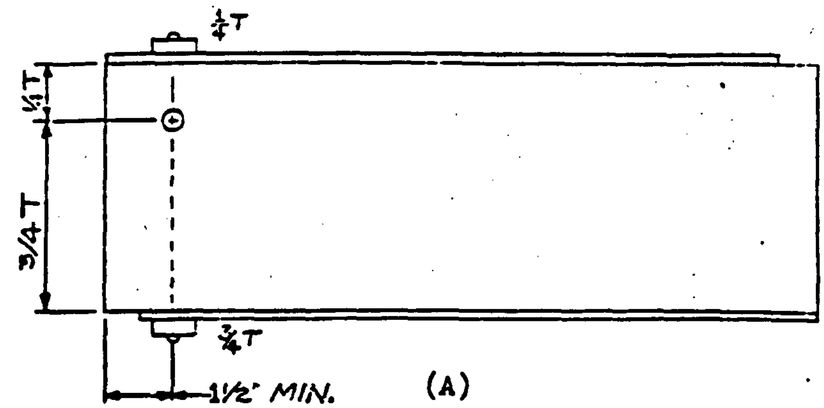
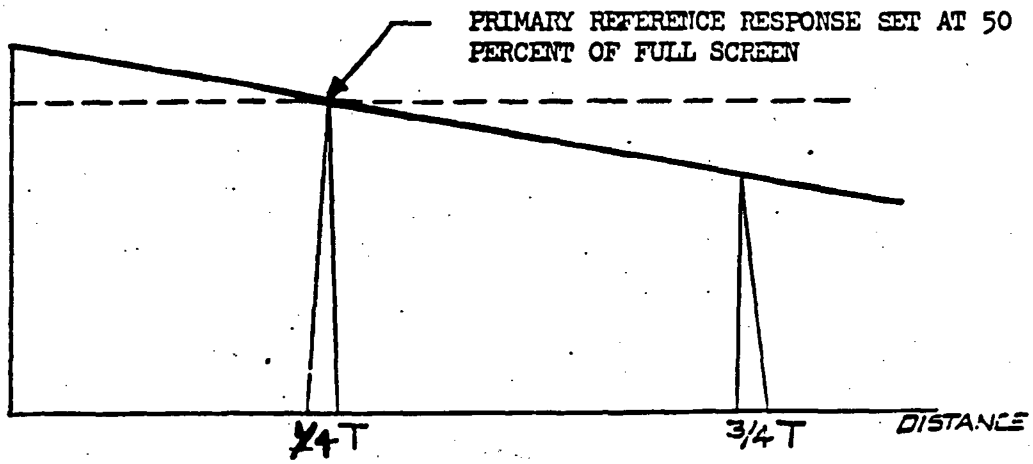
FIGURE 2

TYPICAL DISTANCE AMPLITUDE CORRECTION CURVE
(STRAIGHT BEAM METHOD)

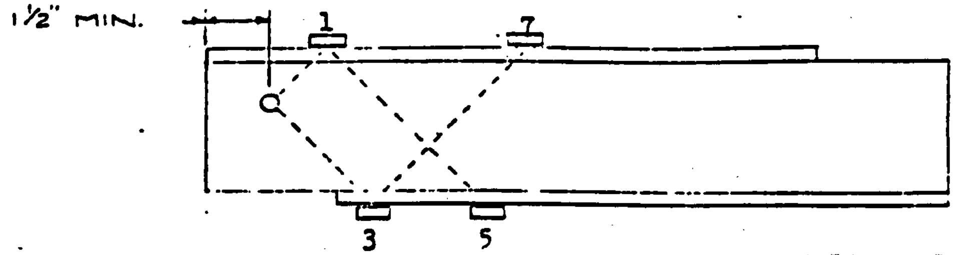


(A)

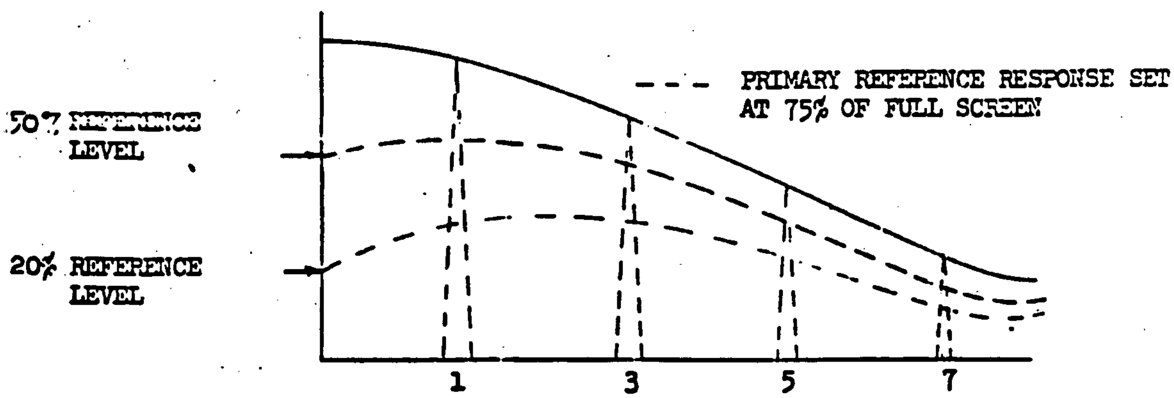


ISSUED 6/17/71	SUBJECT ULTRASONIC INSPECTION OF SIMILAR AND DISSIMILAR WELD SEAMS AND ATTACHMENT WELDS FOR THE PURPOSE OF MAPPING	SPEC. NO. BLI-1

Node Positions



FOR THICKNESS OVER 1 INCH



TYPICAL DISTANCE AMPLITUDE CORRECTION CURVE

FIGURE 3

REV. NO. 3	REV. BY GDE/GAF	REVISION SEE PAGE 1	SPEC. NO. BLI-1
REVISION DATE 2/25/71			PAGE NO. 11 OF 13

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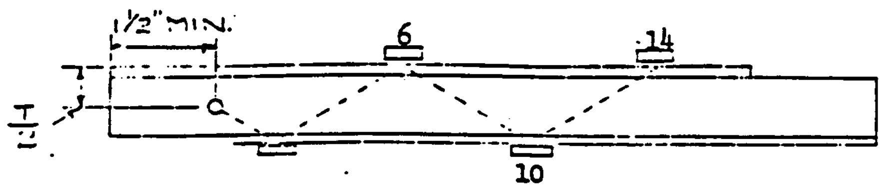
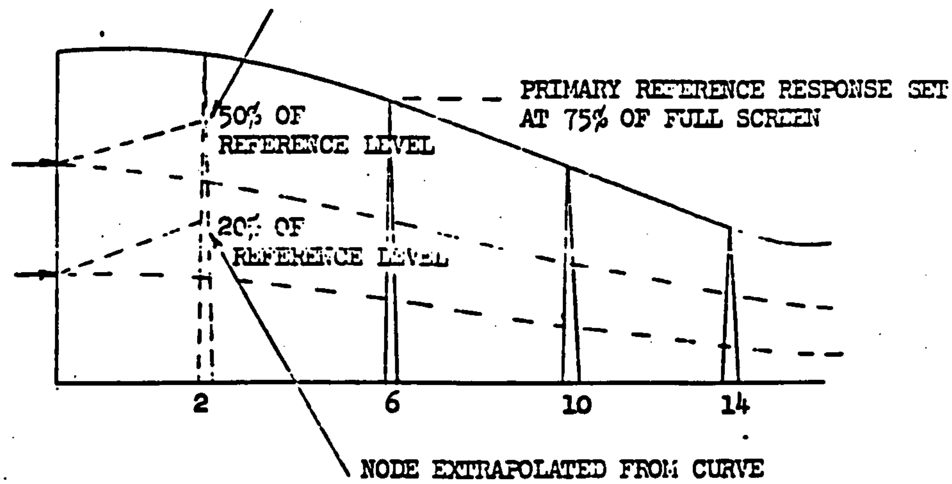


FOR THICKNESS 1 INCH OR LESS



Typical Distance Amplitude Correction Curve (Angle Beam Method)
(Distance in eighths of a node. For Example, 14 is 14/8 node).

FIGURE 3A

ISSUED	SUBJECT	SPEC. NO.
6/17/70	ULTRASONIC INSPECTION OF SIMILAR AND DISSIMILAR WELD SEAMS AND ATTACHMENT WELDS FOR THE PURPOSE OF MAPPING	BLI-1

ULTRASONIC TEST DATA

STRUCTURE :

MARK NO. | TO MARK NO. | PROCEDURE : | I.S.I. ITEM NO.

POSITION A	POSITION B	LENGTH	DEPTH	BACK REF.	MAX. SIG.	BEAM DIRECTION	INDICATED DEFECT TYPE	ANGLE	DISTANCE FROM A	DISTANCE FROM B	DISTANCE FROM 1	DISTANCE FROM 2	SURFACE	STATUS

COMPONENT :

MATERIAL :

THICKNESS :

TEST SURFACE :

REMARKS

Ⓐ — DISTANCE BETWEEN POINTS → Ⓑ =

INSTRUMENT : | CAL. BLOCK :

TRANSDUCER TYPE : | SIZE : | FREQ.

ANGLES : | MODE :

BEAM DIRECTION :

NUMBER POSITIONS : | DISTANCE :

REFERENCE POINTS :

DATE : | PERFORMED BY :

SKETCHES :

FIGURE 4

REV. NO.	REV. BY	REVISION	SPEC. NO.
3	GDE/GAF	SEE PAGE 1	BLI-1
REVISION DATE 2/25/71			PAGE NO. 13 OF 13

ATTACHMENT D

REPAIR WELDING PROCEDURE
AND PROCEDURE QUALIFICATION
TEST REPORT FOR THE REPAIR OF
ZION 1 VESSEL

BABCOCK & WILCOX	W	3	M	2	R	

WELDING PROCEDURE

MANUAL METAL ARC PROCESS

ISSUED BY: V. O'NEAL	
DATE: 1-7-71	
CONTRACT NO. 610-0144	
SERIAL NO. C8-0144-1	
BASE METAL TYPE: Mn-Mo-Ni P-3 (Q & T) P-12B	MARK NO. MK. 8
	UNIT NO. 1
DESCRIPTION OF WELD: Repair of Weld Using Half	SEQUENCE NO.
Bead Technique	WLD. SPEC. W-6 REV 3
First Two Layers Only	W ALT REV

STEP NO. 1

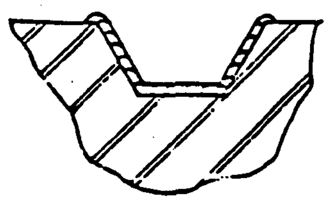
Record actual cavity Dimensions. Deposit first layer using 3/32" Dia. E8015-8018C3 Electrode.

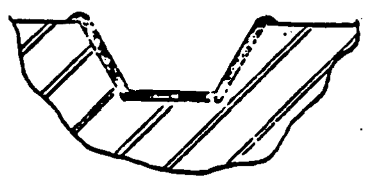
STEP NO. 2

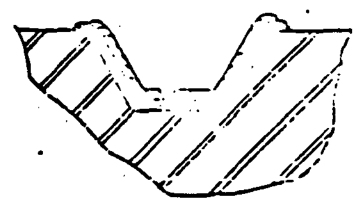
Establish and record thickness of first layer. Remove by grinding 1/2 of this thickness. Verify by dimensional inspection MT after grinding.

STEP NO. 3

Deposit second layer using 1/8" Dia. E8015-8018C3 Electrodes. MT second layer.







WELD BEAD NO.	1st Layer	2nd Layer			
WELDING POSITION	Flat-Horiz.	Flat-Horiz.			
ELECTRODE: TYPE	E8015-8018C3	E8015-8018C3			
SIZE	3/32"	1/8"			
LOT NO.					
WLD'G CURRENT TYPE	D.C.-R.P.	D.C.-R.P.			
AMPS	75-110	110-155			
PREHEAT TEMP. °F	350	350			
INTERPASS TEMP. °F	500	500			
BEAD WIDTH (MAX")	4 X DIA.	4 X DIA.			
P.Q. - MT. VERNON					
P.Q. - BARBERTON	2692				

N.D.T. REQUIREMENTS: MT First Layer After Grinding in Step 2 and Second Layer After Deposition in Step 3

POSTHEAT REQUIREMENTS: Maintain 350° F. Preheat

See Sheet 2

WLD. ENG. O'NEAL | PAGE 1 OF 2

BABCOCK & WILCOX

WELDING PROCEDURE

MANUAL METAL ARC PROCESS

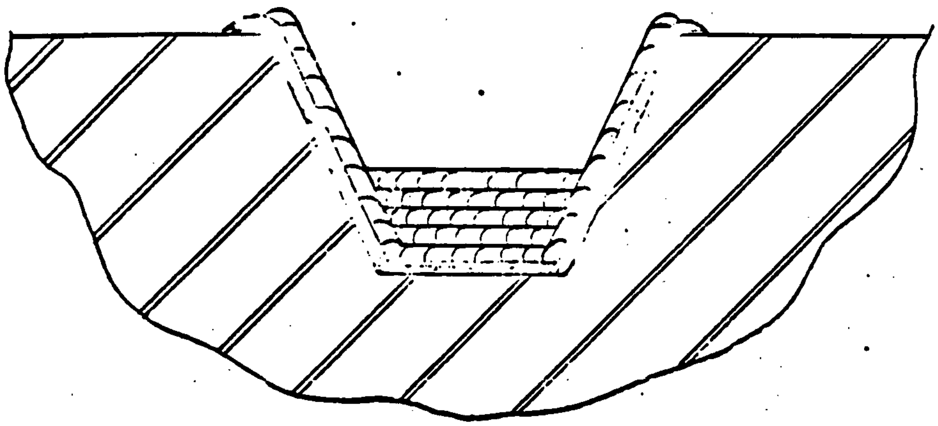
W	3	M	2	R	

ISSUED BY: V O'NEAL
DATE: 1-7-71
CONTRACT NO. 610-0144
SERIAL NO. C8-0144-1
MARK NO. MK. 8
UNIT NO. 1
SEQUENCE NO.
WLD. SPEC. WS-69 REV 5
W ALT REV

BASE METAL TYPE: Mn-Mo-Ni P-3 (Q & T) P-12B

DESCRIPTION OF WELD: Repair of Weld

See Sheet No. 1



WELD BEAD NO.	---	---	---		
WELDING POSITION	Flat -Horiz	Flat-Horiz	Flat		
ELECTRODE: TYPE	E8015-18C3	E8015-18C3	E8015-18C3		
SIZE	1/8 DIA.	5/32 DIA.	3/16 DIA.		
LOT NO.					
WLD'G CURRENT TYPE	D.C.-R.P.	D.C.-R.P.	D.C.-R.P.		
AMPS	110-170	150-210	180-360		
PREHEAT TEMP. °F	350	350	350		
INTERPASS TEMP. °F	500	500	500		
BEAD WIDTH (MAX")	4 X DIA.	4 X DIA.			
P.Q. - MT. VERNON					
P.Q. - BARBERTON	2492				

N.D.T. REQUIREMENTS: MT Root Layer and Each 1/4" of Weld Deposit.
MT, RT, UT - Completed Weld Repair Prior to and After Final P.W.H.T.

POSTHEAT REQUIREMENTS: After Welding is Completed, Remove All Stress Risers. Heat to 500°F, Hold for Two Hours Then Slowly Cool to Room Temperature After Initial NDT Weld Repair Area to Be P.W.H.T. 1100/1150 1Hr/In. Thickness In Accordance with Specification Procedure.

WLD. ENG. O'NEAL | PAGE 2 OF 2

THE BABCOCK & WILCOX COMPANY
BARBERTON, OHIO

CONTRACT NO. 610-0144
SPECIFICATION NO. [illegible]-50
Code No. [illegible]

1-6
[illegible]ORD OF PROCEDURE AND/OR OPERATOR QUALIFICATION TEST-QC 2E4-122

[illegible] ☐	ASME SECTION 3 ☒	ASME SECTION 1 & 8 ☐	OTHER ☒ ASME Sect. IX	PLATE ☒ PIPE ☐ ☐
SINGLE PASS ☐	MULTIPLE PASS ☒			

WELDING PROCESS: [illegible]anual Metal Arc	QUALIFICATION POSITION: Flat & Horizontal	METAL THICKNESS: 2 IN.
SINGLE LAYER ☐	MULTIPLE LAYER ☒	
MATERIAL SPECIFICATION: -3 A-302 Gr-B Q & T (P-12B)	SINGLE ARC ☒	MULTIPLE ARC ☐
HEAT TREATMENT: [illegible] hours at 1100° - 1150°F	MACRO. EXAM. N.R.	MICRO. EXAM. N.R.

FLUX NAME OR COMPOSITION: .A.	PREHEAT TEMPERATURE: 350°F MIN.	INTERPASS TEMPERATURE: 500°F MAX.	WELDER SYMBOL NO.: Saltzman 12[illegible], Hite 686, Jones 630
TYPE OF BACKUP (STRIP OR GAS) AND COMPOSITION: Strip P-1	FILLER METAL GROUP NO.: F-4	SHIELDING GAS: N.A.	CUP SIZE: N.A. — TORCH GAS FLOW RATE: N.A.

AMPS, VOLTS, CURRENT, POLARITY: 85-425 Amps., 20 - 26 Volts. D.C.R.P.

SIZE OF ELECTRODE, IN. DIA.: 3/32, 1/8, 5/32 — SIZE OF FILLER WIRE, IN. DIA.: 1/4	ELECTRODE EXT. BEYOND CUP IN.: N.A	TRAVEL SPEED IPM: N.A.	WIRE FEED, IPM: N.A.	OSCILLATION: N.A.	[illegible]: N.A

LIQUID PENETRANT: N.R.	VISUAL INSPECTION: Acceptable

MAGNETIC PARTICLE: Build-up for half bead, Final Surface -Acceptable	RADIOGRAPH: Final Weld Acceptable	ULTRASONIC TEST: N.R.

REMARKS: Procedure qualification for repairing P-3 material using the half-bead technique with M.M.A. using 3/32 and 1/8 B&W 8015 electrodes. See reverse side for groove configuration and additional information. Valid for P-12B Material.

CHEMICAL ANALYSIS - % [illegible] NO. ______

[illegible]	[illegible]	C	MN	P	S	SI	CR	NI	MO	FE	CU	CB	CO	[illegible]	[illegible]	[illegible]
							N.A.									

REDUCED SECTION TENSILE (TRANSVERSE TO WELD)

SPECIMEN NO.	DIMENSIONS, INCHES WIDTH	THICKNESS	AREA SQ. IN.	ULTIMATE LOAD LBS.	ULTIMATE TENSILE STRENGTH PSI	FRACTURE LOCATION
[illegible] 178-1A	.928	1.860	1.72[illegible]	14[illegible]	[illegible]2,5[illegible]	Base Metal
[illegible] 178-1B	1.011	1.855	1.875	155000	[illegible]4,27[illegible]	Base Metal

BEND TEST

[illegible] 4 transverse side bends (split into [illegible]) - Acceptable
[illegible] with no defects. 1 with one 1 [illegible]" indication.

ALL WELD METAL TENSILE

SPECIMEN NO.	DIAMETER, IN.	AREA SQ. IN.	YIELD POINT PSI	TENSILE STR. PSI	ELONG % IN 2"	RED. AREA %
			N.R.			

TYPE CHARPY [illegible] IMPACT TEST AT +10 °F 240 FT. LBS. ENERGY LOAD

METAL				FT. LBS.		FT. LBS.
METAL [illegible]	91	98	76	FT. LBS.		FT. LBS.
AFFECTED ZONE	101	84	68	FT. LBS.		FT. LBS.

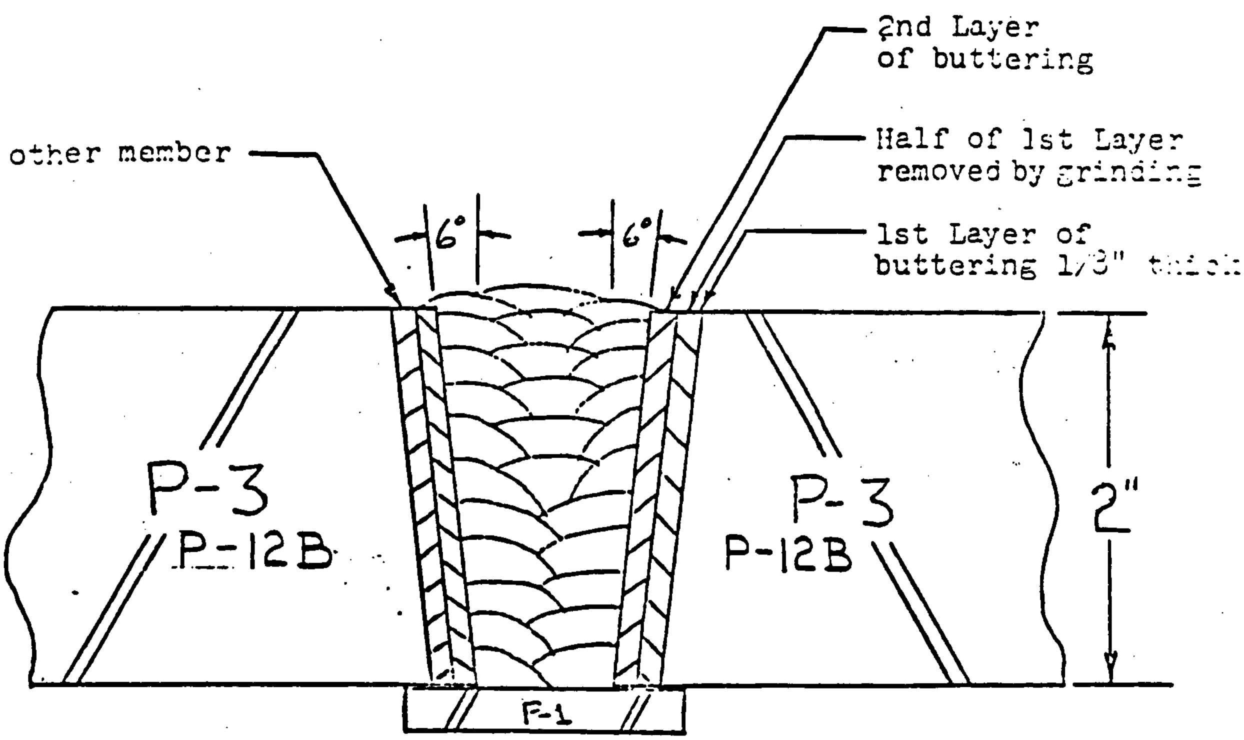
WE [illegible] CERTIFY THAT TO THE BEST OF OUR KNOWLEDGE THE STATEMENTS MADE IN TH[illegible] RECORD ARE CORRECT AND THAT THE TEST WELDS WERE PREPARED, WELDED AND TESTED IN ACCORDANCE WITH THE APPLICABLE SPECIFICATIONS.

N.R. = NOT REQUIRED N.A. = NOT APPLICABLE

PO 26[illegible]
N.[illegible]-178C[illegible]

WITNESSED [illegible] - [illegible] - [illegible] DATE 1[illegible]-17-7[illegible]
WITNESSED N. [illegible] - [illegible] - [illegible]

BY B. J. Frisbie [illegible]
BABCOCK & WILCOX



HALF BEAD REPAIR TECHNIQUE

Position of Welding - Flat and Horizontal

First Layer of Buttering
Electrode - 3/32" dia. E-8015 (B&W)
Welding Current and Polarity - 85-110 Amps., D.C.R.P.
Welding Voltage - 20-24 Volts
First Layer was 1/8" thick
One-half (1/16") of this was removed by grinding.

Second Layer of Buttering
Electrode - 1/8" dia. E-8015 (B&W)
Welding Current and Polarity - 130-150 Amps., D.C.R.P.
Welding Voltage - 20-25 Volts

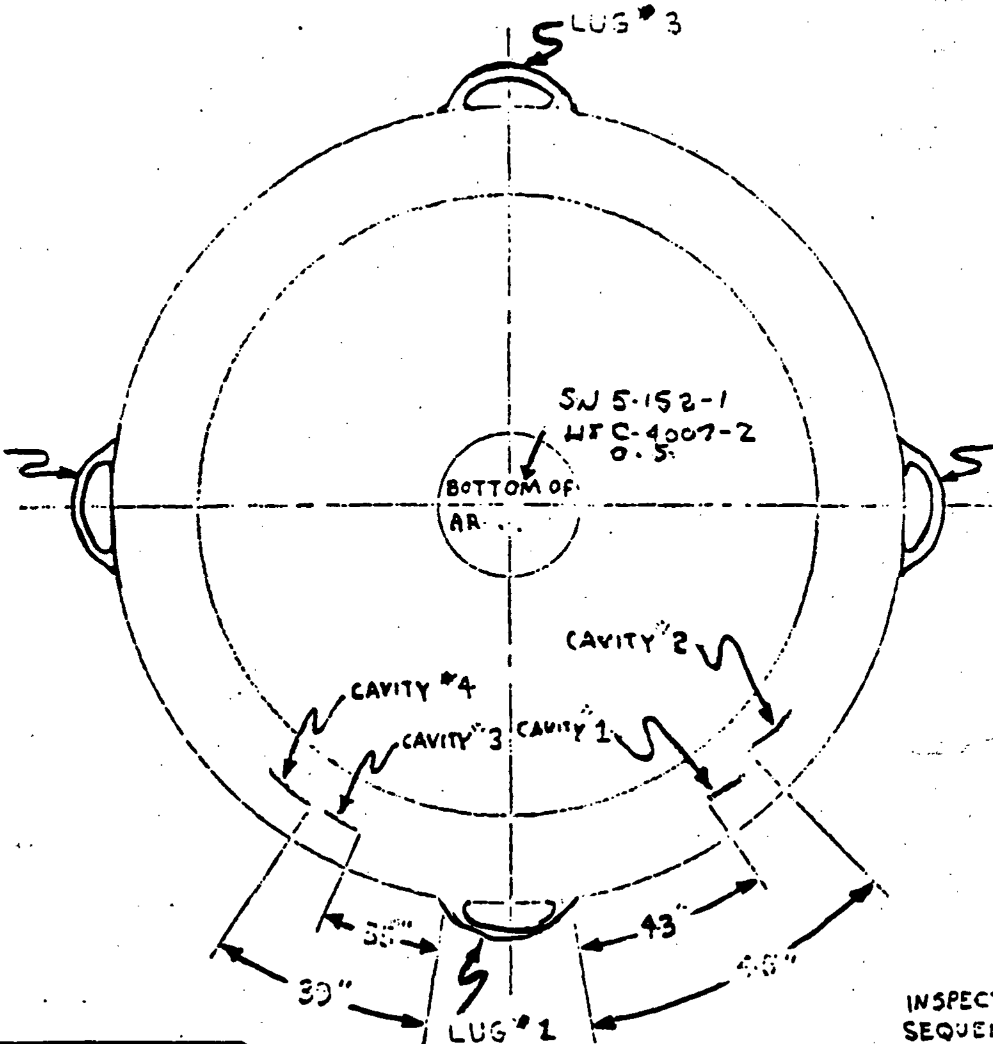
Groove Weld
Electrode - 5/32" dia. E-8015 (B&W) and 1/4" dia. E-8018 (RACO)
Welding Current and Polarity - 150-170 Amps. & 325-425 Amps. respectively DCRP
Welding Voltage - 20-26 Volts

Date 12-17-70

P.Q. 2[illegible]
MTV-17[illegible]

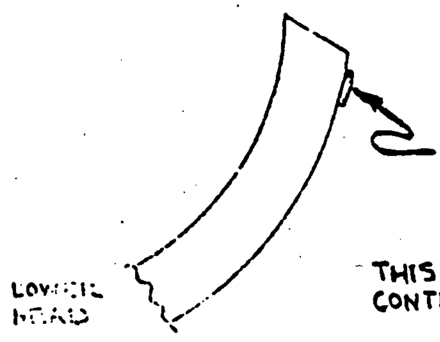
QUALITY CONTROL
INSPECTION

BABCOCK & WILCOX POWER GENERATION DIVISION BARBERTON, OHIO



INSPECTION AFTER SEQUENCE 224 ON SHEET 1 OF 2 OF DEVIATION DATED 7-15-69 REFERENCE REJECT TICKET # K-6916

CAVITIES			
NR	LENGTH	WIDTH	DEPTH
1	2 1/2"	3/8"	5/32"
2	4"	3/8"	3/8"
3	4"	5/8"	3/8"
4	3 1/2"	3/8"	5/16"
5	1 1/2"	1/4"	1/8"



CAVITY # 5 IS ON THE O.D. OF THE HEAD 12" FROM LUG # 1 TO THE RIGHT

THIS HEAD TO BE USED ON 610-0146 CONTRACT PER LETTER DATED JUNE 13 1969

WESTINGHOUSE

JOB. NO. 610-0162-51-1

DWG. NO. 133332E-2 MKP 5

BY GARY STEWART

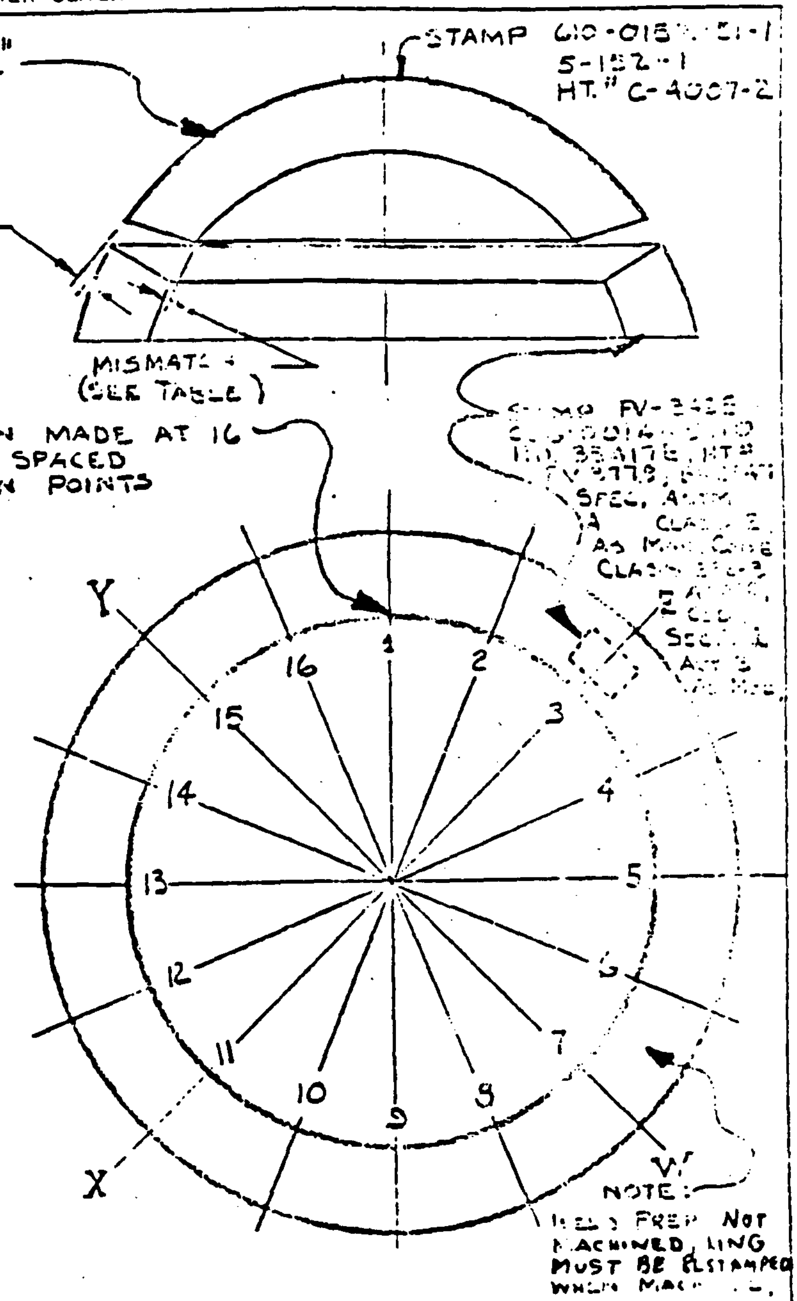
DATE 7-24-69

ACCEPT (M.T. ACCEPT TICKET # K-3390)

BDS 1243-3

QUALITY CONTROL
INSPECTION

BABCOCK & WILCOX POWER GENERATION DIVISION BARBERTON, OHIO



OUTSIDE DIA.			INSIDE DIA.		
NO.	WELD GAP	MIS-MATCH	NO.	WELD GAP	MIS-MATCH
1	1 11/16	9/16	1	[illegible]	7/16"
2	1 11/16"	11/16"	2	2 3/4"	3/16"
3	1 11/16"	5/8"	3	2 3/4"	[illegible]
4	1 11/16"	1/2"	4	2 3/4"	1/4"
5	1 11/16	3/8"	5	2 3/4	3/8"
6	1 11/16"	1/2"	6	2 11/16"	3/8"
7	1 11/16"	5/8"	7	2 11/16"	7/32
8	1 11/16"	9/16"	8	2 11/16"	[illegible]
9	1 11/16"	5/8"	9	[illegible]	3/32"
10	1 11/16"	3/4"	10	2 3/4"	1/16"
11	1 11/16"	3/8"	11	2 3/4"	1/16"
12	1 11/16"	5/8"	12	[illegible]	1/16
13	1 11/16	1/2"	13	2 3/4"	3/32"
14	1 11/16	1/2"	14	2 11/16	1/4"
15	1 11/16	5/8"	15	2 11/16	1/4"
16	1 11/16	7/16"	16	[illegible]	15/32"

INSPECTION AFTER [illegible]
10 ON 2 OF 4[illegible]
REJECT TICKET # [illegible]

S/N A5-144-1
LOWER HEAD SUB-ASSY.

CUSTOMER WESTINGHOUSE

JOB. NO. 610-0144-51-1

SUBJECT INSPECT COLD FIT-UP OF LOWER HEAD RING (6-144-1) TO LOWER HEAD DISK (5-152-1)
SKETCH # 2

DWG. NO. 133[illegible] MK# A5

BY GARY [illegible]

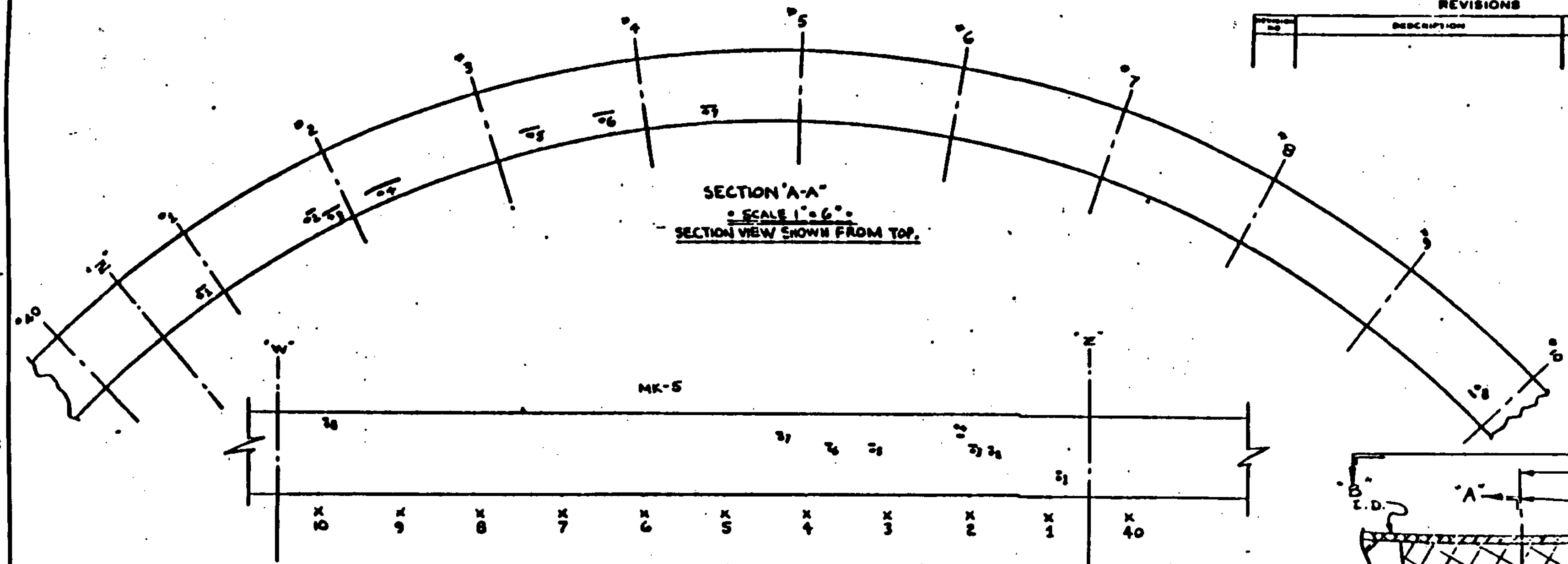
DATE 7-29-6[illegible]

COLLATION OF ULTRASONIC

TEST RESULTS WITH FABRICATION

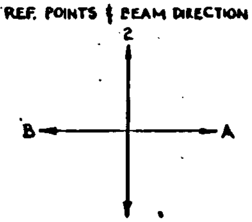
HISTORY DRAWING 12071 SHEETS

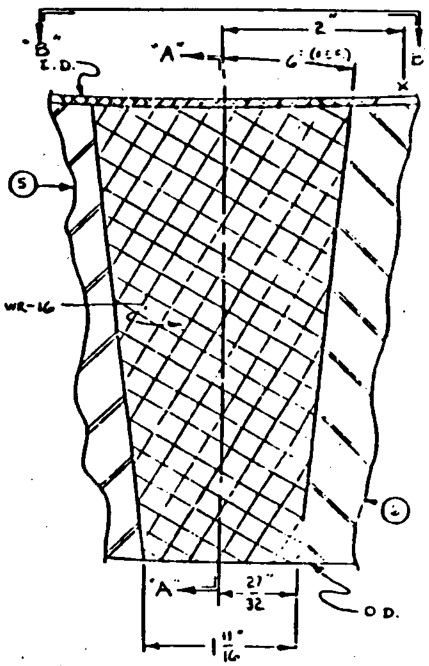
1 THRU 4 INCLUSIVE



SECTION "B-B"
• THIS VIEW NOT TO SCALE •
LOOKING IN AT INSIDE DIA.

REF. POINTS & BEAM DIRECTION





• SCALE : FULL •

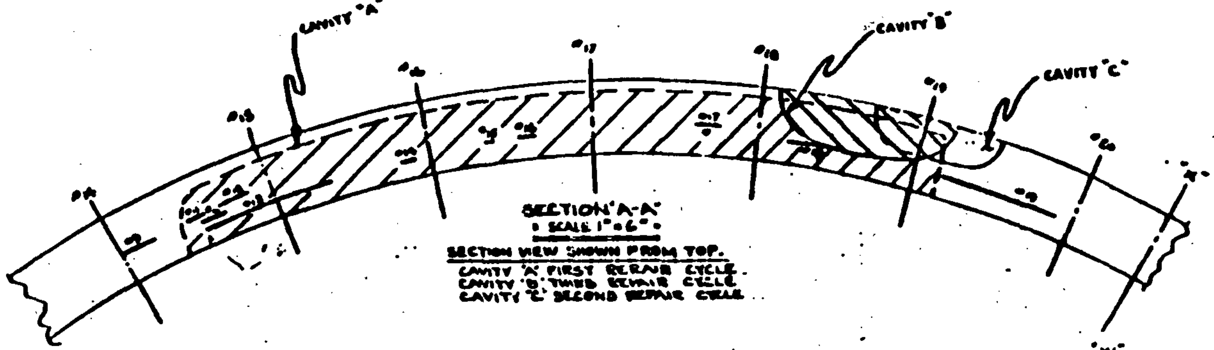
POSITION A	POSITION B	LENGTH	DEPTH	BACK REF.	MAX. SIG.	BEAM DIRECTION	INDICATED DEFECT TYPE	ANGLE	DISTANCE FROM A	DISTANCE FROM B	DISTANCE FROM 1	DISTANCE FROM 2	PROBE SURFACE	STATUS	MAX. SIG.	ANGLE	
10	1	3/4	4		40	1		45°		1/8	5 1/2		5		0	60°	•1-OK G.W.
1	2	3/4	3 1/2		50	1		45°		2 1/4	5 3/4		5		0	60	•2-OK
		1 7/8	4		40	1		45°		1/2	6 1/4		5		0	60	•3-OK
2	3	2 3/4	4		60	1		45°	1 1/3		6 7/8		5		0	60	•4-OK
3	4	1 7/8	3 3/4		50	1		45°	2 1/3		6 1/2		5		0	60	•5-OK
		1 1/2	3 3/4		50	1		45°		2 3/4	6 1/4		5		0	60	•6-OK
4	5	1	4		50	1		45°	4		6 1/2		5		0	60	•7-OK
6	7													OK			
7	8													OK			
8	9													OK			
9	10	5/8	4		50	1		45°		1 1/8	7		5		60	60°	•8-OK G.W.
10	11													OK			

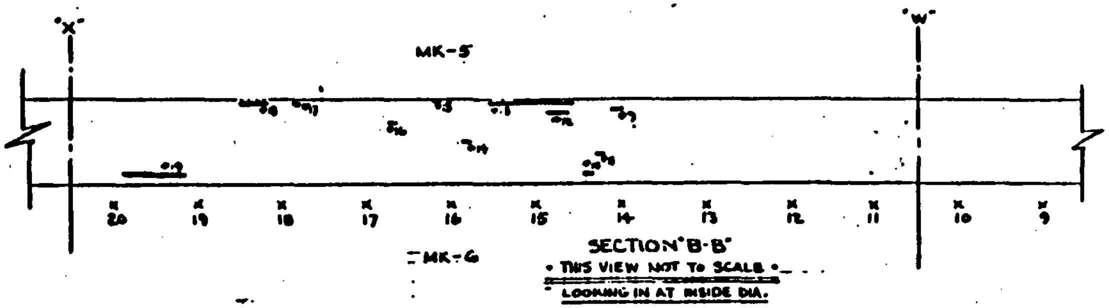
NOTES :-
(1) INSIDE DIA. 144 3/8"
(2) THICKNESS 5 1/2"
(3) R.T. LAYOUT - O.D. 12" ± 3/8" = 9°
(4) R.T. LAYOUT - I.D. 11 7/8" ± 3/8" = 9°

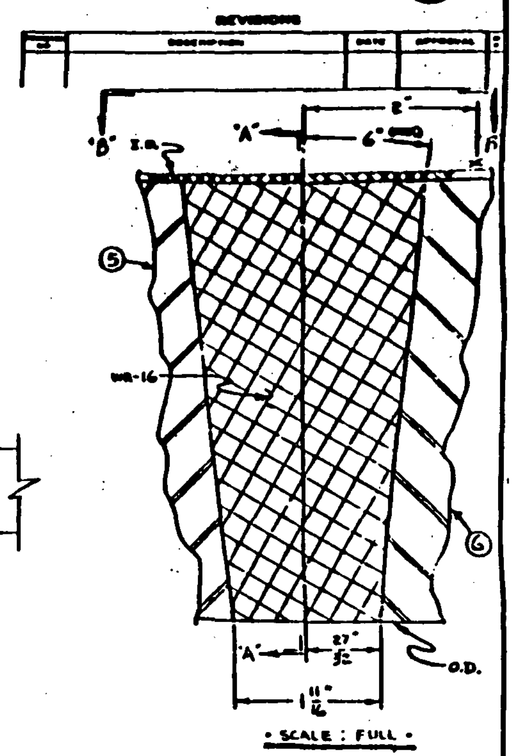
DRN BY G. REMO	ULTRASONIC TEST DATA	THE BABCOCK & WILCOX CO
CHKD BY	610-144-51	
APPROVED	MK 5 - MK 6 CIRCLE SEAM	
SCALE	SHEET 1 OF 4	MTV 12071 C — C

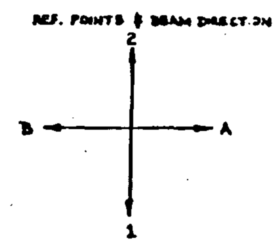
POSITION A	POSITION B	LENGTH	DEPTH	BKG. REF.	MAX. SIG.	BEAM DIRECTION	INDICATES DEFECT TYPE	ANGLE	DISTANCE FROM A	DISTANCE FROM B	DISTANCE FROM 1	DISTANCE FROM 2	INSIDE SURFACE	STATUS	MAX. SIG.	ANGLE	
10	11													OK			
11	12													OK			
12	13													OK			
13	14													OK			
14	15	2¼	3		200	1		45°	0		6½		5		180	60°	#9
		2	3½		400	1		45°	6		3½		5		450	60°	#10
		1½	3		160	1		45°	5		7		5		200	60°	#11
		2½	3		200	1		45°		1½	6½		5		400	60°	#12
15	16	9½	4		150	1		45°		7½	3		5		500	60°	#13
		1	3½		50	1		45°		1	6½		5		[illegible]	60°	#14
16	17	½	3½		90	1		45°	4		6½		5		175	60°	#15
		1	3½		50	1		45°		5	7		5		200	60°	#16
17	18	1½	3		300	1		45°		2½	7		5		120	60°	#17
18	19	3¼	3½		300	1		45°	2		7½		5		300	60°	#18
19	20	9½	3½		200	1		45°	4		5		5		250	60°	#19

N/A









NOTES :-
(1) INSIDE DIA. 144 ⅞"
(2) THICKNESS 5½"
(3) R.T. LAYOUT - O.D. 12" ± ⅜" = 9°
(4) R.T. LAYOUT - I.D. 11⅞" = ⅛" = 9°
(5) LEGEND : — — — DENOTES CAVITIES PROBED & REPAIRED FROM I.D.
——— DENOTES CAVITIES PROBED & REPAIRED FROM O.D.

CAVITY	LENGTH	WIDTH	DEPTH
"A"	53¼"	4½"	* 4 7/16
"B"	11 ⅞"	3"	4 9/32
"C"	9 ¼"	3"	2 ¾"

* DEPTH DOES NOT INCLUDE CLAD THICKNESS

ULTRASONIC TEST DATA
610-144-51
MK 5 - MK6 CIRCLE SEAM

THE BABCOCK & WILCOX CO

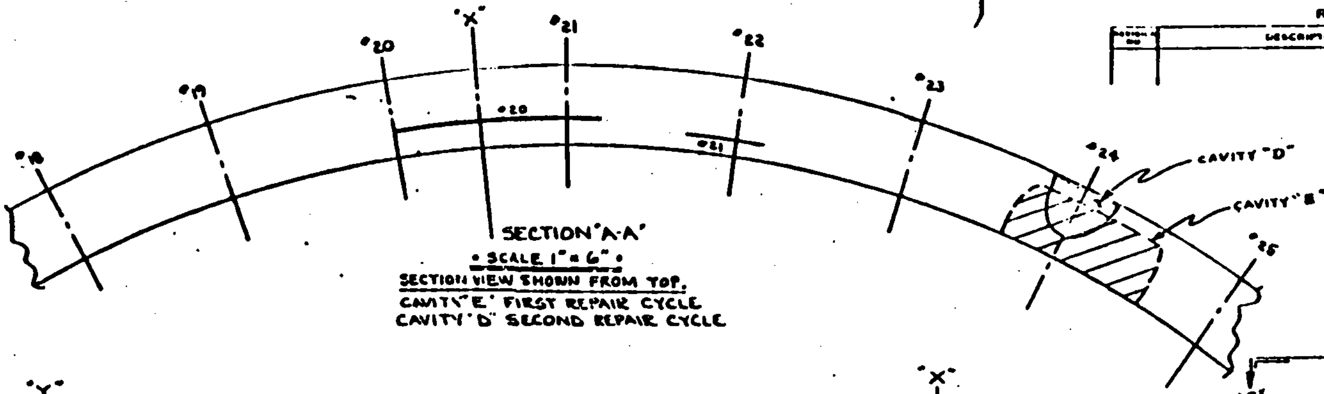
MTV 12071 C

5C-90

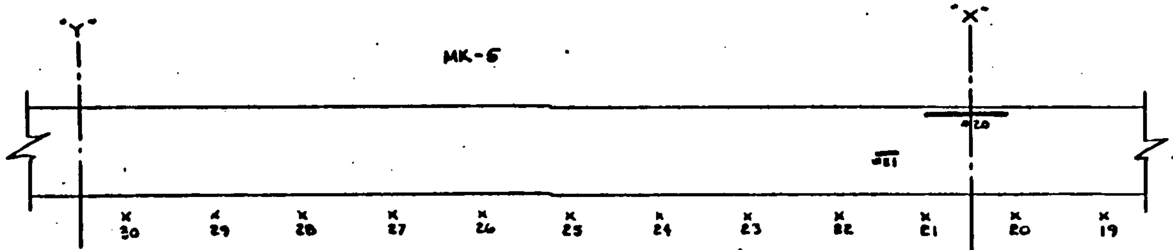
JUNE 1992

REVISIONS

SYM	DESCRIPTION	DATE	APPROVAL

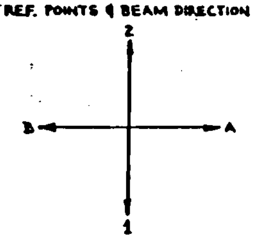


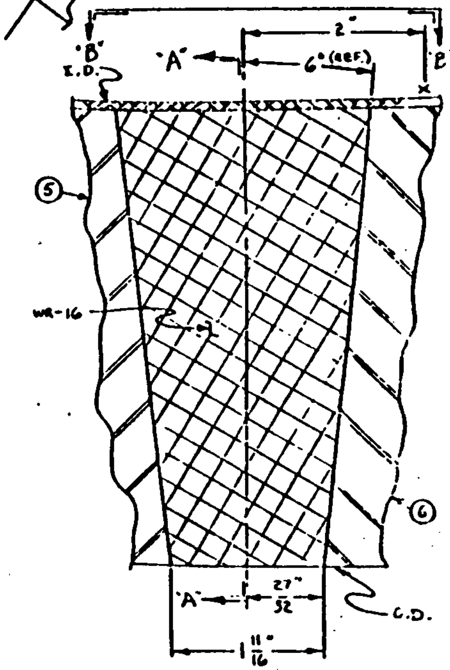
SECTION "A-A"
• SCALE 1" = 6" •
SECTION VIEW SHOWN FROM TOP.
CAVITY "E" FIRST REPAIR CYCLE
CAVITY "D" SECOND REPAIR CYCLE



SECTION "B-B"
• THIS VIEW NOT TO SCALE •
LOOKING IN AT INSIDE DIA.

REF. POINTS & BEAM DIRECTION





• SCALE : FULL •

POSITION A	POSITION B	LENGTH	DEPTH	SCAN REF.	MAX. SIG.	BEAM DIRECTION	INDICATED DEFECT TYPE	ANGLE	DISTANCE FROM A	DISTANCE FROM B	DISTANCE FROM 1	DISTANCE FROM 2	PROBE SURFACE	STATUS	MAX. SIG.	ANGLE	
[illegible]	21	13 3/4	3 1/2		[illegible]	1		45°	0		5 1/2		5		250	60°	#20 - N/A
[illegible]	[illegible]	5	4		40	1		45°	3		6		5		10	60°	#21 - OK G.W.
22	23																
[illegible]	24																
24	25																
[illegible]	[illegible]																
[illegible]	27																

NOTES :-
(1) INSIDE DIA. 144 3/8"
(2) THICKNESS 5 1/2"
(3) R.T. LAYOUT - O.D. 12" ± 3/8" = 9°
(4) R.T. LAYOUT - I.D. 11 1/8" ± 1/4" = 9°
(5) LEGEND : DENOTES CAVITIES PROBED & REPAIRED FROM I.D.
DENOTES CAVITIES PROBED & REPAIRED FROM O.D.

CAVITY	LENGTH	WIDTH	DEPTH
"D"	4 1/16"	3 3/16"	4 1/16"
"E"	9 1/2"	5 1/2"	* 4 5/16"

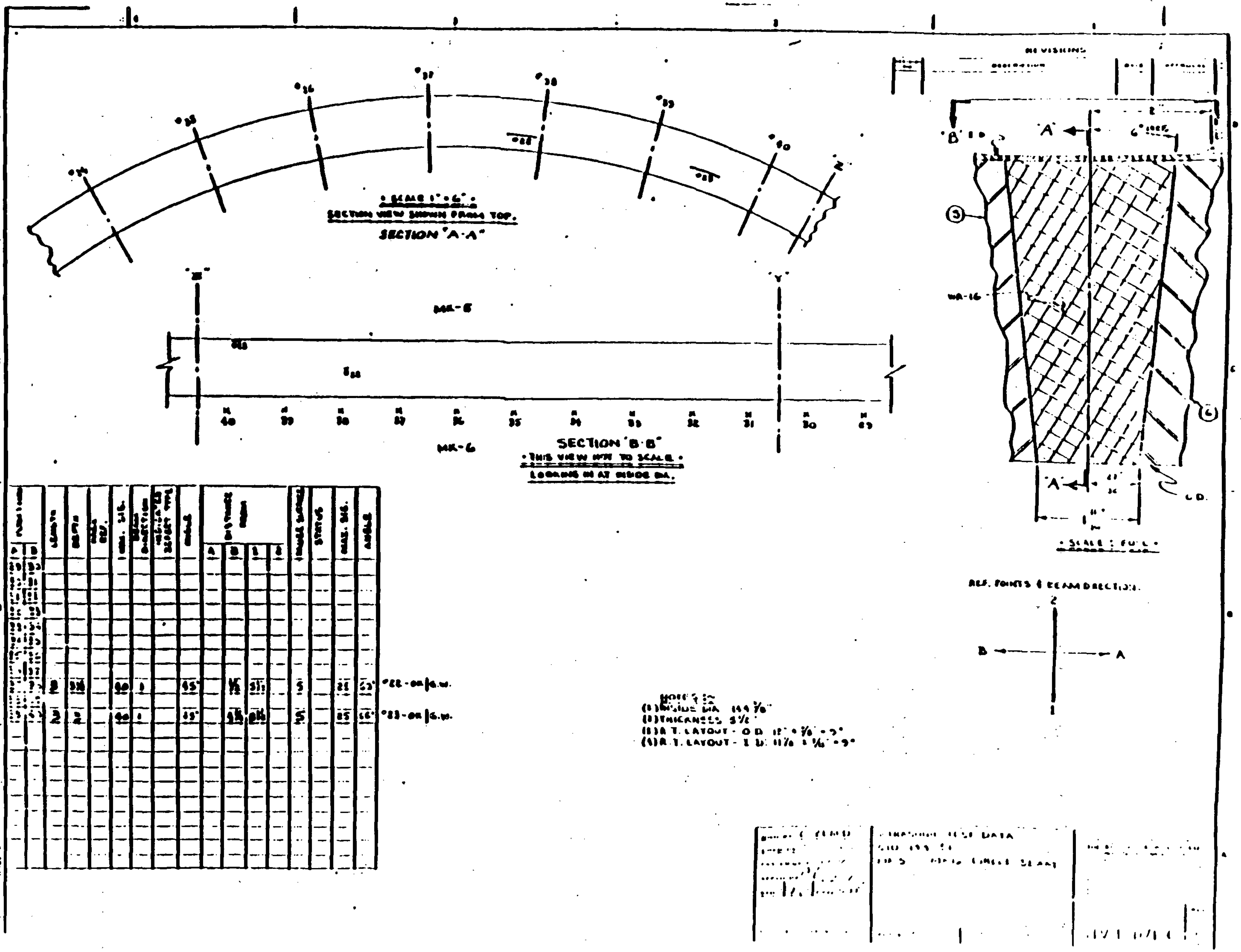
* DEPTH DOES NOT IN[illegible] DE CLAD THICKNESS (3/16")

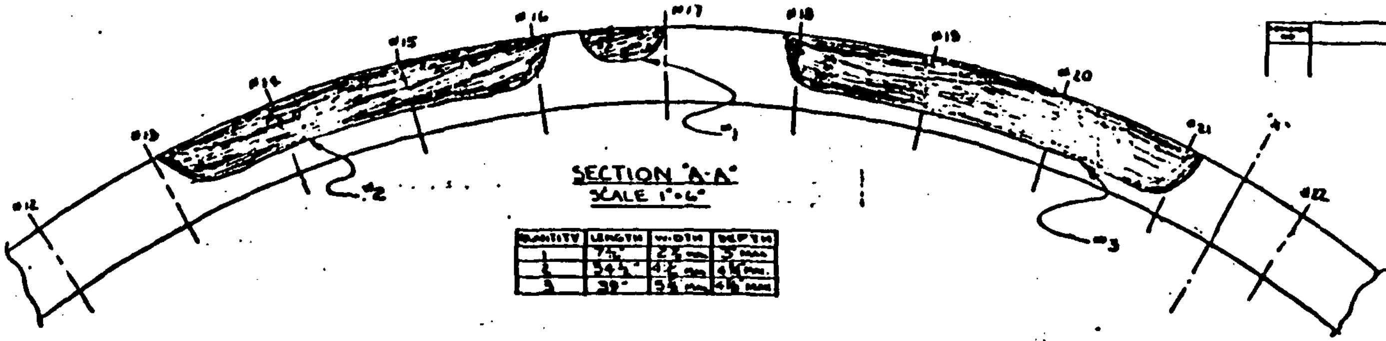
DWN BY G. LEARD
CHKD BY [illegible]
REVISED BY [illegible]
APPVD BY [illegible]

ULTRASONIC TEST DATA
610-144-51
MK 5 - MK6 CIRCLE SEAM

THE BABCOCK & WILCOX CO

[illegible] 12071 C | 0


SECTION "A-A"
SECTION VIEW SHOWN FROM TOP.
MK-5
MK-6
SECTION "B-B"
THIS VIEW NOT TO SCALE.
LOOKING IN AT INSIDE DIA.
"A"
"B"
SCALE : FULL
REF. POINTS & BEAM DIRECTION.
B
A
1
2
NOTES:
O.D.



SECTION "A-A"
SCALE 1"=6"

QUANTITY	LENGTH	WIDTH	DEPTH
1	75"	22 mm	3 mm
2	54½"	42 mm	4½ mm.
3	39"	53 mm	4½ mm

REVISIONS

NO	DESCRIPTION	DATE	APPROVAL

DRWN BY D G HEAD
CHKD BY SWC
APPD BY [illegible]
DATE 2-[illegible]-91

CHART CAVITIES
PRIOR TO REPAIR
610-0144-51
MK 5 - MK 6 CIRCLE SEAM

THE BABCOCK & WILCOX CO

SCALE 1"=6"
SHT 5 of 5
MTV 1 C
REV 0

THE BABCOCK & WILCOX COMPANY
POWER GENERATION DIVISION

REVISIONS				MICRO-FILM
DASH NO	DATE	DESCRIPTION	ORIG	

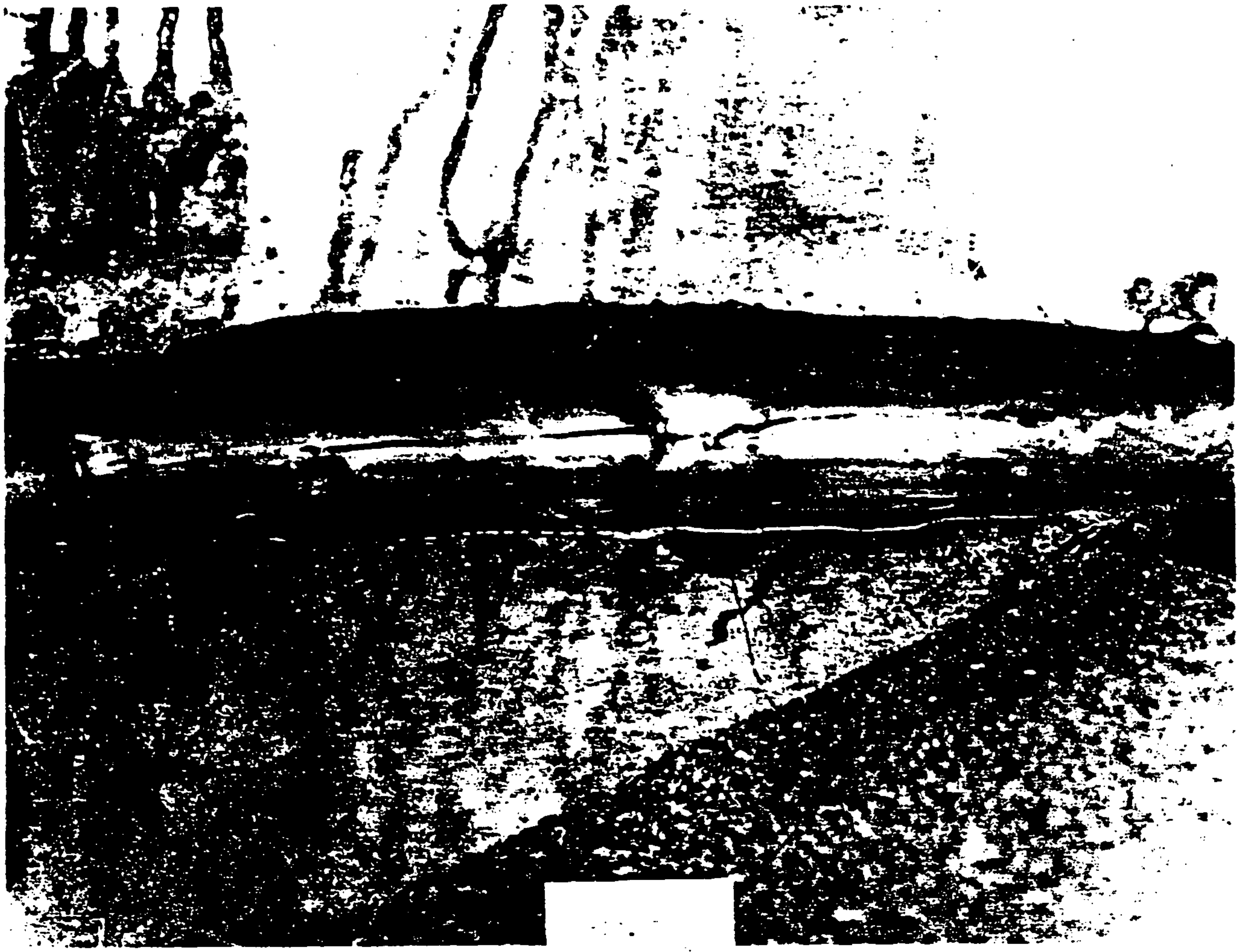
TABLE *1
CORRELATION OF ACTUAL DEFECT DIMENSIONS WITH ULTRASONIC REPORT.

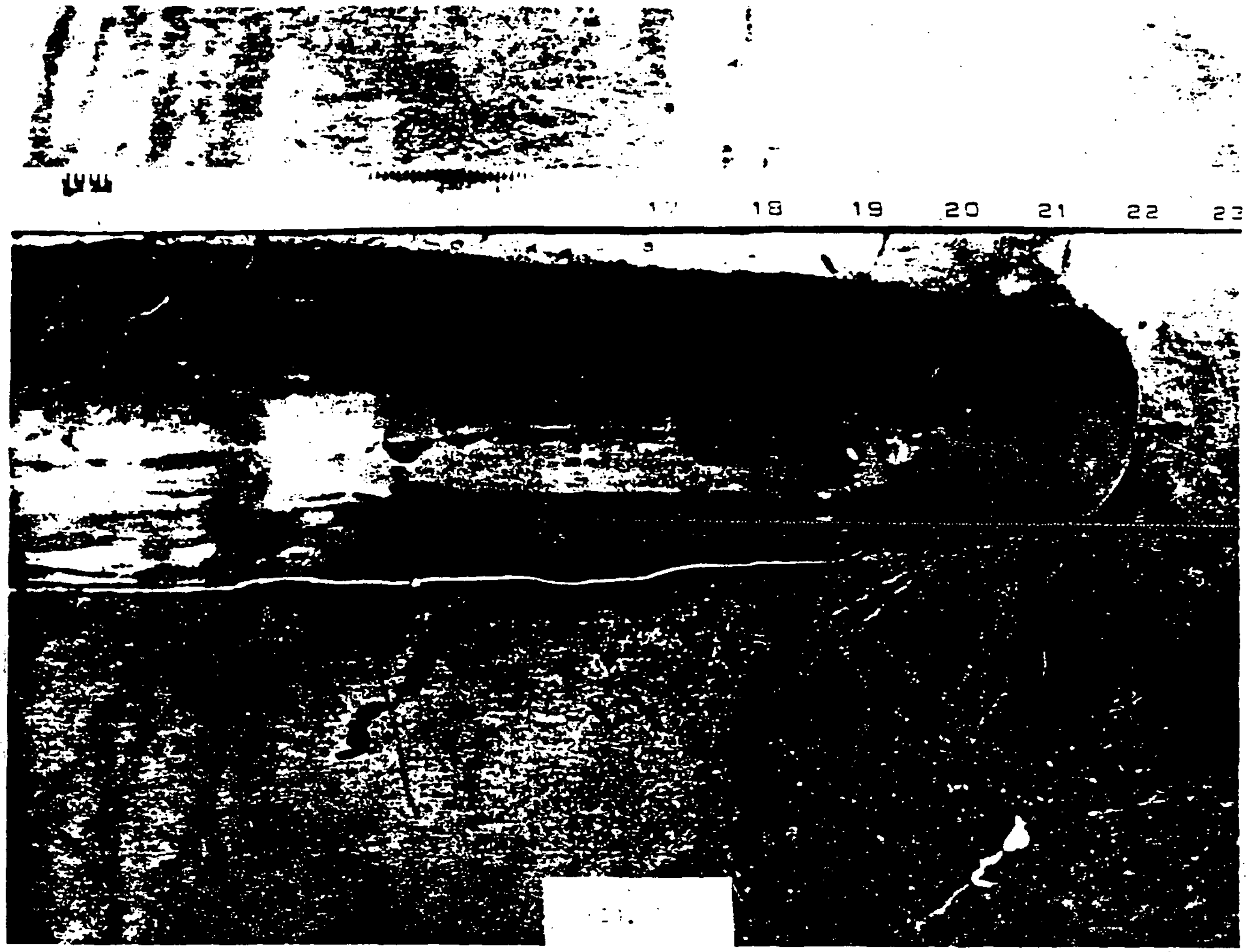
BETWEEN X-RAY POINTS	DEFECT No.	ACTUAL DEPTH O.D.	U.T. PREDICTED LENGTH I.D.	UT PREDICTED DEPTH I.D.
40 - 1	1	NOT PROBED	1/4	4
1 - 2	2		3/4	3½"
1 - 2	3		1 1/2	4
2 - 3	4		2 1/4	4
3 - 4	5		1 1/2	3 3/4
3 - 4	6		1 1/2"	3 3/4
4 - 5	7		1	4
9 - 10	8		5/8"	4
14-15	9	2 7/8" TO 3 1/2" COMBINED	2 1/2"	3"
	10		2"	3 1/2
	11		1 1/2"	3"
	12		2 1/2"	3"
	*13		9 1/2"	4"
15 - 16	14	2 1/8 TO 2 5/8	1	3 3/4
16 - 17	15	2 1/2" TO 3"	1/2"	3 1/2
	16		1"	3 1/2
17 - 18	17	2 7/8 TO 3 1/2	1 1/4"	3"
18 - 19	18	2 7/8" TO 3 1/8" COMBINED	3/4"	3 1/2
19 - 20	19		9 1/2	3 1/2
20 - 21	20		13 3/4	3 1/2
21 - 22	21	NOT PROBED	5	4
[illegible]	22		3"	3 1/2
39 - 40	23		3	3

* DEFECT #13 IS DIVIDED BY X-RAY POINT #15 5/4" BETWEEN X-RAY POINTS #14 & 15 1 4¾" [illegible]

DWN BY [illegible]	CHKD		SCALE	DATE
PASSED BY	APPD		DRWG. NO	A-

FORM 60M 1087 A 2




18
19
20
21
22

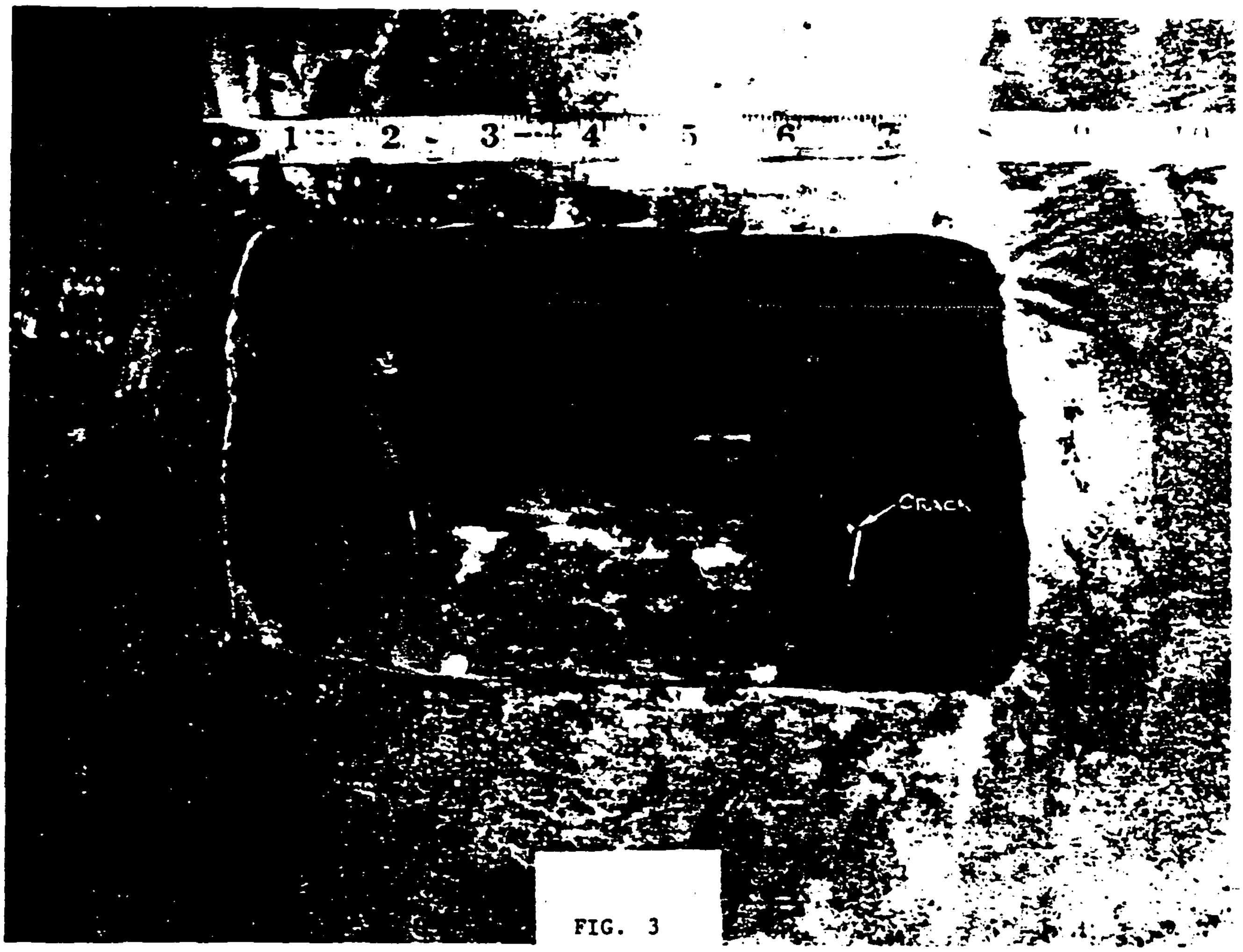
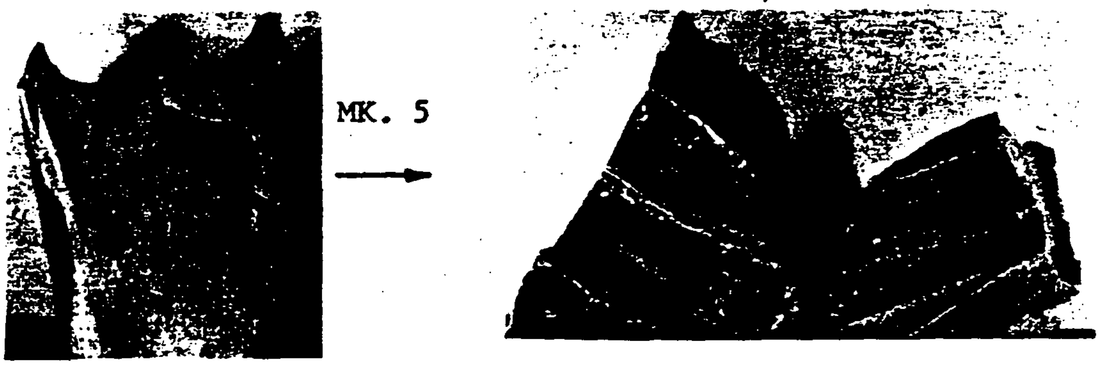


FIG. 3



4(a) 4(b)

Fig. 4 · Two views of a sample removed from radiographic station 14-15. X2½

MK. 5 MK. 5

5(a) 5(b)

Fig. 5 Two additional samples removed from radiographic station 14-15. Note the heat affected zone caused by the arc-air electrode during sample removal. X2

Fig. [illegible] Sample removed from radiographic station 15-16. Note non-fusion. X2½

Fig. 7 (X11) Showing non-fusion at boundary of manual metal arc deposit and submerged arc weld. Radiographic station 15-16.

Fig. 8a (X300) Typical structure of submerged arc weld metal. Radiographic station 15-16.

Fig. 8b (X300) Structure of submerged arc weld adjacent to non-fusion area. Radiographic station 15-16.