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ACCESSION NBR: 8704290121 DOC. DATE: 87/04/24 NOTARIZED: NO DOCKET #
FACIL: 50-361 San Onofre Nuclear Station, Unit 2, Southern Californ 05000361
50-362 San Onofre Nuclear Station, Unit 3, Southern Californ 05000362
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SUBJECT: Forwards response to request for addl info to facilitate NRC review of CEN-259, "An Evaluation of Natural Circulation of Cooldown Test Performed at San Onofre Nuclear Generating Station."

DISTRIBUTION CODE: A052D COPIES RECEIVED: LTR 1 ENCL 1 SIZE: 28
TITLE: OR Submittal: Natural Circ Cooldown GL 81-21 Multiplant Action B-66

NOTES: ELD Chandler 1cy. 05000361
ELD Chandler 1cy. 05000362

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April 24, 1987

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

Gentlemen:

Subject: Docket Nos. 50-361 and 50-362
San Onofre Nuclear Generating Station
Units 2 and 3

By letter dated March 6, 1987, the NRC requested additional information to facilitate NRC review of CEN-259, "An Evaluation of the Natural Circulation Cooldown Test Performed at the San Onofre Nuclear Generating Station." The purpose of this letter is to transmit the requested information to the NRC which is provided as an enclosure to this letter.

If you have any questions, please contact me.

Very truly yours,

M. O. Medford

Enclosure

cc: H. Rood, NRR Senior Project Manager, San Onofre Units 2 and 3
J. B. Martin, Regional Administrator, NRC Region V
F. R. Huey, NRC Senior Resident Inspector, San Onofre Units 1, 2 and 3

A052
1/1

ENCLOSURE

RESPONSE TO NRC REQUEST FOR ADDITIONAL
NATURAL CIRCULATION INFORMATION

1. Provide the steam generator secondary side volumes for the following:

   1) Total
   2) Downcomer
   3) Tube region
   4) Steam separator and dryer
   5) Steam dome

   Response:

   1) Total steam generator secondary side volume - 10,042 ft^3
   2) Downcomer - 1538.5 ft^3

   The remaining three volumes have not been calculated and are not readily available. However, Figure 1 of this enclosure provides all steam generator dimensions from which the desired volumes can be derived.

2. Provide the total water volume (or level) of the steam generator secondary side during the full power operation and the hot shutdown conditions.

   Response:

   Water level at 100% full power - 438.6 inches above the tube sheet. Water level at hot shutdown is the same as full power.

3. FSAR Table 4.4-4 shows delta pressure for the vessel. Similar information regarding delta pressure in other components of the RCS outside the vessel is needed (such as delta pressure from the vessel upper plenum to the vessel outlet nozzle to the steam generator inlet nozzle, from the steam generator inlet nozzle to the outlet nozzle, from the steam generator outlet nozzle to the pump, and from the pump to the vessel inlet nozzle).

   Response:

   See Figure 2 of this enclosure.

4. Provide the bypass flow to and from the upper head (clarification of FSAR Table 4.4-3 and breakdown of the flow <u>6</u> in Figure 4.4-6).

   Response:

The flow into the upper dome region of the San Onofre Units 2 and 3 reactor vessels with reactor coolant pumps operating originates as follows:

	Origin	Percent of Total RV Flow
a.	Alignment Keys	0.1
b.	Rodded Guide Tubes via CEA Shrouds	0.3
c.	Additional CES Shroud Flow	0.8
	Total Inflow	1.2%

The total outflow exits the upper head through the upper guide structure support plate.

5. Provide the total nitrogen gas supply (i.e., number of bottles or total volume) available on site for the atmospheric steam dump valves.

   Response:

   Nitrogen gas supply - 13.63 ft^3 at 1200 psig

6. Provide the boron concentration of the boron injection flow during the natural circulation test of July 27, 1983.

   Response:

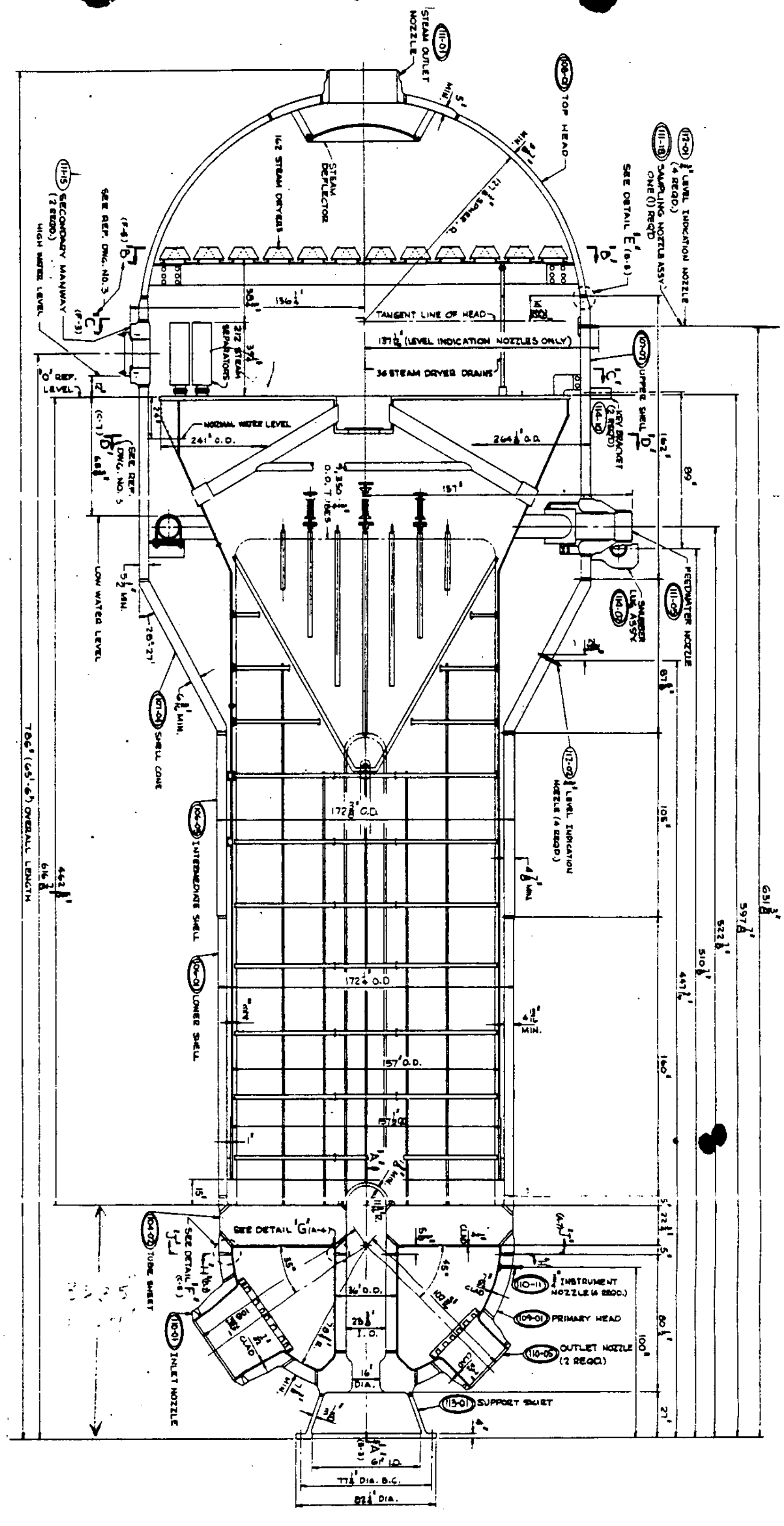
   Boron concentration in the tank at the time of the test was 9 1/2 wt %.

7. Provide detailed geometric information on the upper head including the thickness, surface area and material of the upper head hemisphere, number of the CEDM drive trains, material and structure of CEDM drives (length, penetration into the upper head, cross-sectional geometry), number of guide tubes, material and structure of the guide tubes length, penetration into the upper head and cross-sectional geometry), and structure and material of the upper core support plate.

   Response:

   Provided as an attachment to this enclosure are the applicable sections of the FSAR which provide information regarding reactor vessel materials, the number of CEA's and guide tubes and details regarding the upper core support plate. A reactor vessel arrangement diagram (Figure 3) has been provided to show all reactor vessel dimensions. Figure 4 details the number of reactor vessel head penetrations and their locations. A diagram of a control element drive mechanism has also been included (Figure 5).

Figure I

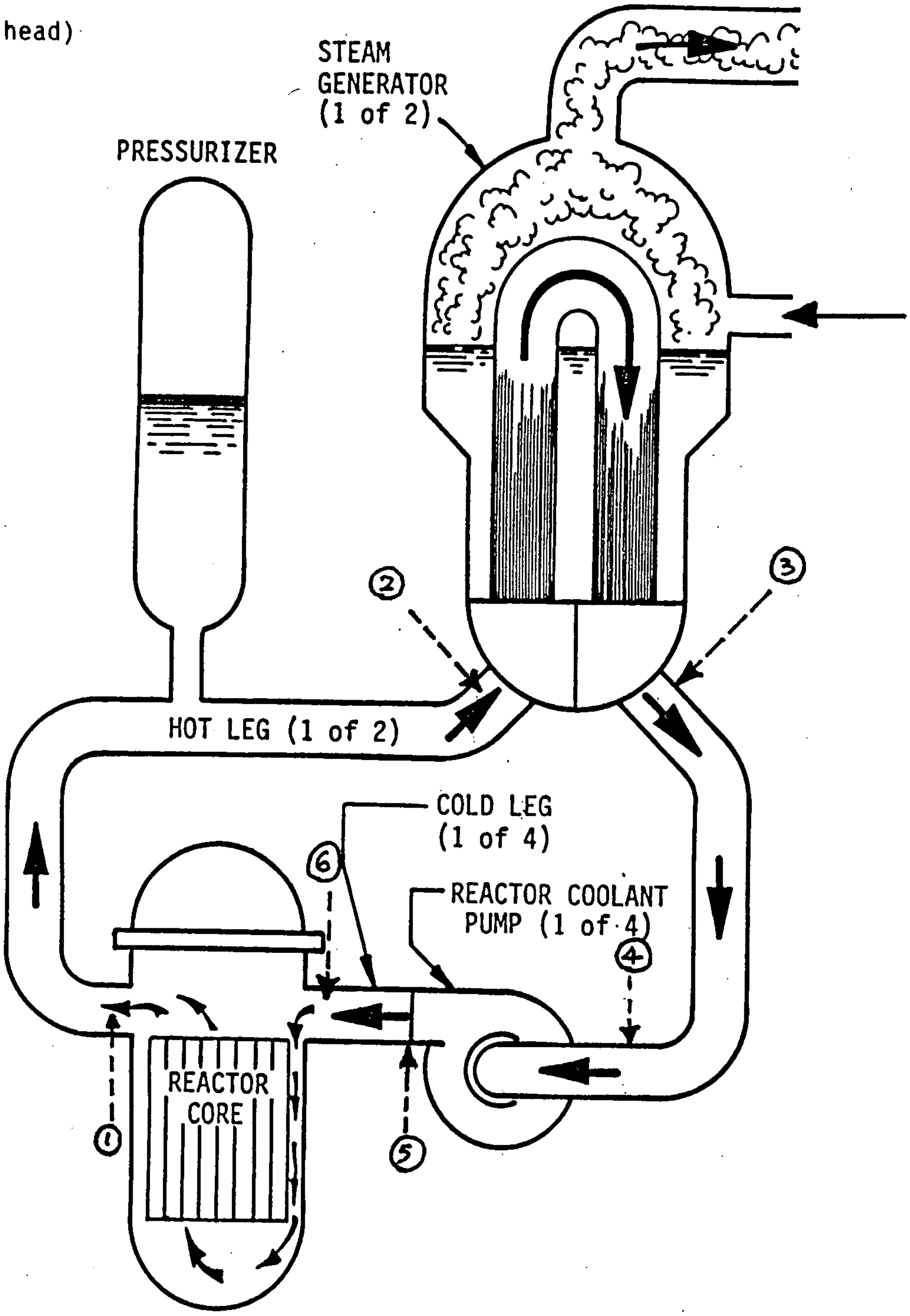


STEAM OUTLET NOZZLE
TOP HEAD
STEAM DEFLECTOR
162 STEAM DRYERS
TANGENT LINE OF HEAD
36 STEAM DRYER DRAINS
NORMAL WATER LEVEL
LOW WATER LEVEL
HIGH WATER LEVEL
SECONDARY MANWAY (2 REQD.)
UPPER SHELL
FEEDWATER NOZZLE
SNUBBER LUG ASSY.
SHELL CONE
INTERMEDIATE SHELL
LOWER SHELL
TUBE SHEET
INLET NOZZLE
INSTRUMENT NOZZLE (4 REQD.)
PRIMARY HEAD
OUTLET NOZZLE (2 REQD.)
SUPPORT SKIRT

Figure 2

Reactor Coolant Loop Pressure Drops

ΔP_{1-2} = 0.5 psi
ΔP_{2-3} = 47.0
ΔP_{3-4} = 5.0
ΔP_{4-5} = 96.7 (pump head)
ΔP_{5-6} = 0.8
ΔP_{6-1} = 43.4



3.9.5.1.2 Upper Guide Structure Assembly

This assembly consists of the upper guide structure (UGS) support plate assembly, control element assembly shrouds, and a fuel assembly alignment plate (figure 3.9-35). The upper guide structure assembly aligns and laterally supports the upper end of the fuel assemblies, maintains the CEA spacing, holds down the fuel assemblies during operation, prevents fuel assemblies from being lifted out of position during a severe accident condition, protects the control element assemblies (CEAs) from the effect of coolant cross flow in the upper plenum, and supports the in-core instrumentation plate assembly. The upper guide structure assembly is handled as one unit during installation and refueling.

The upper end of the assembly is a structure consisting of a support flange welded to the top of a cylinder. A support plate is welded to the inside of the cylinder approximately in the middle. The support plate is welded to a grid array of deep beams, the ends of which are welded to the cylinder. The support flange contains four accurately machined and located alignment keyways, equally spaced at 90-degree intervals, which engage the core barrel alignment keys. This system of keys and slots provides an accurate means of aligning the core with the closure head and thereby with the CEA drive mechanisms. The support plate aligns and supports the upper end of the four and five-element CEA shrouds. The shrouds extend from the fuel assembly alignment plate to an elevation above the upper guide structure support plate. The five-element CEA shrouds consist of a cylindrical upper section welded to a base, and a flow channel structure shaped to provide flow passage for the coolant through the alignment plate, while shrouding the CEAs from cross flow. The shrouds are bolted and lockwelded to the fuel assembly alignment plate. At the upper guide structure support plate. the shrouds are connected to the plate by spanner nuts. The spanner nuts are tightened with proper torque to assure a rigid connection and lock-welded. Four-element CEA shrouds are located at the periphery of the UGS on each of the four major axes. These shrouds consist of a cylindrical upper section welded to a base; the base is bolted and lockwelded to the fuel assembly alignment plate, and the upper section of the shroud is structurally welded to the UGS support plate. Inside the four-element CEA shrouds, a flow bypass insert assembly provides a flow path through the fuel assembly alignment plate to the outlet plenum, reducing axial flow within these shrouds. A torqued and lockwelded spanner nut fastens the flow bypass insert assembly to the four-element CEA shroud base.

The fuel assembly alignment plate is designed to align the upper ends of the fuel assemblies and to support and align the lower ends of the CEA shrouds. Precision machined and located holes in the fuel assembly alignment plate engage machined posts onthe fuel assemby upper end fittings to provide accurate alignment. The fuel assembly alignment plate also has four equally spaced slots on its outer edge that engage with Stellite hardfaced pins protruding from the core shroud to limit lateral motion of the upper guide structure assembly during operation. The fuel alignment plate bears the upward force of the fuel assembly holddown devices. This force is transmitted from the alignment plate through the CEA shrouds to the upper guide structure support plate. The flange of the upper guide

structure support plate is designed to resist axial upward movement of the upper guide structure assembly and to accommodate axial differential thermal expansion between the core barrel flange, upper guide structure and pressure vessel flange support ledge, and head flange recess.

3.9.5.1.3 Flow Skirt

The Inconel flow skirt is right circular cylinder, perforated with flow holes, and reinforced at the top and bottom with stiffening rings. The flow skirt is used to reduce inequalities in core inlet flow distributions and to prevent formation of large vortices in the lower plenum. The skirt provides a nearly equalized pressure distribution across the bottom of the core support barrel. The skirt is supported by nine equally spaced machined sections that are welded to the bottom head of the pressure vessel.

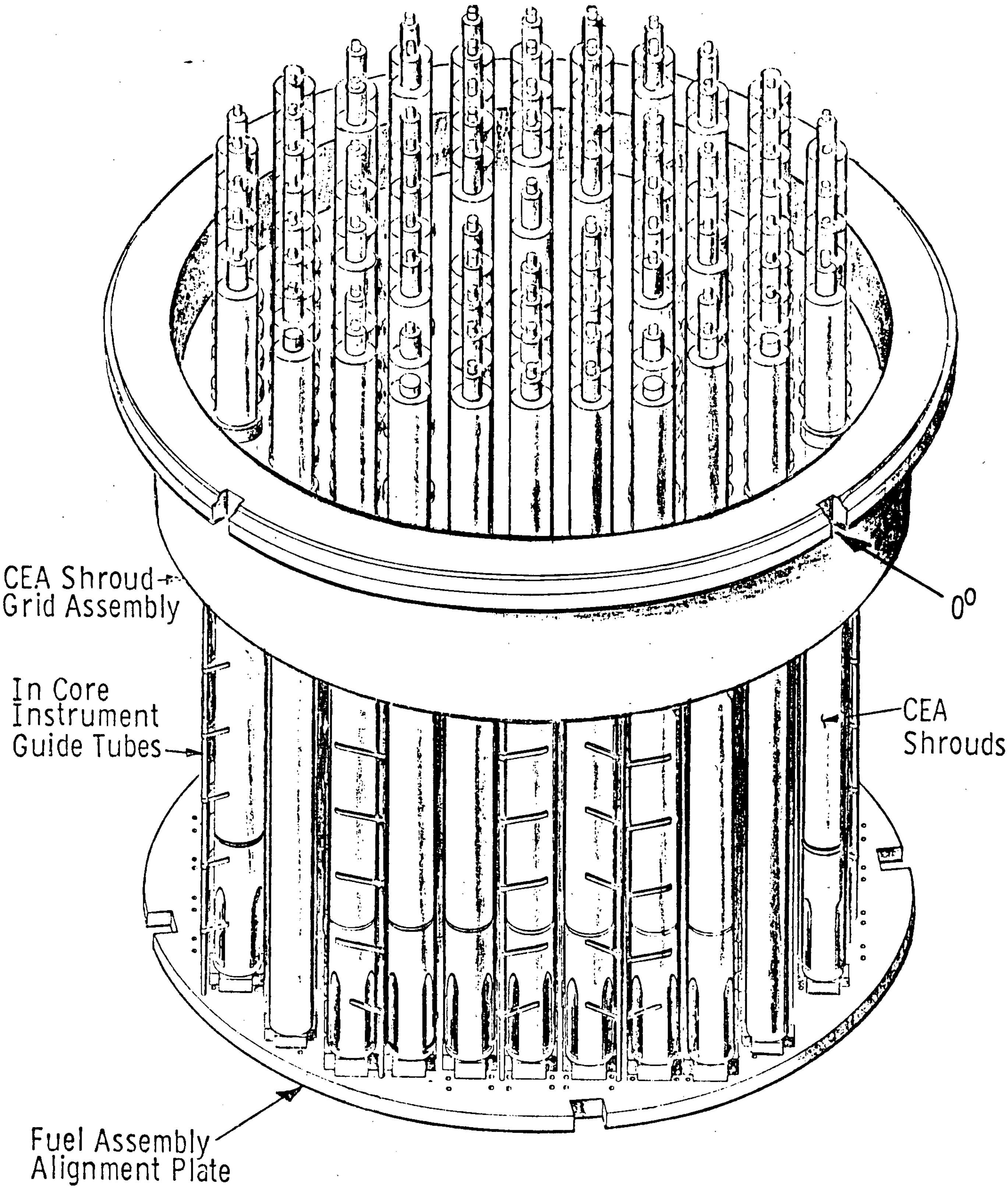
3.9.5.1.4 In-Core Instrumentation Support System

The complete in-core neutron flux monitoring system includes self-powered in-core detector assemblies, supporting structures and guide paths, an external movable detector drive system, and an amplifier system to process detector signals.

The self-powered in-core detector assemblies and the computer system are described in section 7.7. The external movable detector drive system and instrumentation supporting structures and guide paths are described in this section and are shown in figures 3.9-36 through 3.9-39. Figure 3.9-38 shows the core locations of the in-core detector assemblies.

The support system begins outside the pressure vessel, penetrates the vessel boundary, and terminates at the lower end of the fuel assembly. Each instrument is guided over its full length by the external guidance conduit, the instrument plate structure guide tubes, and the thimbles that extend downward into the guide tubes of selected fuel bundles. The in-core instrumentation guide tubes route the instruments so that the detectors are located and spaced throughout the core. The guide tubes and the in-core thimbles are attached to and supported by the instrument plate assembly shown in figure 3.9-36.

The instrumentation plate assembly fits within the confines of the reactor vessel head and rests in the recessed section of the upper guide structure assembly. Its weight is supported by four bearing pins. The upper guide structure CEA shrouds extend through the instrumentation plate clearance
3 holes. Above the instrumentation plate, the guide tubes bend and are gathered to form stalks that extend into the reactor vessel head instrumentation nozzles. The instrumentation plate assembly is raised and lowered during refueling to insert or withdraw all instruments and their thimbles simultaneously. The pressure boundaries for the individual instruments are at the instrumentation nozzle flange, where the external electrical connections to the in-core instruments are also made.



Updated

SAN ONOFRE NUCLEAR GENERATING STATION Units 2 & 3
UPPER GUIDE STRUCTURE ASSEMBLY
Figure 3.9-35

4. REACTOR

4.1 SUMMARY DESCRIPTION

The reactor is of the pressurized water (PWR) type using two reactor coolant loops. A vertical cross-section of the reactor is shown in figure 4.1-1. The reactor core is composed of 217 fuel assemblies and 91 control element assemblies (CEAs). The fuel assemblies are arranged to approximate a right circular cylinder with an equivalent diameter of 136 inches and an active length of 150 inches. The fuel assembly, which provides for 236 fuel rod positions, consists of 5 guide tubes welded to spacer grids and is closed at the top and bottom by end fittings. The guide tubes each displace four fuel rod positions and provide channels that guide the CEAs over their entire length of travel. In selected fuel assemblies, the central guide tube houses in-core instrumentation.

The fuel is low enriched UO_2 in the form of ceramic pellets and is encapsulated in pre-pressurized Zircaloy tubes that form a hermetic enclosure.

The reactor coolant enters the upper section of the reactor vessel, flows downward between the reactor vessel wall and the core barrel, passes through the flow skirt where the flow distribution is equalized, and into the lower plenum. The coolant then flows upward through the core, removing heat from the fuel rods, exits from the reactor vessel and passes through the tube side of the vertical U-tube steam generators where heat is transferred to the secondary system. The reactor coolant pumps return the coolant to the reactor vessel.

Figure 4.1-2 shows the reactor core cross-section and certain dimensional relations between fuel assemblies, fuel rods, and CEA guide tubes.

The reactor internals support and orient the fuel assemblies, control element assemblies, in-core instrumentation, and guide the reactor coolant through the reactor vessel. They also absorb static and dynamic loads and transmit the loads to the reactor vessel flange. They will safely perform their functions during normal operating, upset, emergency, and faulted conditions. The internals are designed to safely withstand forces due to dead weight, handling, temperature and pressure differentials, flow impingement, vibration, and seismic acceleration. All reactor components are considered Category I for seismic design. The design of the reactor internals limits deflection where required by function. The stress values of all structural members under normal operating and expected transient conditions are not greater than those established by Section III, Subsection NG, of the ASME Pressure Vessel Code. The effect of neutron irradiation on the materials concerned is included in the design evaluation. The effect of accident loadings on the internals is included in the design analysis.

Reactivity control is provided by two independent systems - the control element drive system and the chemical and volume control system (CVCS).

The control element drive system controls short-term reactivity changes and is used for rapid shutdown. The CVCS compensates for long-term reactivity changes and can make the reactor subcritical without the benefit of the control element drive system. Design of the core and the reactor protective system prevents fuel damage limits from being exceeded for any single malfunction in either of the reactivity control systems.

The standard control element assemblies consist of five poison rods assembled in a square array, with one rod in the center. The rods are connected to a spider structure that couples to the control element drive mechanism (CEDM) shafting. There are 91 CEAs, of which 83 are full length and 8 contain only a part-length poison column. Of the 83 full-length CEAs, 79 are the standard five-element design and 4 are four-element CEAs (do not have a center poison rod). The four-element CEAs each span two fuel assemblies at the core periphery's major axes.

The CEAs are positioned by magnetic jack control element drive mechanisms mounted on the reactor vessel head. The part-length CEAs are available to control axial power distribution if necessary.

The maximum reactivity worth of the CEAs and the associated reactivity addition rate are limited by system design to prevent sudden large reactivity increases. The design restraints are such that reactivity increases do not result in violation of the fuel damage limits, rupture of the reactor coolant pressure boundary, nor disruption of the core or other internals sufficient to impair the effectiveness of emergency cooling.

Boric acid dissolved in the coolant is used as a neutron absorber to
provide long-term reactivity control. In order to reduce the boric acid
3 concentration required at beginning-of-cycle operating conditions, and
thus reduce the algebraic magnitude of the moderator temperature
coefficient, burnable poison rods are provided in certain fuel assemblies.
The poison is boron carbide dispersed in alumina pellets; the pellets are
clad in Zircaloy to form rods that are similar to the fuel rods.

A three-batch fuel management scheme is employed, where approximately
1/3 of the core is replaced at each refueling. The average burnup will
3 be about 45,000 MWd/MTU over the three-cycle life of the fuel. Sufficient
margin is provided to ensure that peak burnups are within acceptable
limits.

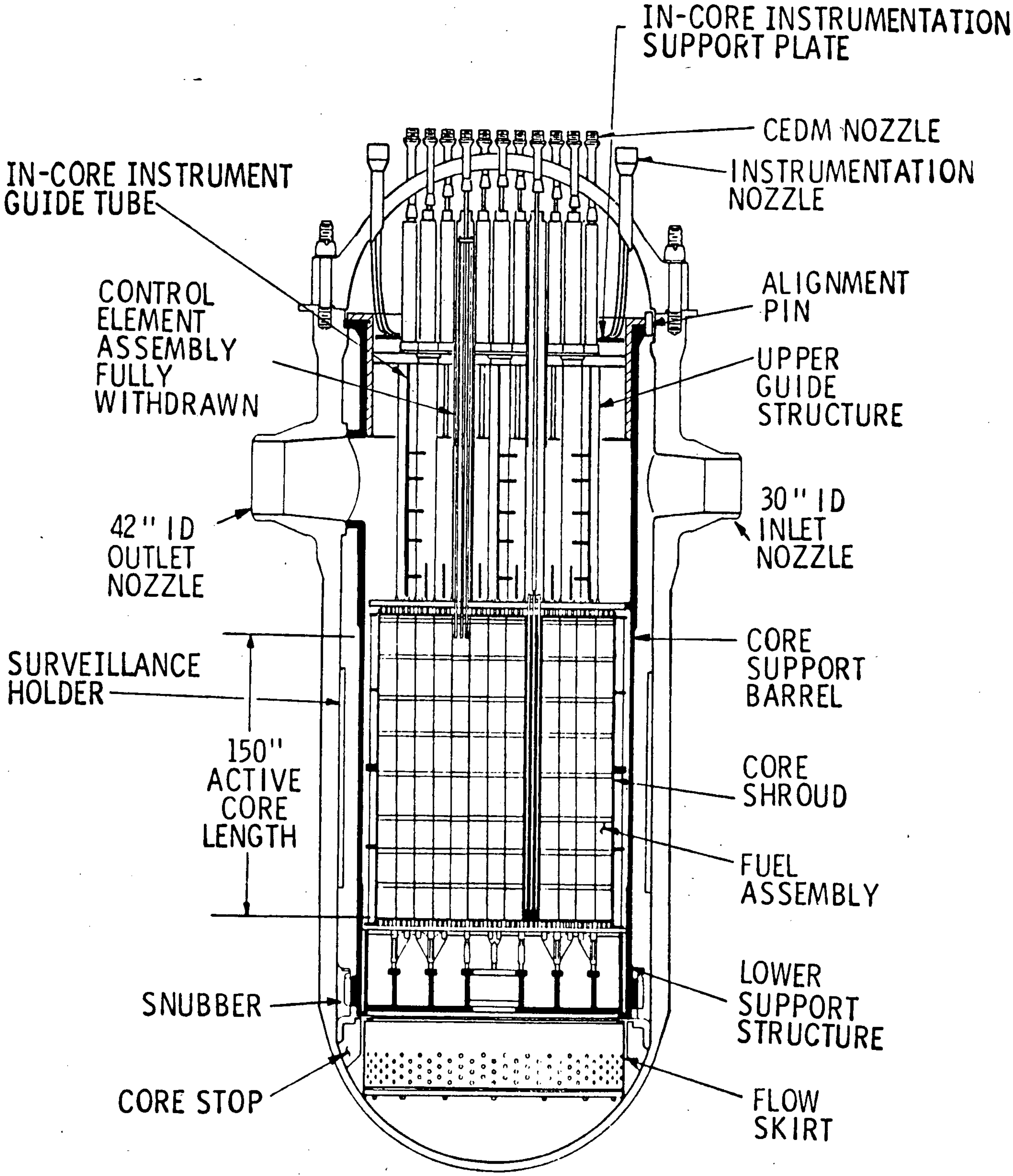
The nuclear design of the core ensures that the combined response of all reactivity coefficients in the power operating range to an increase in reactor thermal power yields a net decrease in reactivity.

Control element assemblies are moved in groups to satisfy the requirements of shutdown, power level changes, and operational maneuvering. The control system is designed to produce power distributions that are within the acceptable limits of overall nuclear heat flux factor (Fq) and departure from nucleate boiling ratio (DNBR). The reactor protective system and administrative controls ensure that these limits are not exceeded.

Axial xenon oscillations, should they occur, can be manually controlled by part-length CEAs, using information provided by the nuclear detectors.

The core also contains two plutonium 238-beryllium neutron sources for initial and subsequent startups. The sources are supported from the fuel assembly upper end fitting and are contained within CEA guide tubes.

Design of the reactor internals is discussed in subsections 3.9.5 and 4.5.2; fuel assembly design is discussed in section 4.2; nuclear design of the core is discussed in section 4.3; and the thermal and hydraulic design is discussed in section 4.4. Summary lists of significant core parameters are presented in tables 4.2-1, 4.3-1, and 4.4-1. A tabulation of the analysis techniques, load conditions, and computer codes utilized in the analyses of various reactor internals components is presented in table 4.1-1.



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SAN ONOFRE NUCLEAR GENERATING STATION Units 2 & 3
REACTOR VERTICAL ARRANGEMENT
Figure 4.1-1

4.5 REACTOR MATERIALS

4.5.1 CONTROL ELEMENT DRIVE STRUCTURAL MATERIALS

4.5.1.1 Material Specifications

A. The materials used in the control element drive mechanism (CEDM) reactor coolant pressure boundary components are as follows:

1. Motor Housing Assembly

   SA-182, Type 348 (austenitic stainless steel)

   SA-182, Type 403, and Code Case 1334-3 (martensitic stainless steel)

   SB-166 (nickel-chromium-iron alloy)

2. Upper Pressure Housing

   SA-213, Type 316 (austenitic stainless steel)

   SA-479, Type 316 (austenitic stainless steel)

   ASTM A276, Type 440 (martensitic stainless steel with yield strength greater than 90,000 lb/in.^{2})

The above listed materials, with the exception of the ASTM A276 Type 440 material, are also listed in Appendix I of the 1971 Edition of Section III of the ASME Boiler and Pressure Vessel Code, including the 1973 Winter Addenda. In addition, the materials comply with the 1971 Edition of Sections II and IX of the ASME Boiler and Pressure Vessel Code including the 1973 Winter Addenda.

The functions of the above listed components are described in paragraph 3.9.4.1.1.1.

B. The materials in contact with the reactor coolant used in the CEDM motor assembly components are as follows:

1. Latch Guide Tubes

   ASTM A269, Type 316 (austenitic stainless steel)

   Chrome Oxide (plasma spray treatment)

2. Magnet and Spacer

   ASTM A276, Type 410 (martensitic stainless steel)

3. Latch and Magnet Housing

   ASTM A276, Type 316 (austenitic stainless steel)

   QQ-C-320a, Class 2B (chrome plating)

4. Spacer

   ASTM A240, Type 304 (austenitic stainless steel)

5. Alignment Button

   AMS 5643J, Type 17-4 PH (martensitic stainless steel)

6. Spring

   AMS 5698B, Inconel X-750 (nickel base alloy)

7. Pin

   Haynes Stellite No. 6B (cobalt base alloy)

8. Dowel Pin

   ASTM A314, Type 410 (martensitic stainless steel)

9. Spacer and Screw

   ASTM A276, Type 321 (austenitic stainless steel)

10. Stop

    ASTM A276, Type 304 (austenitic stainless steel)

11. Latch and Pin

    Haynes Stellite No. 36 (cobalt base alloy)

12. Locking Cup and Screws

    300 Series austenitic stainless steel

The functions of the CEDM motor assembly components are described in paragraph 3.9.4.1.1.2.

C. The materials in contact with the reactor coolant used in the extension shafts are listed below:

1. Shafts, Rod, and Plunger

   ASTM A276, Type 304 (austenitic stainless steel)

   ASTM A269, Type 304 (austenitic stainless steel)

2. Gripper

   ASTM B446 (nickel-chromium-molybdenum-columbium alloy)

   QQ-C-320a, Class 2B (chrome plating)

3. Spring

   AMS 5699B, Inconel X-750 (nickel base alloy)

4. Pin

   304 austenitic stainless steel

The functions of the extension shaft components are described in paragraph 3.9.4.1.1.5.

D. The weld rod filler materials used with the above listed components are 308 stainless steel, 316 stainless steel, and Inconel 82.

All of the material listed in the above listings, A through D, were used in an extensively tested CEDM assembly that exceeded lifetime requirements, as described in paragraph 3.9.4.4.1. Also, all of the materials have performed satisfactorily in service in Maine Yankee (Docket 50-309), Millstone II (Docket 50-236), Calvert Cliffs (Docket 50-317), in addition to other designed reactors.

4.5.1.2 Control of the Use of 90 ksi Yield Strength Material

The only control element drive structural material identified in paragraph 4.5.1.1 which has a yield strength greater than 90 ksi is ASTM A276, Type 440, martensitic stainless steel. Its usage is limited to the steel ball in the vent valve on the top of the CEDM. The ball is used as a seal and is not a primary load bearing member of the pressure boundary. This material was tested and exceeded lifetime requirements. Also, this material is presently being used in operating reactors such as Maine Yankee (Docket 50-209), Millstone II (Docket 50-236), and Calvert Cliffs (Docket 50-317) and has performed satisfactorily for the same application.

4.5.1.3 Control of the Use of Sensitized Austenitic Stainless Steel

Control of the use of sensitized austenitic stainless steel is consistent with the recommendations of Regulatory Guide 1.44 as described in paragraphs 4.5.1.3.1 through 4.5.1.3.3, except for the criteria used to demonstrate freedom from sensitization. The ASTM A393 Strauss Test was used in lieu of the ASTM A262 Method E Modified Strauss Test to demonstrate freedom from sensitization in fabricated unstabilized austenitic stainless steel. The former test has shown, through experimentation, excellent correlation with the type of corrosion observed in severely sensitized austenitic stainless steel NSSS components.

4.5.1.3.1 Solution Heat Treatment Requirements

All raw austenitic stainless steel, both wrought and cast, employed in the fabrication of the control element drive system structural components is supplied in the solution annealed condition as described in paragraph 4.5.2.4.2.1.

4.5.1.3.2 Material Inspection Program

Extensive testing on stainless steel mockups, fabricated using production techniques, was conducted to determine the effect of various welding procedures on the susceptibility of unstabilized 300 series stainless steels to sensitization-induced intergranular corrosion. Only those procedures and/or practices demonstrated not to produce a sensitized structure were used in the fabrication of control element drive system structural components. The ASTM Standard A393 (Strauss Test) is the criterion used to determine susceptibility to intergranular corrosion. This test has shown excellent correlation with a form of localized corrosion peculiar to sensitized stainless steels. As such, ASTM A393 is used as a go/no-go standard for acceptability.

4.5.1.3.3 Avoidance of Sensitization

Homogeneous or localized heat treatment of unstabilized austenitic stainless steel in the temperature range 800-1500F is prohibited.

Weld heat affected zone sensitized austenitic stainless steel (which will fail the Strauss Test, ASTM A393) is avoided in control element drive system structural components by careful control of:

A. Weld heat input to less than 60 kJ/in.

B. Interpass temperature to 350F maximum.

4.5.1.4 Control of Delta Ferrite in Austenitic Stainless Steel Welds

The austenitic stainless steel, primary pressure-retaining welds in the control element drive system structural components are consistent with the recommendations of the Interim Position of Regulatory Guide 1.31, MTEB 5-1, as described in paragraph 4.5.2.4.5.

4.5.1.5 Cleaning and Contamination Protection Procedures

The procedure and practices followed for cleaning and contamination protection of the control element drive system structural components are as described in paragraph 4.5.2.4.1.

4.5.2 REACTOR INTERNALS MATERIALS

4.5.2.1 Material Specifications

The materials used in fabrication of the reactor internal structures are primarily Type 304 stainless steel. The flow skirt is fabricated from Inconel. Welded connections are used where feasible; however, in locations where mechanical connections are required, structural fasteners are used which are designed to remain captured in the event of a single failure. Structural fastener material is typically a high-strength austenitic stainless steel; however, in less critical applications Type 316 stainless steel is employed. Hardfacing of Stellite material is used at wear points. The effect of irradiation on the properties of the materials is considered in the design of the reactor internal structures. Work hardening properties of austenitic stainless steels are not used.

The following is a list of the major components of the reactor internals together with their material specifications:

A. Core Support Barrel Assembly

1. Type 304 austenitic stainless steel to the following specifications:

   ASTM A182

   ASTM A240

   ASTM A479

2. Precipitation hardening stainless steel to the following specifications:

   ASTM A453, Grade 660

   ASTM A638, Grade 660

B. Upper Guide Structure Assembly

1. Type 304 austenitic stainless steel to the following specifications:

   ASTM A182

   ASTM A240

   ASTM A269

   ASTM A312

   ASTM A451

   ASTM A479

2. Precipitation hardening stainless steel to the following specification:

   ASTM A453, Grade 660

C. Core Shroud Assembly

   1. Type 304 austenitic stainless steel to the following specifications:

      ASTM A182

      ASTM A240

D. Holddown Ring

   ASTM A182, Grade F-6, modified to ASME Code Case 1337-6 with exception to the temper temperature which shall be 1150F for 4 hours.

   The ASTM A182, Grade F-6 used for the holddown ring is heat treated to a minimum yield strength of 90,000 lb/in.^{2}. Under reactor operating conditions of low oxygen and slightly alkaline pH, a slightly higher (than austenitic stainless steel) but acceptable general corrosion rate is anticipated to occur. No localized corrosion is anticipated under these conditions. When heat treated in accordance with Code Case 1337, i.e., BHN 226-277 (HRC 21-29), Type 403 can be expected to be resistant to stress corrosion in the primary coolant. Stress corrosion failures in PWR environments have occurred only where the material has been heat treated to hardness levels higher than specified.(1)

E. Incore Instrument Support System

   1. Type 304 austenitic stainless steel to the following specifications:

      ASTM A193

      ASTM A194

      ASTM A240

      ASTM A249

      ASTM A269

      ASTM A276

      ASTM A312

      ASTM A473

ASTM A479

ASTM B353

ASTM B446

2. Zircaloy-4

ASTM B353

F. Bolt and Pin Material

ASTM A453 and ASTM A638, Grade 660 material (trade name A286) is used for bolting and pin applications. This alloy is heat treated to a minimum yield strength of 85,000 lb/in.^{2}. Its corrosion properties are similar to those of the 300 series austenitic stainless steels. It is austenitic in all conditions of fabrication and heat treatment. This alloy was used for bolting in previous reactor systems and test facilities in contact with primary coolant and has proven completely satisfactory.

G. Chrome Plating and Hardfacing

Chrome plating or hardfacing are employed on reactor internals components or portions thereof where required by function. Chrome plating complies with Federal Specification No. QQ-C-320a. The hardfacing material employed is Stellite 25.

All of the materials employed in the reactor internals and incore instrument support system have performed satisfactorily in operating reactors such as Palisades (Docket 50-255), Fort Calhoun (Docket 50-285), and Maine Yankee (Docket 50-309).

4.5.2.2 Welding Acceptance Standards

Welds employed on reactor internals and core support structures meet the acceptance standards delineated in article NG-5000 Section III, Division I, 1974 Edition, and control of welding has been performed in accordance with Sections III Division 1, and IX of the applicable ASME Code. In addition, consistency with the recommendations of Regulatory Guides 1.31 and 1.44 is described in paragraph 4.5.2.4.

4.5.2.3 Nondestructive Examination of Wrought Seamless Tubular Products and Fittings

Quality Group A components in the reactor internals which are wrought seamless tubular products or fittings are consistent with the recommendations of Regulatory Guide 1.66.

4.5.2.4 Fabrication and Processing of Austenitic Stainless Steel

The following information applies to unstabilized austenitic stainless steel as used in the reactor internals.

4.5.2.4.1 Cleaning and Contamination Protection Procedures

Specific requirements for cleanliness and contamination protection are included in the equipment specifications for components fabricated with austenitic stainless steel. The provisions described below indicate the type of procedures utilized for components to provide contamination control during fabrication, shipment, and storage.

Contamination of austenitic stainless steels of the 300 type by compounds that can alter the physical or metallurgical structure and/or properties of the material are avoided during all stages of fabrication. Painting of 300 series stainless steels is prohibited. Grinding is accomplished with resin or rubber-bonded aluminum oxide or silicon carbide wheels that have not previously been used on materials other than 300 series stainless alloys.

Internal surfaces of completed components are cleaned to the extent that grit, scale, corrosion products, grease, oil, wax gum, adhered or embedded dirt, or extraneous material are not visible to the unaided eye.

Cleaning is effected by either solvents (acetone or isopropyl alcohol) or inhibited water (30-200 ppm hydrazine). Water will conform to the following requirements:

Halides	0.60
Chloride, ppm	<0.60
Fluoride, ppm	<0.40
Conductivity, μmhos/cm	<5.0
pH	6.0-8.0
Visual clarity	No turbidity, oil, or sediment

To prevent halide-induced intergranular corrosion that could occur in an aqueous environment with significant quantities of dissolved oxygen, flushing water is inhibited via additions of hydrazine. Experiments have proven this inhibitor to be effective.[2] Operational chemistry specifications preclude halides and oxygen (both prerequisites of intergranular attacks) and are shown in subsection 9.3.4 and the Technical Specifications.

4.5.2.4.2 Control of the Use of Sensitized Austenitic Stainless Steel

The recommendations of Regulatory Guide 1.44, as described in paragraphs 4.5.2.4.2.1 through 4.5.2.4.2.5, were followed except for the criteria used to demonstrate freedom from sensitization. The ASTM A393 Strauss Test was used in lieu of the ASTM A262 Method E Modified Strauss Test to demonstrate freedom from sensitization in fabricated unstabilized austenitic stainless steel, since the former test has shown, through experimentation, excellent correlation with the type of corrosion observed in severely sensitized austenitic stainless steel NSSS components. Both ASTM A262 Method E and A393 were used as the acceptance criteria for raw austenitic stainless steel material.

4.5.2.4.2.1 Solution Heat Treatment Requirements. All raw austenitic stainless steel material, both wrought and cast, employed in the fabrication of the reactor internals is supplied in the solution annealed condition as specified by the pertinent ASTM or ASME B&PV Code material specification; viz, 1900 to 2050F for 1/2 to 1 h/in. of thickness and rapidly cooled to below 700F. The time at temperature is determined by the size and type of component.

Solution heat treatment is not performed on completed or partially-fabricated components. Rather, the extent of chromium carbide precipitation is controlled during all stages of fabrication as described in paragraph 4.5.2.4.2.4.

4.5.2.4.2.2 Material Inspection Program. Extensive testing of stainless steel mockups, fabricated using production techniques, was conducted to determine the effect of various welding procedures on the susceptibility of unstabilized 300 series stainless steels to sensitization-induced intergranular corrosion. Only those procedures and/or practices demonstrated not to produce a sensitized structure were used in the fabrication of reactor internals components. The ASTM Standard A393 (Strauss Test) is the criterion used to determine susceptibility to intergranular corrosion. This test has shown excellent correlation with a form of localized corrosion peculiar to sensitized stainless steels. As such, ASTM A393 is utilized as a go/no-go standard for acceptability.

As a result of the above tests, a relationship was established between the carbon content of Type 304 stainless steel and weld heat input. This relationship is used to avoid weld heat affected zone sensitization as described in paragraph 4.5.2.4.2.4.

4.5.2.4.2.3 Unstabilized Austenitic Stainless Steels. The unstabilized grade of austenitic stainless steel with a carbon content greater than 0.03% used for components of the reactor internals is Type 304. This material is furnished in the solution annealed condition. The acceptance criteria used for this material as furnished from the steel supplier is ASTM A262 Method E or ASTM A393.

Exposure of completed or partially fabricated components to temperatures ranging from 800 to 1500F is prohibited except as described in paragraph 4.5.2.4.2.5.

Duplex, austenitic stainless steels, containing >5 vol% delta ferrite (weld metal, cast metal, weld deposit overlay), are not considered unstabilized since these alloys do not sensitize; i.e., form a continuous network of chromium-iron carbides. Specifically, alloys in this category are:

- CF8M — Cast stainless steels (delta ferrite controlled to 5-25 vol%)
- CF8
- Type 308 — Singly and combined
- Type 309 — Stainless steel weld filler metals. (delta ferrite controlled to 5-18 vol% as deposited.)
- Type 312
- Type 316

In duplex austenitic/ferrite alloys, chromium-iron carbides are precipitated preferentially at the ferrite/austenitic interfaces during exposure to temperatures ranging from 1000 to 1500F. This precipitate morphology precludes intergranular penetrations associated with sensitized 300 series stainless steels exposed to oxygenated or otherwise faulted environments.

4.5.2.4.2.4 <u>Avoidance of Sensitization</u>. Exposure of unstabilized austenitic 300 stainless steels to temperatures ranging from 800 to 1600F will result in carbide precipitation. The degree of carbide precipitation or sensitization depends on the temperature, the time at that temperature, and also the carbon content. Severe sensitization is defined as a continuous grain boundary chromium-iron carbide network. This condition induces susceptibility to intergranular corrosion in oxygenated aqueous environments, as well as those containing halides. Such a metallurgical structure will readily fail the Strauss Test, ASTM A393. Discontinuous precipitates (i.e., an intermittent grain boundary carbide network) are not susceptible to intergranular corrosion in a PWR environment.

Weld heat affected zone sensitized austenitic stainless steels (which will fail the Strauss Test, ASTM A393) were avoided by careful control of:

- Weld heat input
- Interpass temperature
- Carbon content

A weld heat input of less than 60 kJ/in. is used during most fabrication stages of the Type 304 stainless steel core support structure. Higher heat inputs are used in some heavy section weld joints. Freedom from weld heat-affected zone sensitization in these higher heat input weldments is demonstrated with weld runoff samples produced at the time of component welding in material having a carbon content equal to or greater than the highest carbon content of those heats of steel being fabricated. Specimens so provided are subjected to the Strauss Test, ASTM A393.

4.5.2.4.2.5 Retesting Unstabilized Austenitic Stainless Steels Exposed to Sensitizing Temperature. Sensitization that may be susceptible to intergranular corrosion is avoided during welding as described in paragraph 4.5.2.4.2.4. Homogeneous or localized heat treatment of unstabilized stainless steels in the temperature range of 800 to 1500F is prohibited except in the case of the core support structure. This complex substructure is thermally stabilized at 900 ±25F for 7 hours after fabrication and prior to final machining. Such treatment produces only minor, discontinuous precipitates. In addition to thermocouple records during this heat treatment, a sample of Type 304 stainless steel having a carbon content equal to or greater than the highest carbon heat of material present in the structure is included as a monitor sample. After heat treatment, the monitor sample is subject to the Strauss Test, ASTM A393, as well as a metallographic examination to verify freedom from sensitization.

4.5.2.4.3 Control of Delta Ferrite in Welds

The recommendations of the Interim Position on Regulatory Guide 1.31, MTEB 5-1, were followed in the following manner:

A. The delta ferrite content of A-7 austenitic stainless steel filler metal used in the fabrication of major components of the reactor internals, was controlled to 5-20 vol% (FN5-23). Delta ferrite content was predicted either by chemical analysis performed on undiluted weld deposits using the Schaeffler or McKay diagram or by a calibrated magnetic measuring instrument. In the case of metal used with a nonconsumable electrode process, the delta ferrite content may be predicted by chemical analysis of the rod, wire, or consumable insert in conjunction with the stainless steel constitution diagram. The ferrite recommendations are met for each heat, lot, or heat/lot combination of weld filler material.

B. The average minimum delta ferrite content of production welds is 3% (FN3) as measured on an audit type basis.

4.5.2.4.4 Control of Electroslag Weld Properties

The electroslag process was not used to fabricate reactor internal components.

4.5.2.4.5 Welder Qualification for Areas of Limited Accessibility

The specific recommendations of Regulatory Guide 1.71 were not followed. However, performance qualifications, for personnel welding under conditions of limited accessibility, are conducted and maintained in accordance with the requirements of ASME Boiler and Pressure Vessel Code Sections III and IX. A requalification is required when:

A. Any of the essential variables of Section IX are changed.

B. When authorized personnel have reason to question the ability of the welder to satisfactorily perform to the applicable requirements.

Production welding is monitored for compliance with the procedure parameters and welding qualification requirements are certified in accordance with Sections III and IX. Further assurance of acceptable welds of limited accessibility is afforded by the welding supervisor assigning only the most highly skilled personnel to these tasks. Finally, weld quality, regardless of accessibility, is verified by the performance of the required nondestructive examination.

4.5.2.4.6 Nonmetallic Thermal Insulation

Nonmetallic thermal insulation is not used on the reactor internals.

4.5.2.5 <u>Contamination Protection and Cleaning of Austenitic Stainless Steel</u>

Comparison with the recommendations of Regulatory Guide 1.37, Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants, is discussed in Appendix 3A.

Figure 3

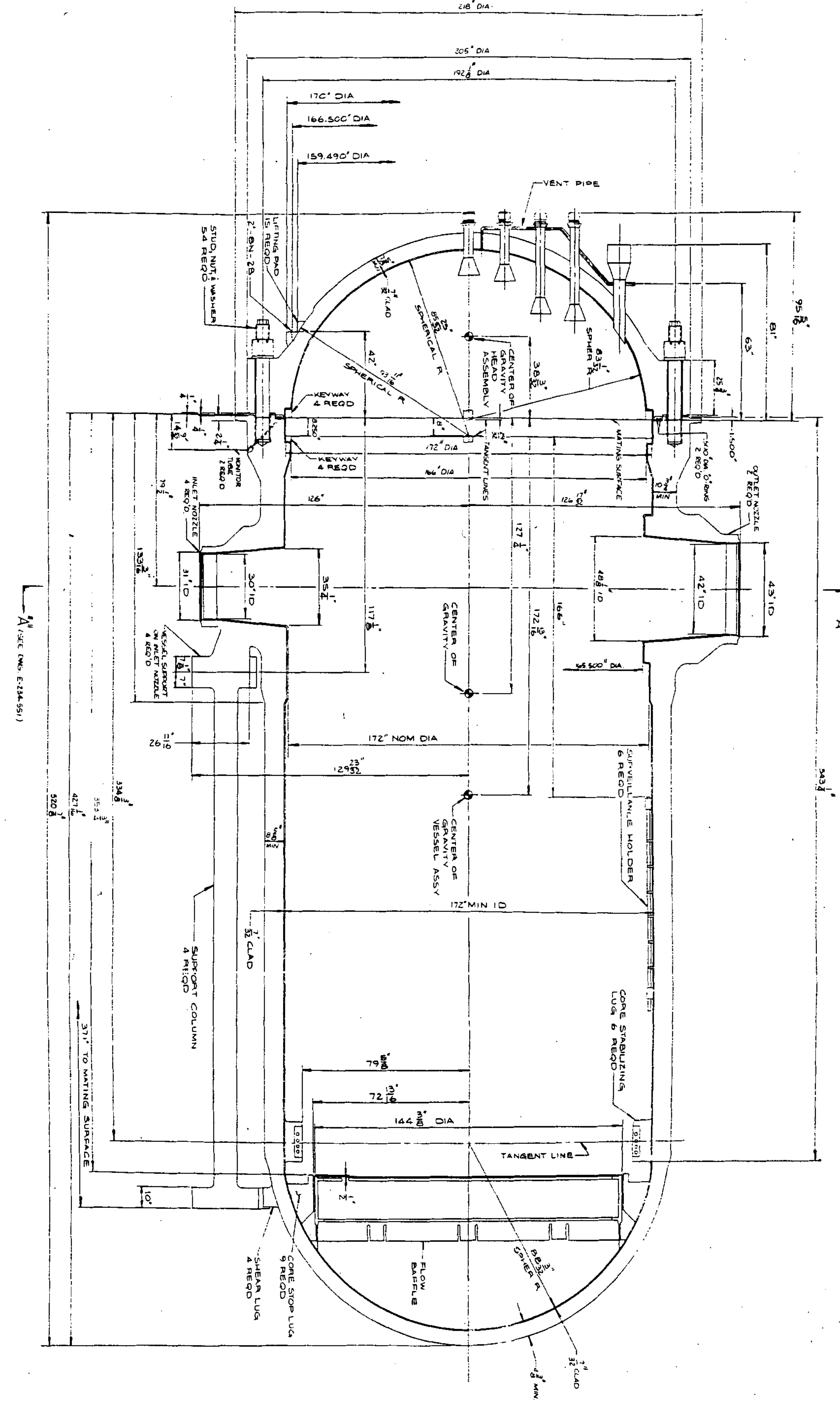
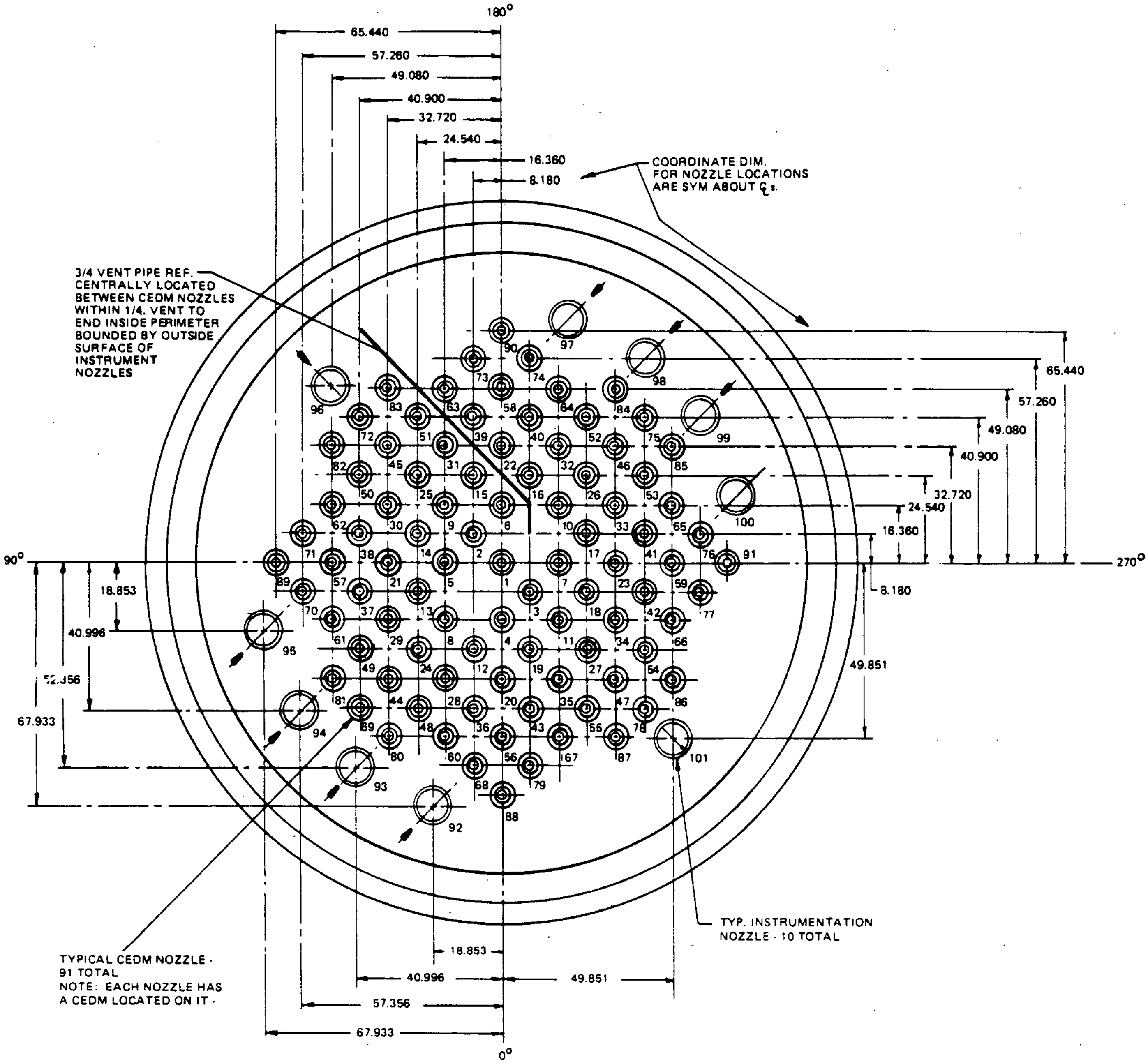

VENT PIPE
STUD, NUT & WASHER 54 REQD
LIFTING PAD 15 REQD
KEYWAY 4 REQD
SPHERICAL R
CENTER OF GRAVITY HEAD ASSEMBLY
SPHER R
MONITOR TUBE 2 REQ'D
TANGENT LINES
MATING SURFACE
OUTLET NOZZLE 2 REQ'D
INLET NOZZLE 4 REQ'D
CENTER OF GRAVITY
VESSEL SUPPORT ON INLET NOZZLE 4 REQ'D
172" NOM DIA
CENTER OF GRAVITY VESSEL ASSY
SURVEILLANCE HOLDER 6 REQD
172" MIN I.D
SUPPORT COLUMN 4 REQD
CORE STABILIZING LUG 6 REQD
371" TO MATING SURFACE
TANGENT LINE
FLOW BAFFLE
SHEAR LUG 4 REQD
CORE STOP LUG 9 REQD
218" DIA
205" DIA
170" DIA
166.500" DIA
159.490" DIA
30" ID
31" ID
42" ID
43" ID
"A"

Figure 4



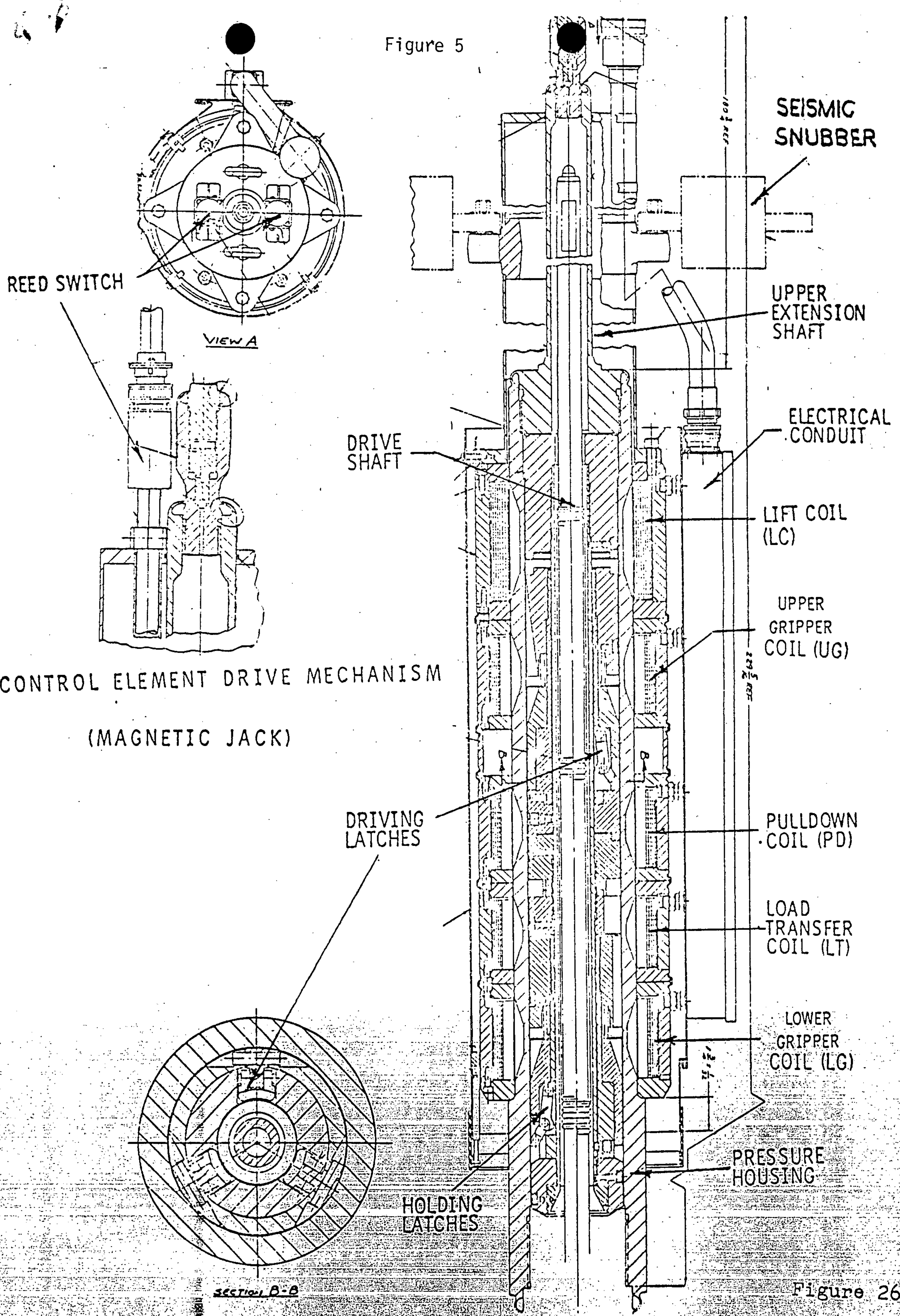
Updated

SAN ONOFRE NUCLEAR GENERATING STATION
Units 2 & 3

REACTOR VESSEL
CLOSURE HEAD PLAN VIEW
CEDM LAYOUT

Figure 4.6-1

Figure 5



CONTROL ELEMENT DRIVE MECHANISM

(MAGNETIC JACK)

Figure 26