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4.0 REACTOR COOLANT SYSTEM

The Reactor Coolant System, shown in Flow Diagrams, Figures 4.2-1, and 4.2-9 through 4.2-13 consists of three similar heat transfer loops connected in parallel to a reactor vessel. Each loop contains a circulating pump and a steam generator. The system also includes a pressurizer, pressurizer relief tank, connecting piping, and instrumentation necessary for operational control and protection.

4.1 DESIGN BASES

4.1.1 PERFORMANCE OBJECTIVES

The Reactor Coolant System transfers the heat generated in the core to the steam generators where steam is generated to drive the turbine generator. Borated demineralized light water is circulated at the flow rate and temperature consistent with achieving the reactor core thermal-hydraulic performance presented in Section 3. The water also acts as a neutron moderator and reflector, and as a solvent for the neutron absorber used in chemical shim control.

The Reactor Coolant System provides a boundary for containing the coolant under operating temperature and pressure conditions. It serves to confine radioactive material and limits to acceptable values its release to the secondary system and to other parts of the unit under conditions of either normal or abnormal reactor operation. During transient operation the system's heat capacity attenuates thermal transients generated by the core or extracted by the steam generators. The Reactor Coolant System accommodates coolant volume changes within the protection system criteria.

The thermal hydraulic effects consequent on loss of power to the reactor coolant pumps are reduced to acceptable levels by appropriate selection of the inertia of the reactor coolant pumps. The layout of the system ensures natural circulation capability following a loss of flow to permit cooldown without overheating the core. Part of the system's piping is used by the Emergency Core Cooling System to deliver cooling water to the core during a loss-of-coolant accident.

4.1.2 1967 NRC PROPOSED GENERAL DESIGN CRITERIA (GDC)

The following discussion refers to Turkey Point Plant commitments to the 1967 Proposed General Design Criteria as documented by the U.S. Atomic Energy Commission in Reference 1. Due to the vintage of the Turkey Point Plant, there is no correlation between the 1967 Proposed GDC and those criteria currently contained in 10 CFR 50, Appendix A.

General design criteria which apply to the Reactor Coolant System are given below.

<u>Quality Standards</u>

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. (1967 Proposed GDC 1)

The Reactor Coolant System is of primary importance with respect to its safety 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 (Section 4.1.7). Details of the quality assurance programs, test procedures and inspection acceptance levels are given in Section 4.3.1 and 4.4. 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.

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 phenomenon such as earthquake, tornado, flooding condition, high wind or heavy ace. 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.
(1967 Proposed GDC 2)

All piping, components and supporting structures of the Reactor Coolant System are designed as Class I equipment; i.e., they are capable of withstanding:

(a) The design seismic ground acceleration within code allowable working stresses.

(b) The maximum potential seismic ground acceleration acting in the horizontal and vertical direction simultaneously with no loss of capability to perform their safety function.

Details are given in Section 4.1.4.

The Reactor Coolant System is located in the containment building whose design, in addition to being a Class I structure, also considers accidents or other applicable natural phenomena. Details of the containment design are given in Section 5.

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. (1967 Proposed GDC 5)

Records of the design, of the major Reactor Coolant System components and the related engineered safety features components are maintained in the offices of Florida Power and Light Company and will be retained there throughout the life of the unit.

Records of fabrication are maintained in the manufacturers' plants as required by the appropriate Code, or other requirements pending submittal to Westing-house or Florida Power and Light Company. They are available at any time to Florida Power and Light throughout the life of the unit. Construction records are available at the construction site and in the offices of Florida Power and Light Company where they will be retained for the life of the unit.

Missile Protection

Criterion: Adequate protection for those engineered safety features, the failures of which could cause an undue risk to the health and safety of the public, shall be provided against dynamic effects and missiles that might result from plant equipment failures. (1967 Proposed GDC 40)

The dynamic effects during blowdown following a loss-of-coolant accident are evaluated in the detailed layout and design of the high pressure equipment and barriers which afford missile protection. Fluid and mechanical driving forces are calculated, and consideration is given to possible damage due to fluid jets and secondary missiles which might be produced.

The steam generators are supported, guided and restrained in a manner which prevents rupture of the steam side of a generator, the steam lines and the feedwater piping as a result of forces created by a Reactor Coolant System pipe rupture. These supports, guides and restraints also prevent rupture of the primary side of a steam generator as a result of forces created by a steam or feedwater line rupture.

The mechanical consequences of a pipe rupture are restricted by design such that the functional capability of the engineered safety features is not impaired.

A discussion on missile protection is given in Section 6.2.

Environmental and Dynamic Effects

The following general design criteria is contained in 10 CFR 50, Appendix A.

Criterion: The reactor coolant system shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing and postulated accidents (10 CFR 50, Appendix A - GDC 4).

The NRC documents in their letter of November 28, 1988 (Reference 1) that the leakage detection systems at Turkey Point Units 3 and 4 satisfy the requirements of Generic Letter 84-04, and that the primary loop piping complies with the criteria of GDC 4 from 10 CFR 50, Appendix A. GDC 4 allows the use of plant-specific Leak-Before-Break analysis to eliminate the dynamic effects of postulated pipe ruptures in high energy piping from the design basis of a plant. Plants with an NRC-approved Leak-Before-Break analysis may remove pipe whip restraints and jet impingement barriers. Turkey Point Units 3 and 4 received NRC approval (Reference 2) for elimination of the dynamic effects of postulated pipe ruptures in reactor coolant piping from the design basis of the plant. The Turkey Point analysis for the Leak-Before-Break Methodology is documented in the Westinghouse report WCAP-14237 (Reference 3).

4.1.3 PRINCIPAL DESIGN CRITERIA

The criteria which apply solely to the Reactor Coolant System are given below:

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. (1967 Proposed GDC 9)

The Reactor Coolant System in conjunction with its control and protective provisions is designed to accommodate the system pressures and temperatures attained under all expected modes of station operation or anticipated system interactions, and maintain the stresses within applicable code stress limits.

Fabrication of the components which constitute the pressure boundary of the Reactor Coolant System is carried out in strict accordance with the applicable codes. In addition there are areas where equipment specifications for Reactor

Coolant System components go beyond the applicable codes. Details are given in Section 4.4.1.

The materials of construction of the pressure boundary of the Reactor Coolant System are protected by control of coolant chemistry from corrosion phenomena which might otherwise reduce the system structural integrity during its service lifetime.

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 to a safe level.

The system is protected from overpressure by means of pressure relieving devices, as required by Section III of the ASME Boiler and Pressure Vessel Code.

Isolatable sections of the system are provided with overpressure relieving devices discharging to closed systems such that the system code allowable relief pressure is not exceeded within the protected section.

Monitoring Reactor Coolant Leakage

Criterion: Means shall be provided to detect significant uncontrolled leakage from the reactor coolant pressure boundary. (1967 Proposed GDC 16)

The methods by which significant leakage from the Reactor Coolant System is detected are discussed in Section 6.5.

Further details are supplied in Section 4.2.7.

Monitoring of reactor vessel flange leakage is discussed in Section 4.2.2.

Reactor Coolant Pressure Boundary Capability

Criterion: The reactor coolant pressure boundary shall be capable of accommodating without rupture the static and dynamic load 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. (1967 Proposed GDC 33)

The reactor coolant boundary is shown to be capable of accommodating without rupture, the static and dynamic loads imposed as a result of a sudden reactivity insertion such as a rod ejection. Details of this analysis are provided in Section 14.

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

By using the flexibility in the selection of control rod groupings, radial locations 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 primary system pressure boundary from 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 Reactor Coolant System and the reactor containment. The environmental consequences of rod ejection are less severe than from the maximum hypothetical loss-of-coolant accident, for which public health and safety is shown to be adequately protected. Reference is made to Section 14.

Rod drop out is positively prevented by the mechanical design of the core and rod cluster control assemblies.

Since no means are provided to isolate individual loops, and since natural circulation occurs if the system is hot and the reactor coolant pumps are not running, it is not possible to preferentially cool a large volume of water that could be swept into the core. It is also not possible to rapidly add cold unborated water to the system. Therefore reactivity insertion from cold water addition does not pose any threat to the integrity of the Reactor Coolant System.

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. (1967 Proposed GDC 34)

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 exposure to fast neutrons. 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 reached until the affected vessel material is above the Design Transition Temperature (DTT), and in the ductile material region. The DTT is a minimum of NDTT plus 60°F and dictates the procedures to be followed in the hydrostatic test and in unit operations to avoid excessive cold stress. The pressure during startup and shutdown, at temperatures below NDTT, is maintained below the threshold of concern for safe operation.

The value of the DTT is increased during the life of the unit as required by the expected shift in the NDTT temperature, and as confirmed by the experimental data obtained from irradiated specimens of reactor vessel materials. Further details are given in Section 4.1.6 and Appendix 4A.

All pressure-containing components of the reactor coolant system are designed, fabricated, inspected and tested in conformance with the applicable codes. Further details are given in Section 4.1.7.

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.
(1967 Proposed GDC 36)

The design of the reactor vessel and its arrangement in the system permits access during the service life to the entire internal surfaces of the vessel and to the following external zones of the vessel: the flange seal surface, the flange O.D. down to the cavity seal ring, the closure head except around the drive mechanism adapters and the nozzle to reactor coolant piping welds. The reactor arrangement within the containment provides sufficient space for inspection of the external surfaces of the reactor coolant loop components and 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 the version of ASTM E185,"Recommended Practice for Surveillance Tests on Structural Materials in Nuclear Reactors," required by 10 CFR 50, Appendix H. 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 tests. 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 on heatup and cooldown 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 credible.

4.1.4 DESIGN CHARACTERISTICS

Design Pressure

The Reactor Coolant System design and operating pressure together with the safety, power relief and pressurizer spray valves set points, and the protection system set point pressures are listed in Table 4.1-1. The design pressure allows for operating transient pressure changes. The selected design margin considers core thermal lag, coolant transport times and pressure drops, instrumentation and control response characteristics, and system relief valve characteristics. The design pressures and data for the respective system components are listed in Tables 4.1-2 through 4.1-6. Table 4.1-7 gives the design pressure drop of the system components.

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 4.1-2 through 4.1-6.

Seismic Loads

The seismic loading conditions are established by the "design earthquake" and "maximum potential earthquake". The former is selected to be typical of the largest probable ground motion based on the site seismic history. The latter is selected to be the largest potential ground motion at the site based on seismic and geological factors and their uncertainties.

For the "design earthquake" 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 normal design limits. The seismic design for the "maximum potential earthquake" is intended to provide a margin in design that assures capability to shut down and maintain the nuclear facility in a safe condition. In this case, it is only necessary to ensure that the Reactor Coolant System 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 "no-loss-of-function earthquake" loading condition.

The criteria adopted for allowable stresses and stress intensities in vessels and piping subjected to normal loads plus seismic loads are defined in Appendix 5A.

For the combination of normal plus design earthquake loadings, the stresses in the support structures are kept within the limits of the applicable codes.

For the combination of normal plus no-loss-of-function earthquake loadings, the stresses in the support structures are limited to values necessary to ensure their integrity, and to keep the stresses in the Reactor Coolant System components within the allowable limits as given in Appendix 5A.

4.1.5 CYCLIC LOADS

All components in the Reactor Coolant System are designed to withstand the effects of cyclic loads due to reactor coolant system temperature and pressure changes. These cyclic loads are introduced by normal power changes, reactor trip, and startup and shutdown operation. The number of thermal and loading cycles used for design purposes and their bases are given in Table 4.1-8. During unit startup and shutdown, the rates of temperature and pressure changes are limited as indicated in Section 4.4.1. The cycles are estimated for equipment design purposes and are not intended to be an exact representation of actual transients or actual operating experience. For example the number of cycles for unit heatup and cooldown at 100°F per hour was selected as a conservative estimate based on an evaluation of the expected requirements. The resulting number could be increased significantly; however, it is the intent to represent a conservative realistic number rather than the maximum allowed by the design.

Although loss of flow and loss of load transients are not included in Table 4.1-8 since the tabulation is only intended to represent normal design transients, the effect of these transients have been analytically evaluated and are included in the fatigue analysis for primary system components.

Table 4.1-10 provides the component cyclic or transient limits for the Reactor Coolant System and Secondary Coolant System as was relocated from Technical Specification Section 5.6 by License Amendments 251 and 247 (Reference 8) for Units 3 and 4, respectively. C26

Over the range from 15% full power up to but not exceeding 100% of full power, the Reactor Coolant System and its components are designed to accommodate 10% of full power step changes in unit load and 5% of full power per minute ramp changes without reactor trip. The turbine bypass and steam dump system make it possible to accept a step load decrease of 50% of full power without reactor trip.

4.1.6 SERVICE LIFE

The service life of Reactor Coolant System pressure components depends upon the material irradiation, unit operational thermal cycles, quality manufacturing standards, environmental protection, and adherence to established operating procedures.

The reactor vessel is the only component of the Reactor Coolant System which is exposed to a significant level of neutron irradiation and it is therefore the only component which is subject to any appreciable material irradiation effects. The NDTT shift of the vessel material and welds, due to radiation damage effects, is monitored by a radiation damage surveillance program which conforms with ASTM - E 185 standards.

Reactor vessel design is based on the transition temperature method of evaluating the possibility of brittle fracture of the vessel material, as result of operations such as leak testing and heatup and cooldown.

To establish the service life of the Reactor Coolant System components as required by the ASME (part III), Boiler and Pressure Vessel Code for Class "A" Vessels, the unit operating conditions have been established for the initial 40 year design life. These operating conditions include the cyclic application of pressure loadings and thermal transients. The evaluation for extended plant design life concludes that the 40-year design cycles envelope the 80-year extended design life. C31

The number of thermal and loading cycles used for design purposes are listed in Table 4.1-8. Component Cyclic or Transient Limits are listed in Table 4.1-10

Metal fatigue considerations, including Reactor vessel underclad cracking, have also been evaluated for the extended plant life as discussed in UFSAR Chapter 16. The analysis associated with reactor vessel underclad crack growth has been projected to the end of the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(ii) as indicated in UFSAR Chapter 16. Underclad cracking has been evaluated by Westinghouse in WCAP-15338, “A Review of Cracking Associated with Weld Deposited Cladding in Operating PWR Plants.”

4.1.7 CODES AND CLASSIFICATIONS

All pressure-containing components of the Reactor Coolant System are designed, fabricated, inspected and tested in conformance with the applicable codes listed in Table 4.1-9. The Reactor Coolant System is classified as Class I for seismic design, requiring that there will be no loss of function of such equipment in the event of the assumed maximum hypothetical ground accelerations acting in the horizontal and vertical directions simultaneously, when combined with the primary steady state stresses.

Reactor coolant system valves, fittings and piping were designed, fabricated, inspected and tested in conformance with the Code requirements listed in Table 4.1-9. Hydrostatic testing of piping and fittings is done after installation at the pressure given in Table 4.1-6, which is the reactor coolant system test pressure also. This is 1 1/4 times design pressure and is a necessary deviation from Code Case N-10.

Reactor Coolant System inservice Inspection of ASME Code Class 1, 2, and 3 components shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).

Prior approval is not required for ASME BP&V Code Cases listed in Regulatory Guide 1.147, Inservice Inspection Code Case Acceptability, ASME XI, Division 1. All provisions of the Code Case must be met along with limitations issued in Regulatory Guide 1.147, if any. Code Cases not listed in Regulatory Guide 1.147 must receive specific permission for use (i.e. Relief Request) from the USNRC prior to their use, pursuant to 10 CFR 50.55a(g)(6)(i).

Inservice inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

4.1.8 REFERENCES

1. NRC Letter, from G.E. Edison (NRC) to W.F. Conway (FPL), "Turkey Point Units 3 and 4 - Generic Letter 84-04, Asymmetric LOCA Loads," dated November 28, 1988.

2. NRC Letter, from R. P. Croteau (NRC) to J. H. Goldberg (FPL), "Turkey Point Units 3 and 4 - Approval to Utilize Leak-Before-Break Methodology for Reactor Coolant System Piping (TAC Nos. M91494 and M91495)," dated June 23, 1995.

3. Westinghouse WCAP-14237, "Technical Justification for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for the Turkey Point Units 3 and 4 Nuclear Power Plants," dated December 1994.

4. WCAP-15092 Revision 3, Turkey Point Units 3 and 4 WOG Reactor Vessel 60-Year Evaluation Minigroup Heatup and Cooldown Limit Curves for Normal Operation, Westinghouse, May 2000.

5. Low Temperature Overpressure Protection System Setpoints, Florida Power & Light Company, 32 EFPY and 48 EFPY for Turkey Point, Units 3 & 4. Westinghouse Electric Company LLC, June 2000.

6. PC/M 03-057, Rev. 01, "Reactor Vessel Closure Head Replacement"-Unit 3.

7. PC/M 03-058, Rev.0 "Reactor Vessel Closure Head Replacement"-Unit 4.

8. NRC (Paige) to FPL (Nazar), "Turkey Point Units 3 and 4. Issuance of Amendments Regarding Section 5.0 Design Features (TAC Nos. ME6334 and ME6335)," dated June 21, 2012.

9. Westinghouse WCAP-17887-P, Revision 2, Determination of Acceptable Baffle-Former Bolting for Turkey Point Units 3 and 4, December 2018. C30

TABLE 4.1-1

REACTOR COOLANT SYSTEM DESIGN PARAMETERS AND PRESSURE SETTINGS

DESCRIPTION	PARAMETER	
Total Primary Heat Output, MWt	2652	C26
Total Primary Heat Output, Btu/hr	9049×10^6	
Number of Loops	3	
Coolant Volume, including pressurizer volume, ft^3	9343	
Total Reactor Coolant Flow, gpm (MMF)	270,000	C26
Design Pressure	2485 psig	
Operating Pressure (at pressurizer)	2235 psig	
Safety Valves	2465 (+2%,-3%) psig	C26
Power Relief Valves :		
i) Normal Operation	2335 psig	
ii) OMS Actuation During Heatup and Cooldown		
a) RCS ≤ 285°F	440 psig (Note 2)	C26
b) Above 285°F, the OMS setpoints are established by the Plant Curve Book, Section V, Figure 3D. The most restrictive cool-down rate curve (typically 100°F/hr) and the 60°F subcooling curve bound the setpoint curve.[1]		
Pressurizer Spray Valves (Open)	2250 psig	C26
High Pressure Trip	2385 psig	
High Pressure Alarm	2300 psig	C26
Low Pressure Trip	1835 psig	
Low Pressure Alarm	2185 psig	
Hydrostatic Test Pressure	3107 psig	

NOTES:

1. OMS is not normally in-service at RCS temperatures greater than 300°F.

2. Technical Specification LCO 3.4.9.3a indicates a PORV lift setting of ≤ 448 psig, however, the field device for OMS setpoint actuation will be at 440 psig to provide buffer from the Technical Specifications value. C26

TABLE 4.1-2

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 Housing)	42-7
Water Volume, (with core and internals in place), ft^3	3667
Thickness of Insulation, min., in. (Unit 3 & 4 Vessels)	3
Thickness of Insulation, min., in (Unit 3 & 4 RVCH)	5
Number of Reactor Closure Head Studs	58
Diameter of Reactor Closure Head Studs, in.	6
Flange, ID, in.	149.6
Flange, OD, in.	184
ID at Shell, in.	155.5
OD across inlet/outlet nozzles, in.	230-5/16 / 240
Inlet Nozzle ID, in.	Tapered 27-15/32 to 33-13/16
Outlet Nozzle ID, in.	28 31/32
Clad Thickness, min., in.	0.156
Lower Head Thickness, min., in.	4-3/4 plus cladding
Vessel Belt-Line Thickness, min., in.	7-3/4 plus cladding
Closure Head Thickness, in.	6-3/16 plus cladding
Reactor Coolant Inlet Temperature, °F	535.5-549.2
Reactor Coolant Outlet Temperature, °F	604.5-616.8
Reactor Coolant Flow, lb/hr	98.1×10^6 - 99.9×10^6

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TABLE 4.1-2a

CHEMICAL ANALYSES IN WEIGHT PERCENT
REACTOR VESSEL SURVEILLANCE MATERIAL

Element	Intermediate Shell		Lower Shell		
	Unit 3	Unit 4	Unit 3	Unit 4	
C	0.20	0.22	0.20	0.21	
Mn	0.64	0.67	0.61	0.67	
P	0.010	0.010	0.010	0.011	
S	0.010	0.009	0.008	0.009	
Si	0.26	0.20	0.20	0.23	
Ni	0.70	0.68	0.67	0.74	C25
Cr	0.40	0.33	0.38	0.31	
V	0.02	0.002	0.02	0.001	
Mo	0.62	0.56	0.58	0.56	
Co	0.011	0.017	0.015	0.015	
Cu	0.058	0.054	0.079	0.056	
Zr	*0.001	0.005	*0.001	0.004	
Sn	0.010	0.008	0.008	0.008	
Ti	*0.001	*0.001	*0.001	*0.001	
Sb	*0.001		*0.001		
Zn	0.001	*0.001	0.001	*0.001	
As	*0.005	0.004	*0.005	0.005	
B	*0.003	*0.003	*0.003	*0.003	
Al	0.005	0.008	0.005	0.008	
N^2	0.003	0.001	0.003	0.002	
Nb		0.002		0.001	
W		*0.001		*0.001	
Pb		*0.001		0.001	
Ta		0.003		0.002	

* Not detected. The number indicates the minimum limit of detection.

TABLE 4.1-3

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, ft³ *	766
Steam Volume, Full Power, ft³	534
Surge Line Nozzle Diameter, in./Pipe Schedule	14/Sch 140
Shell ID, in./Minimum Shell Thickness, in.	84/4.1
Minimum Clad Thickness, in.	0.188
Electric Heaters Capacity, kw (total)**	1300(Design)
Heatup rate of Pressurizer using Heaters only, °F/hr	55 (approximately with Design heater capacity)
Power Relief Valves: #455C & 456	
Number	2
Set Pressure (open), psig	
i) Normal operation	2335
ii) OMS Actuation during Heatup or Cooldown	
a) RCS ≤ 285°F	440***
b) RCS > 285°F	Setpoint increases to 2335 psig at 554°F as a segmented curve. Above 554°F and up to 750°F, the setpoint is maintained at a constant 2335 psig.
Capacity, lb/hr saturated steam/valve	179,000
Safety Valves	
Number	3
Set Pressure, psig	2465 ±1% (as left) +2%/-3% (as found)
Capacity, lb/hr saturated steam/valve	313,826
Pressurizer Relief Tank	
Design pressure, psig	100
Rupture disc release pressure, psig	100
Design temperature, °F	340
Normal water temperature, °F	120
Total volume, ft³	1300
Rupture disc relief capacity, lb/hr	931,964 Each

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TABLE 4.1-3 Sheet 2 of 2 C26

PRESSURIZER AND PRESSURIZER RELIEF TANK DESIGN DATA

* This volume corresponds to the maximum pressurizer level of 60% of span at full power conditions. Surge line volume is not included.

** Original as-built design. The extended power uprate analysis supports a minimum pressurizer heater capacity of 1000 kw. C26

*** Technical Specifications LCO 3.4.9.3a indicates a PORV lift setting of $\leq$ 448 psig, however, the field device for OMS setpoint actuation will be set at 440 psig to provide buffer from the Technical Specifications value.

TABLE 4.1-4

STEAM GENERATOR DESIGN DATA

Number of Steam Generators	3
Design Pressure, Reactor Coolant/Steam, psig	2485/1085
Reactor Coolant Hydrostatic Test pressure (tube side-cold), psig	3107
Design Temperature, Reactor Coolant/Steam, F	650/556
Reactor Coolant Flow, lb/hr, each	33.83×10^6
Total Heat Transfer Surface Area, ft.^{2}, each	43,467
Steam Conditions at Full Load, Outlet Nozzle:	
Steam Flow, lb/hr, each	3.196×10^6
Steam Temperature, F	516
Steam Pressure, psig	770
Feedwater Temperature, F	436.5
Overall Height, ft.-in.	63-1.6
Shell OD, upper/lower, in.	166/127.5
Shell Thickness, upper/lower, in.	3.5/2.63
Number of U-tubes	3214
U-tube Diameter, in.	0.875
Tube Wall Thickness, (average), in.	0.050
Number of Manways/ID in.	3/16
Number of handholes/ID, in.	6/6

	2200 MWt	Zero Power
Primary Side Fluid Volume, ft.^{3}	935	935
Primary Side Fluid Heat Content, BTU	24.31×10^6	23.7×10^6
Secondary Side Fluid Volume, ft.^{3}	4596	4596
Secondary Side Fluid Mass, lbs.	80,300	134,000

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* The above Steam Generator design data has not been revised as part of the Steam Generator Repair Project or Thermal Uprate Project and should be regarded as historical reference only. Refer to FSAR Table 4C-1 for updated design data resulting from the Thermal Uprate Project.

TABLE 4.1-5

REACTOR COOLANT PUMPS DESIGN DATA

Number of Pumps	3
Design Pressure/Operating Pressure, psig	2485/2235
Hydrostatic Test Pressure (cold), psig	3107
Design Temperature (casing), ^{o}F	650
RPM at Nameplate Rating	1188
Suction Temperature, ^{o}F	548.9
Net Positive Suction Head, ft. (required)	168
Developed Head, ft.	266
Capacity, gpm	88,500
Seal Water Injection, gpm	7.5
Seal Water Return, normal, gpm	2.5
Pump Discharge Nozzle, ID, in.	27-1/2
Pump Suction Nozzle ID, in.	31
Overall Unit Height, ft.	28.242
Water Volume, $ft.^3$	192
Pump-Motor Moment of Inertia, $lb\text{-}ft^2$	70,000
Motor Data:	
Type	AC Induction Single Speed, Air Cooled
Voltage	4000
Insulation Class	B Thermalastic Epoxy
Phase	3
Frequency, Hz	60
Starting Current, maximum, amp	4800
Input (hot reactor coolant), kw	4360
Input (cold reactor coolant), kw	5674
Power, HP (nameplate)	6000

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TABLE 4.1-6

REACTOR COOLANT PIPING DESIGN DATA

Design/Operating Pressure, psig	2485/2235
Hydrostatic Test Pressure, (cold) psig	3107
Design Temperature, °F	650
Design Temperature, (pressurizer surge line), °F	680
Reactor Inlet Piping, ID, in.	27-1/2
Reactor Inlet Piping, nominal thickness, in.	2.375
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.	2.625
Pressurizer Surge Line Piping, ID, in./Pipe Schedule	12/Sch 140
Pressurizer Surge Line Piping, nominal thickness, in.	1.125
Water Volume, (all 3 loops) ft^3	783

* Surge line fitted with a 14"/12" adapter at the pressurizer

TABLE 4.1-7

REACTOR COOLANT SYSTEM DESIGN PRESSURE DROP

	Pressure Drop, psi (estimated)
Across Pump Discharge Leg	1.3 / 1.2
Across Vessel, including nozzles	40.8 / 38.4
Across Hot Leg	1.2 / 1.1
Across Steam Generator	33.9 / 38.0
Across Pump Suction Leg	3.0 / 2.8
Total Pressure Drop	80.2 / 81.5

C26

NOTE: The first value provided coincides with the maximum Best Estimate Flow (minimum steam generator tube plugging, minimum reactor vessel average temperature) and the second value coincides with the minimum Best Estimate Flow (maximum steam generator tube plugging, maximum reactor vessel average temperature).

C26

TABLE 4.1-8

DESIGN THERMAL AND LOADING CYCLES - 80 YEARS

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	Transient Design Condition	Design Cycles
1.	Station heatup at 100°F per hour	200
2.	Station cooldown at 100°F per hour	200
3.	Station loading at 5% of full power/min	14,500[5] [6]
4.	Station unloading at 5% of full power/min	14,500[5] [6]
5.	Step load increase of 10% of full power (but not to exceed full power)	2000[5] [6]
6.	Step load decrease of 10% of full power	2000[5] [6]
7.	Step load decrease of 50% of full power	200[5] [6]
8.	Reactor trip	400[5] [6]
9.	Hydrostatic test at 3107 psig pressure, 100°F temperature	1[3] [5] [6]
10.	Hydrostatic test at 2485 psig pressure and 400°F temperature	5[4] [5] [6]
11.	Steady state fluctuations	00[1]
12.	Feedwater Cycling at Hot Standby	2000[2]

Notes:

(1) Not counted, not significant contributor to fatigue usage factor.

(2) Not counted, Intermittent slug feeding at hot standby not performed.

(3) Limited by Steam Generator Analysis. Represents pre-operational hydrostatic test.

(4) Limited by Reactor Coolant Pump Analysis.

(5) Transients 3 through 10 design cycle limit for Unit 3 baffle-former bolts only is being lowered due to EPU RCS conditions. Station loading and unloading from 14,500 to 408; Step load increase 10% from 2,000 to 73; Step load decrease 10% from 2,000 to 120; Step load decrease 50% from 200 to 142; Reactor trip from 400 to 320; Hydrostatic test from 6 to 2.

(6) Transients 3 through 10 design cycle limit for Unit 4 baffle-former bolts only is being lowered due to EPU RCS conditions. Station loading and unloading from 14,500 to 599; Step load increase 10% from 2,000 to 70; Step load decrease 10% from 2,000 to 77; Step load decrease 50% from 200 to 67; Reactor trip from 400 to 272; Hydrostatic test from 6 to 2.

TABLE 4.1-9 Sheet 1 of 2

REACTOR COOLANT SYSTEM - CODE REQUIREMENTS

Component	Codes
Reactor Vessel(Note 1, Note 4, Note 5)	ASME III* Class A
Control Rod Drive Mechanism Housings (Note 2)	ASME III* Class A
Steam Generator	
Tube Side	ASME III* Class A
Shell Side ***	ASME III* Class C
Reactor Coolant Pump Casing	No Code (Design per ASME III-Article 4)
Pressurizer	ASME III* Class A
Pressurizer Relief Tank (Note 7)	ASME III* Class C
Pressurizer Safety Valves	ASME III*
Reactor Coolant Piping	ASA B31.1**
System valves, fittings, piping and tubing (Note 5, Note 6)	ASA B31.1**
Core Exit Thermocouple Seal Assemblies (Note 3) (Head Port Adapters, Drive Sleeves	ASME III* Subsection NB, Class 1, 1986 Edition

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

** ASA B31.1-1955 Code for Pressure Piping, plus Code Cases N-7 and N-10 where applicable.

*** 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.

Notes:

1. The Reactor Vessel Closure Head (RVCH) for Unit 3 and 4 has been replaced (Reference 6 and 7).The replacement RVCH design Code is ASME Boiler and Pressure Vessel Code, Section III, Subsection NB, Class 1, 1989 Edition, no Addenda.

2. The Control Rod Drive Mechanism (CRDM) pressure housings for Unit 3 and Unit 4 have been replaced (Reference 6 and 7). The replacement CRDM pressure housing design Code is ASME Boiler and Pressure Vessel Code, Section III, Subsection NB, Class 1, 1989 Edition, no Addenda.

3. The Core Exit Thermocouple Nozzle Adapter (CETNA) and Reactor Vessel Level Monitoring System (RVLMS) Nozzle Adapter for Unit 3 and Unit 4 have been replaced (Reference 6 and 7). The replacement components' design Code is ASME Boiler and Pressure Vessel Code, Section III, Subsection NB, Class 1, 1989 Edition, no Addenda.

4. Per Reference 6 and Reference 7 - The spare CRDM housing adapters are closed with a CRDM plug. The CRDM Plug design Code is ASME Boiler and Pressure Vessel Code, Section III, Subsection NB, Class 1, 1989 Edition, no Addenda.

REACTOR COOLANT SYSTEM - CODE REQUIREMENTS

Notes (cont'd)

5. The design Code for the Bottom Mounted Instrumentation (BMI) Tubing is the ASME Boiler and Pressure Vessel Code, Section III, Subsection NB, 1989 edition, no addenda. The design Code for the BMI Supports is the ASME Boiler and Pressure Vessel Code, Section III, Subsection NF, 1998 edition, no addenda.

6. With the exception of the thermal stratification analysis of the pressurizer surge line which uses fatigue stress limits from the 1986 edition of the ASME B & PV Code, Section III, Subsection NB.

7. The Unit 3 Pressurizer Relief Tank has not been maintained as a ASME III vessel in service. C30

TABLE 4.1-10

COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	CYCLIC OR TRANSIENT LIMIT[1]	DESIGN CYCLE OR TRANSIENT
Reactor Coolant System	200 heatup cycles at ≤ 100°F/h and 200 cooldown cycles at ≤ 100°F/h.	Heatup cycle - T_{avg} from ≤ 200°F to ≥ 550°F. Cooldown cycle - T_{avg} from ≥ 550°F to ≤ 200°F.
	200 pressurizer cooldown cycles at ≤ 200°F/h from nominal pressure.	Pressurizer cooldown cycle temperatures from ≥ 650°F to ≤ 200°F.
	200 pressurizer cooldown cycles at ≤ 200°F/h from 400 psia.	Pressurizer cooldown cycle temperatures from ≥ 650°F to ≤ 200°F.
	80 loss of load cycles, without immediate Turbine or Reactor trip.	≥ 15% of RATED THERMAL POWER to 0% of RATED THERMAL POWER.
	40 cycles of loss-of-offsite A.C electrical power.	Loss-of-offsite A.C electrical ESF Electrical System.
	80 cycles of loss of flow in one reactor coolant loop.	Loss of only one reactor coolant pump.
	400 Reactor trip cycles.	100% to 0% of RATED THERMAL POWER.
	10cycles of inadvertent auxiliary spray.	Spray water temperature differential to 560°F.
	150 primary to secondary side leak tests.	Pressurized to 2435 psig.
	15 primary to secondary side leak tests.	Pressurized to 2250 psig.
	5 hydrostatic pressure tests.	Pressurized to 2485 psig and 400°F.
Secondary Coolant System	50 hydrostatic pressure tests.	Pressurized to 1085 psig.
	10 hydrostatic pressure tests.	Pressurized to 1356 psig.
	15 secondary to primary side leak tests.	Pressurized to 840 psig.

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(1) See Table 4.1-8 for design cycle limits on baffle-former bolts only.

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4.2 SYSTEM DESIGN AND OPERATION

4.2.1 GENERAL DESCRIPTION

The Reactor Coolant Systems of the two nuclear power units are essentially identical and do not share any components. The following description applies to either unit.* Each Reactor Coolant System consists of three similar heat transfer loops connected in parallel to the reactor vessel. Each loop contains a steam generator, a pump, loop piping, and instrumentation. The pressurizer surge line is connected to one of the loops. Auxiliary system piping connections into the reactor coolant piping are provided as necessary. A flow diagram of the system is shown on Figures 4.2-1 and 4.2-9 through 4.2-13 and of the reactor coolant pump on Figures 4.2-10 and 4.2-14.

The containment boundary shown on the flow diagram indicates those major components which are to be located inside the containment. The intersection of a process line with this boundary indicates a containment penetration.

Reactor Coolant System and components design data are listed in Table 4.1-1 through 4.1-6.

Pressure in the system is controlled by the pressurizer, where water and steam pressure is maintained 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 Section 7. Spring-loaded code steam 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.

* See Appendix 4C for description of replaced steam generator lower assemblies.

4.2.2 COMPONENTS

Reactor Vessel

The reactor vessel is cylindrical in shape with a hemispherical bottom and a flanged and gasketed removable upper head. Figure 4.2-2 is a schematic of the reactor vessel. The materials of construction of the reactor vessel are given in Table 4.2-1. The upper, intermediate and lower shell courses, and the lower head ring, are cylindrical machined forgings.

The Reactor Vessel Closure Heads (RVCH) for Unit 3 and Unit 4 have been replaced with a heads manufactured from a mono-block forging instead of the forged flange and dished plate weldment head. This design eliminated the circumferential weld that welded the head dome plate to the forged flange section. Figure 4.2-2 Parts 3 and 4 is a schematic of the replacement head for the Unit 3 & 4 Reactor Vessel.

Coolant enters the reactor vessel through inlet nozzles in a plane just below the vessel flange and above the core. The coolant flows downward through the annular space between the vessel wall and the core barrel into a plenum at the bottom of the vessel where it reverses direction. Approximately ninety-five per cent of the total coolant flow is effective for heat removal from the core. The remainder of the flow includes the flow through the RCC guide thimbles, the leakage across the fuel assembly outlet nozzles, and the flow deflected into the head of the vessel for cooling the upper flange. All the coolant is united and mixed in the upper plenum, and the mixed coolant stream then flows out of the vessel through exit nozzles located on the same plane as the inlet nozzles.

A one-piece thermal shield, concentric with the reactor core, is located between the core barrel and the reactor vessel. The shield is bolted and welded to the top of the core barrel. The shield, which is cooled by the coolant on its downward pass, protects the vessel by attenuating much of the gamma radiation and some of the fast neutrons which escape from the core. This shield minimizes thermal stresses in the vessel which result from heat generated by the absorption of gamma energy. It is illustrated in Figure 3.2.3 and is further described in Section 3.2.3.

Fifty core instrumentation nozzles are located on the lower head.

The reactor closure head and the reactor vessel flange are joined by 58 - 6 in. diameter studs. Two metallic O-rings seal the reactor vessel when the reactor closure head is bolted in place. A leakoff connection is provided between the two O-rings to monitor leakage across the inner O-ring. In addition, a leak-off connection is also provided beyond the outer O-ring seal.

The reactor vessel insulation is of the reflective type, supported from the nozzles and consisting of inner and outer sheets of stainless steel spaced 3 inches apart and with multilayer aluminum foil. For Unit 3, The removable vessel flange is of similar construction, while stainless steel foil filler is used for the Unit 4 Reactor Vessel Flange insulation. The clearance to the reactor vessel is 1/2 inch. The insulation provided for the reactor closure flange is supported on the refueling seal ledge and vent shroud support rings.

The Reactor Vessel Head Permanent insulation (i.e., within the IHA) for Unit 3 & 4 consists of self supporting panels, constructed of metallic reflective insulation, that are attached to one another with stainless steel buckles. This configuration ensures clearance between the reactor vessel head and the bottom of the insulation. The vertical portions of this permanent insulation have removable portions to allow access to the reactor vessel head and CRDM nozzles.

The reactor vessel contains the core support assembly, upper plenum assembly, fuel assemblies, control rod cluster assemblies, surveillance specimens, and in-core instrumentation. The reactor vessel internals are designed to direct the coolant flow, support the reactor core, and guide the control rods in the withdrawn position.

The reactor internals are described in detail in Section 3.2.3 and the general arrangement of the reactor vessel and internals is shown in Figure 3.2.3-2.

Reactor vessel design data are listed in Table 4.1-2.

Reactor Vessel Support Structure

The reactor vessel support structure is described in Section 5 and shown in Figure 5.1-20.

Pressurizer

The general arrangement of the pressurizer is shown in Figure 4.2-3, and the design data are listed in Table 4.1-3.

The pressurizer maintains the required reactor coolant pressure during steady-state operation, limits the pressure changes caused by coolant thermal expansion and contraction during normal load transients, and prevents the pressure in the Reactor Coolant System from exceeding the design pressure.

The pressurizer contains replaceable direct immersion heaters, multiple safety and relief valves, a spray nozzle and interconnecting piping, valves and instrumentation. The electric heaters located in the lower section of the vessel maintain the pressure of the Reactor Coolant System by keeping the water and steam in the pressurizer at system saturation temperature. The heaters are capable of raising the temperature of the pressurizer and contents at approximately 55°F/hr during startup of the reactor.

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 a positive surge, caused by a decrease in unit load, the spray system, which is fed from the cold leg of a coolant loop, condenses steam in the vessel to prevent the pressurizer pressure from reaching the set point of the power operated relief valves. Power operated spray valves on the pressurizer limit the pressure during load transients. In addition the spray valves can be operated manually from the control room. A small continuous spray flow is provided to assure that the pressurizer liquid is homogeneous with the coolant and to prevent excess cooling of the spray and surge line piping.

During a negative pressure surge, caused by an increase in unit load, flashing of water to steam and generation 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 positive surges to heat the subcooled surge water entering the pressurizer from the reactor coolant loop.

The pressurizer is constructed of carbon steel with internal surfaces clad with austenitic stainless steel. The heaters are sheathed in austenitic stainless steel. Unit 3 pressurizer #11 heater well nozzle has been modified with a half nozzle design welded to the outside of the pressurizer shell instead of the internal cladding. This change was submitted under an ASME Section XI Relief Request to the NRC and approved (Ref. ADAMS Access1on No.: ML 15271A325). C28

The pressurizer vessel surge nozzle is protected from thermal shock by a thermal sleeve. A thermal sleeve also protects the pressurizer spray nozzle connection.

<u>Steam Generators</u>

Each loop contains a vertical shell and U-tube steam generator.* A steam generator of this type is shown in Figure 4.2-4. Principal design parameters are listed in Table 4.1-4.

Reactor coolant enters the inlet side of the channel head at the bottom of the steam generator through the inlet nozzle, flows through the U-tubes to an outlet channel and leaves the generator through another bottom nozzle. The inlet and outlet channels are separated by a partition. Manways are provided to permit access to the U-tubes and moisture separating equipment. The general procedural activities for plugging a defective steam generator tube is outlined in Appendix 4B. Defective steam generator tubes having indications may require corrective maintenance actions such as plugging or plugging and staking. Design drawings for plugs and stakes and related procedures shall be approved in accordance with plant administrative procedures.

Feedwater to the steam generator enters just above the top of the U-tubes through a feedwater ring. The water flows downward through an annulus between the tube wrapper and the shell and then upward through the tube bundle where part of it is converted to steam.

* See Appendix 4C for description of replaced steam generator lower assemblies.

The steam-water mixture from the tube bundle passes through a steam swirl vane assembly which imparts a centrifugal motion to the mixture and separates the water particles from the steam. The water spills over the edge of the swirl vane housing and combines with the feedwater for another pass through the tube bundle.

The steam rises through additional separators which limit the moisture content of the steam to one fourth of one per cent or less under all design load conditions.

The steam generator is constructed primarily of carbon steel. The heat transfer tubes are Inconel. The interior surfaces of the channel heads and nozzles are clad with austenitic stainless steel, and the side of the tube sheet in contact with the reactor coolant is clad with Inconel. The tube to tube sheet joint is welded.

Steam Generator Support Structure

The steam generator support structures are described in Section 5 and shown in Figure 5.1-20.

Reactor Coolant Pumps

Each reactor coolant loop contains a vertical single stage centrifugal pump which employs a controlled leakage seal assembly. A view of a controlled leakage pump is shown in Figure 4.2-5 and the principal design parameters for the pumps are listed in Table 4.1-5. The reactor coolant pump estimated performance and NPSH characteristics are shown in Figure 4.2-6. The performance characteristic is common to all of the higher specific speed centrifugal pumps, and the 'knee' at about 40% design flow introduces no operational restrictions, since the pumps operate at full speed.

Reactor coolant is pumped by the impeller attached to the bottom of the rotor shaft. The coolant is drawn up through the impeller, discharged through passages in the diffuser and out through a discharge nozzle in the side of the casing. The motor-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. The seal package is a multi-seal cartridge containing three identical stages in series. Only one stage is required to function to prevent excessive leakage from the Reactor Coolant System. Seal staging flow, designated Control Bleed Off, exits the RCP through lines to the Chemical and Volume Control seal return line that goes to the volume control tank. The flow that passes across the upper seal stage, designated Seal Leak off, is routed to the reactor coolant drain tank to minimize leakage of water and vapor into the containment atmosphere. To mitigate the effects caused by failure of all three stages, the seal package includes a Abeyance (shutdown) seal. The flow rates that result from various failure modes of the three stages vary and could result in flows that are not high enough to activate the abeyance seal. With the RCP tripped, the abeyance seal is designed to stop leakage from upper, third stage seal for an indefinite period and RCP seal leakage is limited to flow through the Control Bleed Off seal return line C28

A portion of the high pressure water flow from the charging pumps is injected into the reactor coolant pump between the thermal barrier (above the RCP impeller) and the controlled leakage seal in the lower pump shaft housing 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 upper (third stage) seal is also collected and removed from the pump. C28

The RCP seals are provided with redundant means of cooling, seal injection via the charging system and thermal barrier cooling via the Component Cooling Water (CCW) system. When both systems are operating, either is sufficient to provide adequate seal cooling for up to 24 hours.

For an event which occurs causing a safety injection signal with off site power available (i.e., RCPs remain running), thermal barrier cooling will continue so long as the High-High containment pressure signal setpoint of 20 PSIG is not reached. Seal injection will be lost on an S.I. signal (charging pumps tripped on S.I.). On High-High containment pressure, Phase B containment isolation is initiated and thermal barrier cooling will be automatically isolated. If seal injection is re-established, operating procedures permit continued RCP operation until upper or lower motor bearing temperatures reach 195°F or RCS subcooling is lost. Upon reaching 195°F the RCPs are manually stopped. If neither CCW or Seal Injection are available, the RCPs will be tripped.

The RCPs must be shutdown when the RCP motor bearing temperature reaches 195°F. The RCP seals will not leak excessively even if cooling water is lost for an extended period of time after the RCPs are tripped Flowserve has conducted a series of tests to explore the ability of the N-Seal RCP package to withstand pump running operation under a Loss of Seal Cooling (LOSC) situation. Additionally, a model for failure of the three stage N-seal package with an Abeyance (shutdown) seal that addresses the probability of occurrence of activation conditions for the shutdown seal has been developed and testing evaluates the ability of the N-Seal package to withstand LOSC during pump shutdown conditions. Failure modes that result in flow rates lower than required to activate the abeyance seal will not actuate the seal. The RCP seal package can remain intact for 20 minutes with the RCS at full temperature and pressure while the pump is operating in line with the criteria established in WCAP-16175. The Abeyance seal provides assurance that after degradation to the mechanical seal package occurs with the RCPs stopped, the coping time is extended to at least 96 hours under the most extreme operating parameters. The N-Seal design has been installed and operated at plants such as Surry, Oconee, and Crystal River 3 (Reference PRA Model for Flowserve 3 Stage N-Seals with Abeyance Seal, Revision 0, dated 12/20/2013). C31

The squirrel cage induction motor driving the pump is air cooled and has oil lubricated thrust and radial bearings. The lube oil leakage drain arrangement for protection against fire at the reactor coolant pump motor discussed in Engineering Guidelines for Fire Protection for Turkey Point Units 3 & 4 (Reference 14). A water lubricated bearing provides radial support for the pump shaft. Component cooling water is supplied to the motor bearing cooler and the thermal barrier cooling coil.

A reactor coolant pump motor vibration monitor system is provided to assist in balancing the pump-motor combination and to provide alarm and recording level capability. Two shaft vibration proximity probes are mounted 90° apart and a third probe develop the key phasor. Two velocity probes are mounted on the motor.

The Turkey Point reactor coolant pump casings were electroslag welded. The following efforts were performed for quality assurance of the components.

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 2 wire and the 3 wire techniques, respectfully. They are:

    A. Two wire electroslag process - 8" thick weldment.
        1. 6 Transverse Tensile Bars - 750°F post weld stress relief
        2. 12 Guided Side Bend Test Bars

    B. Three wire electroslag process - 12" thick weldment
        1. 6 Transverse Tensile Bars - 750°F post weld stress relief
        2. 17 Guided Side Bend Test Bars
        3. 21 Charpy Vee Notch Specimens
        4. Full section macroexamination of weld and heat affected zone.
        5. Numerous microscopic examinations of specimens removed from the weld and heat affected zone regions.
        6. Hardness survey across weld and heat affected zone.

    C. A separate weld test was made using the 2 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 8 inch thick weldment in the stop-restart-repaired region and successfully tested. They are:

        1. 2 Transverse Tensile Bars - as welded
        2. 4 Guided Side Bend Test Bars
        3. Full section macroexamination of weld and heat affected zone.

D. All of the weld test blocks in (A), (B) and (C) above were radiographed using a 24 Mev Betatron. The radiographic quality level (as defined by ASTM E-94) obtained was between one-half of 1% to 1%. There were no discontinuities evident in any of the electroslag welds.

1. 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.

2. 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.

3. 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.

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

5. Heat treatment furnace charts were recorded and certified.

Pump Support Structure

The reactor coolant pump support structures are described in Section 5 and shown in Figure 5.1-20.

RCP Trip Criteria

RCS pressure, RCS subcooling and secondary pressure dependent RCS pressure (RCS - $P_{Steam\ Line}$) are the most appropriate in providing pump trip discrimination between Loss of Coolant Accident (LOCA) and Steam Generator Tube Rupture (SGTR) or non-LOCA events.

The RCS pressure does not meet the acceptance criteria for discrimination between LOCA and SGTR or non-LOCA events since the secondary pressure dependent RCS pressure trip parameter requires the reactor operator to look at two instruments (RCS pressure and Steam Generator pressure).

The RCS subcooling is the desired parameter for pump trip since it only requires the operator to look at one instrument (subcooled margin monitor). The Subcooled Margin Monitor (SMM), as described in Section 4.2.10, is a fully redundant, qualified system as required by TMI Action Item II.F.2, "Instrumentation for Detection of Inadequate Core Cooling". The SMM takes input from redundant hot leg pressure transmitters and temperature elements. These inputs are then used in a calculational program that determines RCS subcooling which is then displayed in the control room.

It is desirable to keep the RCPs running during a Steam Generator Tube Rupture (SGTR) and other non LOCAs to 1) maintain normal pressure control using pressurizer spray and thereby avoiding opening of the pressurizer PORVs, 2) prevent the formation of a stagnant water volume in the upper head region which may flash and form a steam bubble during subsequent cooldown and depressurization, 3) minimize potential pressurized thermal shock challenges and 4) minimize operator action such as tripping the RCPs and then restarting them later.

The LOFTRAN computer code was used to perform the alternate RCP trip criteria analyses. Both Steam Generator Tube Rupture (SGTR) and non-LOCA event were simulated in these analyses. Results for the SGTR analyses were used to obtain all but three of the trip parameters. LOFTRAN is a Westinghouse licensed code used for FSAR SGTR and non-LOCA analyses.

The following are considered to have the most impact on the determination of the RCP trip criteria:

1. Break flow
2. SI flow
3. Decay heat
4. Auxiliary feedwater flow

The effects of all these uncertainties with the models and input parameters were evaluated and it was concluded that the contributions from the break flow conservatism and the SI uncertainty dominate. The calculated overall uncertainty in the WOG analyses as a result of these considerations for the Turkey Point units is +1 to +5°F for the RCS subcooling RCP trip setpoint. Due to the minimal effects from the decay heat model and AFW input, these results include only the effects of the uncertainties due to the break flow model and SI flow inputs.

Manual Trip

Manual trip of an RCP motor requires the availability of 125V DC power, the motor control switch, and the motor breaker. This provides a reliable means of tripping the RCP. With the exception of the motor and cabling, all the components associated with the RCP motor are outside containment. Therefore, adverse environmental conditions will not prevent RCP trip when required.

Instrumentation Uncertainties for use of the RCS Subcooling for both Normal and Adverse Containment Conditions

The minimum RCS pressure for SGTRs and non-LOCAs is approximately 1135 psig. The subcooling uncertainty for normal containment conditions at this pressure is 22.3°F. At Residual Heat Removal (RHR) system pressure, 450 psig, the subcooling uncertainty is 25.5°F. Therefore, the trip setpoint for the RCPs under normal containment conditions is less than 25.5°F subcooling.

Under the adverse containment conditions, the RCP trip setpoint at the non-LOCA SGTR lower pressure limit of 1135 psig was determined to be subcooling 65°F.

The instrument uncertainties consider uncertainties from the transmitter or temperature sensor, through the electronics to the display itself. While the temperature sensors associated with the SMM are not sensitive to containment conditions, the RCS pressure transmitters exhibit higher uncertainty under adverse containment conditions. The permissive for using the adverse containment setpoint is either 180°F containment temperature or 1.3×10^5 R/hr.

The design of the Subcooled Margin Monitor (SMM) software is such that invalid or failed instrument inputs, such as might be caused by pipe whip, are not used. The arrangement of instrumentation precludes failure of the SMM due to pipe whip or single failure.

Pressurizer Relief Tank

Principal design parameters of the pressurizer relief tank are given in Table 4.1-3.

Steam discharged from the power relief and safety valves passes to the pressurizer relief tank which is partially filled with water at or near containment ambient conditions. The tank normally contains water in a predominantly nitrogen atmosphere. Steam is discharged under the water level to condense and cool by mixing with the water. The tank is equipped with a spray, and a drain to the Waste Disposal System, which are operated to cool the tank following a discharge.

The tank size is based on the requirement to condense and cool a discharge of pressurizer steam equivalent to 110 percent of the volume above the full power water level setpoint.

The tank is protected against a discharge exceeding the design value by two rupture discs which discharge into the reactor containment. The two rupture discs on the pressurizer relief tank have the combined relief capacity to relieve the total capacity of the three pressurizer safety valves. The tank design pressure (and the rupture disc setting) is twice the calculated pressure resulting from the maximum safety valve discharge described above. This margin is to prevent deformation of the disc. The tank and rupture disc holder are also designed for full vacuum to prevent tank collapse if the tank contents cool without nitrogen being added.

The discharge piping from the safety and relief valves to the relief tank is sufficiently large to prevent backpressure at the safety valves from exceeding 20 per cent of the set point pressure at full flow.

The pressurizer relief tank, by means of its connection to the Waste Disposal System, provides a means for removing any non-condensable gases from the Reactor Coolant System which might collect in the pressurizer vessel.

The tank is constructed of carbon steel and as supplied included a corrosion resistant coating on the internal surface. C30

Piping

The general arrangement of the reactor coolant system piping is shown on the layout drawings in Section 1. Piping design data are presented in Table 4.1-6.

The austenitic stainless steel reactor coolant piping and fittings which make up the loops are 29 in. ID in the hot legs, 27-1/2 in. ID in the cold legs and 31 in. ID between each loop's steam generator outlet and its reactor coolant pump suction. The pressurizer relief line, which connects the pressurizer safety and relief valves' outlets to the inlet nozzle flange on the pressurizer relief tank, is constructed of carbon steel.

Smaller piping, including the pressurizer surge and spray lines, drains and connections to other systems are austenitic stainless steel. All joints and connections are welded except for stainless steel flange connections to the carbon steel pressurizer relief tank and the connections at the relief and safety valves.

The Turkey Point 90° elbows were 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 5 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° 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 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.

Thermal sleeves are installed at the following locations where high thermal stresses could otherwise develop due to rapid changes in fluid temperature during normal operational transients:

a) Return line from the residual heat removal loop.
b) Both ends of the pressurizer surge line.
c) Pressurizer spray line connection to the pressurizer.
d) Charging lines and auxiliary charging line connections.

Valves

All valve surfaces in contact with reactor coolant are austenitic stainless steel or equivalent corrosion resistant materials. Connections to stainless steel piping are welded. Valves that perform a modulating function may be equipped with two sets of packing and an intermediate leakoff connection.

4.2.3 PRESSURE-RELIEVING DEVICES

The Reactor Coolant System is protected against overpressure by control and protective circuits such as the high pressure trip and by code relief valves connected to the top head of the pressurizer. The relief 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 4.2-1, and the valve design parameters are given in Table 4.1-3. Valve sizes are determined as indicated in Section 4.3.4. Power-operated relief valves and code safety valves are provided to protect against pressure surges which are beyond the pressure limiting capacity of the pressurizer spray. Each pressurizer safety valve has an acoustic accelerometer mounted on the discharge of the valve to provide the control room operator with positive indication of the pressurizer safety valve position.

The pressurizer relief tank is protected against a steam discharge exceeding the design pressure value by two rupture discs which discharges into the reactor containment. The rupture disc relief conditions are given in Table 4.1-3.

4.2.4 PROTECTION AGAINST PROLIFERATION OF DYNAMIC EFFECTS

Engineered Safety Features and associated systems are protected from loss of function due to dynamic effects and missiles which might result from a loss-of-coolant accident. Protection is provided by missile shielding and/or segregation of redundant components. This is discussed in Section 6.1.

The Reactor Coolant System is surrounded by concrete shield walls. These walls provide shielding to permit access into the containment during full power operation for inspection and maintenance of miscellaneous equipment. These shielding walls also provide missile protection for the containment liner plate.

The concrete deck over the Reactor Coolant System also provides for shielding and missile damage protection.

Steam generator lateral bracing is provided near the upper tube support elevation to resist lateral loads, including those resulting from seismic forces and pipe rupture forces. Additional bracing is provided at a lower elevation to resist pipe rupture loads.

The NRC documents in their letter of November 28, 1988 (Reference 3) that the leakage detection systems at Turkey Point Units 3 and 4 satisfy the requirements of Generic Letter 84-04, and that the primary loop piping complies with the criteria of GDC 4 from 10 CFR 50, Appendix A. GDC 4 allows the use of plant-specific Leak-Before-Break analysis to eliminate the dynamic effects of postulated pipe ruptures in high energy piping from the design basis of a plant. Plants with an NRC-approved Leak-Before-Break analysis may remove pipe whip restraints and jet impingement barriers. Turkey Point Units 3 and 4 received NRC approval (Reference 4) for elimination of the dynamic effects of postulated pipe ruptures in reactor coolant piping from the design basis of the plant. The Turkey Point analysis for the Leak-Before-Break Methodology is documented in the Westinghouse report WCAP-14237 (Reference 5). Therefore, the dynamic loads associated with a rupture of the reactor coolant piping need not be considered in the design of the reactor support structures.

Missile protection afforded by the arrangement of the Reactor Coolant System is illustrated in the containment structure drawings which are given in Section 5.

4.2.5 MATERIALS OF CONSTRUCTION

Each of the materials used in the Reactor Coolant System is selected for the expected environment and service conditions. The major component materials are listed in Table 4.2-1. Those pressure-containing or strength-bearing stainless steel component parts in the reactor vessel and associated reactor coolant systems that have become furnace sensitized during the fabrication sequence are listed in Table 4.2-4.

All reactor coolant system 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. During mode 1, the chemical composition of the reactor coolant is maintained within the specification given in Table 4.2.2. Reactor coolant chemistry is further discussed in Section 4.2.8.

In Mode 1 the water in the secondary side of the steam generators is normally maintained within the chemistry parameters given in Table 4.2-3 to control deposits and corrosion inside the steam generators. Specific operating chemistry specifications and limits are maintained as outlined in the Nuclear Chemistry Parameters Manual and the Turkey Point Chemistry Procedures.

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

It is characteristic of stress corrosion 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 a concentrating mechanism and the presence of chlorides 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, 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.

Considerable experience with Inconel in steam generator and heat exchanger applications has been accumulated in the industry. Since 1962, widespread adoption of Inconel for steam generator tubes in nuclear stations is evident: as for example, Connecticut-Yankee; San Onofre; PM-1, Sundance; PM-3A, McMurdo Sound; CVTR; NPD, and Hanford N-Reactor. In none of these plants has there been any evidence of steam generator tube leakage. Materials with lead traces in the overall composition were present in the secondary side of the referenced plants. The use of lead in the materials of the secondary side of this plant has been minimized to the practical limit of that occurring as trace elements in metallurgical alloys and, as such, is insignificant.

All external insulation of Reactor Coolant System components is compatible with the component materials. The cylindrical shell exterior and closure flanges to the reactor vessel are insulated with metallic reflective insulation. The Unit 3 & 4 closure heads are insulated with a self supporting panel insulation arrangements that are constructed of a metallic reflective material and woven fiberglass blanket. All other external corrosion-resistant surfaces in the Reactor Coolant System are insulated with low halide or halide-free insulating material as required.

The Nil Ductility Transition Temperature (NDTT) of the reactor vessel plate or C25
forging material opposite the core is established at a Charpy V-notch test value of 30 ft-1b or greater. The material is tested to verify conformity to specified requirements and to determine the actual NDTT value. In addition, this plate was initially 100 per cent volumetrically inspected by ultrasonic test using both longitudinal and shear wave methods.

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

The reactor vessel material is heat-treated specifically to obtain good notch-ductility which ensures a low NDTT, and thereby gives assurance 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. During fabrication initial maximum values of NDTT have been established at 40°F for No. 3 vessel and 50°F for No. 4 vessel forgings.

Framatome ANP Topical Report BAW-2308, Revisions 1A and 2A (References 10 and 11) provide new initial weld materials properties. The NRC approved the exemption request to use these values in a letter dated March 11, 2010 (Reference 12).

The techniques used to measure and predict the integrated fast neutron (E > 1 Mev) fluxes at the sample locations are described in Appendix 4A. The calculation method used to obtain the maximum neutron (E > 1 Mev) exposure of the reactor vessel is identical to that described for the irradiation samples. Since the neutron spectra at the sample can be applied with confidence to the adjacent section of reactor vessel, the maximum vessel exposure will be obtained from the measured sample exposure by appropriate application of the calculated azimuthal neutron flux variation.

At uprated conditions, the maximum integrated fast neutron (E>1 Mev) exposure
of the vessel was computed to be 1.08×10^{20} n/cm^2 at the end of the extended
license terms of 72 EFPY*, approximately (Reference 6). Under the same C31
conditions, the maximum vessel exposure at the limiting circumferential vessel
weld is predicted to be 9.86×10^{19} n/cm^2 at the end of the extended license
terms of 72 EFPY*, approximately (Reference 6)**. The predicted extended end C31
of life RT(ndt) is less than the 10CFR50.61 screening criteria (Reference 6).

To evaluate the RT(ndt) 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 described in Section 4A.

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* This value is approximate and will change from year to year based on the unit availability. Fluence prediction is acceptable in the ±20% range, so this value can easily vary within that limit.

** After the (hafnium) pressurized thermal shock absorbers were removed from
the vessel cores in 2009, the maximum vessel exposure at the limiting
circumferential weld is predicted to be 9.86×10^{19} n/cm^2 at the end of the
extended license terms of approximately 72 EFPY (Reference 6). The NRC was C31
notified of this proposed change as captured in Reference 9. The NRC
approved the changes as documented in Reference 12.

The methods used to measure the initial NDTT of the reactor vessel base plate material are given in Appendix 4A.

4.2.6 MAXIMUM HEATING AND COOLING RATES

The reactor system operating cycles used for design purposes are given in Table 4.1-8 and described in Section 4.1.5. During unit heatup and cooldown, the rates of temperature and pressure changes are limited. The system design heatup and cooldown rate of 100°F per hour satisfies stress limits for cyclic operation (ASME B&PV Code, Section III) and is consistent with the expected number of cycles. However, the normal system heatup and cooldown rate is administratively limited to less than or equal to 90°F per hour. Sufficient electrical heaters are installed in the pressurizer to permit a heatup rate, starting with a minimum water level, of 55°F per hour. This rate takes into account the small continuous spray flow provided to maintain the pressurizer liquid homogeneous with the coolant.

For the pressurizer, the allowable heatup rate is 100°F per hour and the maximum cooldown rate for the pressurizer is 200°F per hour. The stresses are within acceptable limits for the anticipated usage. A maximum temperature difference (ΔT) of 320°F between the pressurizer and reactor coolant system is specified up to a maximum pressurizer temperature of 500°F (Reference 1 and 2). This allows steam bubble formation at an earlier time during startup to reduce the chances of an overpressure event by reducing the period during which the plant is solid. At pressurizer temperature greater than 500°F, ΔT is specified as 200°F with a minimum of 100°F. Spray actuation transients during the condition of ΔT greater than 100°F shall be limited to those in Table 2-2, Figure 2-1 and Figure 2-5 in Reference 1.

The fastest cooldown rates which result from the hypothetical case of a break of a main steam line are discussed in Section 14.

4.2.7 LEAKAGE

The existence of leakage from the Reactor Coolant System to the containment regardless of the source of leakage, is detected by one or more of the following conditions:

a) Two radiation sensitive instruments provide capability for detection of leakage from the Reactor Coolant System. The containment air particulate monitor is quite sensitive to low leak rates. The containment radiogas monitor is much less sensitive but can be used as a backup to the air particulate monitor.

b) An increase in the amount of coolant makeup water which is required to maintain normal level in the pressurizer, or an increase in containment sump level are less sensitive means of detection leakage.

c) One radiation sensitive instrument provides capability to detect Reactor Vessel Head Leakage. The leak detection system draws a sample from the Reactor Head Area or containment atmosphere in a skid mounted particulate sampling system located inside containment.

Leakage detection methods are described in detail and evaluated in Section 6.5.

<u>Leakage Prevention</u>

Reactor Coolant System components are manufactured to exacting specifications which exceed normal code requirements (as listed in Table 4.1-9). In addition, because of the welded construction of the Reactor Coolant System and the extensive non-destructive testing to which it is subjected (as outlined in Section 4.4), it is considered that leakage through metal surfaces or welded joints is very unlikely.

However, some leakage from the Reactor Coolant System is permitted by the reactor coolant pump seals. Also, all sealed joints are potential sources of leakage even though the most appropriate sealing device is selected in each case. Thus, because of the large number of joints and the difficulty of assuring complete freedom from leakage in each case, a small integrated leakage is considered acceptable. Leakage from the reactor through its head flange will leak-off between the double O-ring seal and actuate an alarm in the control room.

Locating Leaks

Experience has shown that hydrostatic testing is successful in locating leaks in a pressure containing system.

Methods of leak location which can be used during shutdown include visual observation for escaping steam or water or for the presence of boric acid crystals near the leak. The boric acid crystals are transported outside the Reactor Coolant System in the leaking fluid and deposited by the evaporation process.

4.2.8 WATER CHEMISTRY

The water chemistry is selected to provide the necessary boron content for reactivity control and to minimize corrosion of reactor coolant system surfaces.

All materials exposed to reactor coolant are corrosion resistant. During mode 1, periodic analyses of the coolant chemical composition are performed to monitor the adherence of the system to the reactor coolant water quality listed in Table 4.2-2. Chemistry specifications and limits for shutdown and startup conditions are maintained as outlined in the Florida Power and Light Nuclear Chemistry Parameters Manual and the Turkey Point Chemistry Procedures. Maintenance of the water quality to minimize corrosion is accomplished using the Chemical and Volume Control System and Sampling System which are described in Section 9.

4.2.9 REACTOR COOLANT FLOW MEASUREMENTS

Elbow taps are used in the reactor coolant system as an instrument device that indicates the status of the reactor coolant flow. The basic function of this device is to provide information as to whether or not a reduction in flow rate has occurred. The correlation between flow reduction and elbow tap read out has been well established by the following equation; $\frac{\Delta P}{\Delta P_o} = \left(\frac{\omega o}{\varpi}\right)^2$ where ΔP_o is the referenced pressure differential with the corresponding referenced flow rate ω_o and ΔP is the pressure differential with the corresponding flow rate ω. The full flow reference point is established during initial unit startup. The low flow trip point is then established by extrapolating along the correlation curve.

The technique has been well established in providing core protection against low coolant flow in Westinghouse PWR plants. The expected absolute accuracy of the channel is within ±10% and field results have shown the repeatability of the trip point to be within ±1%. The analysis of the loss of flow transient presented in Section 14.1.9 assumes instrumentation error of ± 3.7%.

4.2.10 REACTOR COOLANT SUBCOOLED MARGIN MONITOR

The reactor coolant system subcooled margin monitor system is an on-line microcomputer based system which uses reactor coolant process signals to provide a continuous indication of the margin from saturation conditions. The subcooled margin monitor system also provides an alarm signal into the main control room annunciator.

The reactor coolant system parameters monitored are the three coolant loops hot leg temperature, and loops A and B hot leg pressure. The operator has the choice of continuous main control board indication of either the pressure or temperature margin from saturation.

The temperature sensors are dual RTD's installed in thermowells. These RTD's are connected to provide the subcooling margin monitor system computing module with a 4-20 ma dc signal.

The reactor coolant pressure transmitters also provide a 4-20 ma dc signal to the computing module.

The computing module selects the highest temperature from those provided and the lowest pressure and calculates the margin to saturation from those two readings. The readings then appear on the display module in the control room.

4.2.11 REACTOR COOLANT VENT SYSTEM

The RCS vent system provides the operator with a means to vent non-condensable gases from the Reactor Coolant System. As shown on Figure 4.2-1 and 4.2-5, the RCS can be vented separately through the reactor vessel head vent or from the pressurizer steam space via the pressurizer relief line.

To vent system discharges to the containment Atmosphere and/or the pressurizer relief tank. A housekeeping drain is provided to the containment sump.

The RCS vent system can vent one-half of the RCS volume (gas) in one hour at operating pressure, but is sized such that the RCS mass inventory will be maintained by the charging pumps should the vent line suffer a guillotine break.

The power for the vent valves is taken from vital DC power outside the containment. The control power fuses are normally removed to prevent inadvertent operation of the vent valves under postulated fire conditions (see Engineering Guidelines for Fire Protection for Turkey Point Units 3 & 4 (Reference 14)). Valve control and position indication is located in the control room. Pressure indication is provided in the control room to assist the operator in determining leakage in the vent line. Each vent is powered from an emergency bus. C28

The vent system has been seismically analyzed.

4.2.12 REACTOR VESSEL DRAINDOWN LEVEL INDICATION SYSTEM

The reactor vessel drain down level indication system (see Figure 4.2-1) provides the continuous measurement of reactor coolant level during drain down operations and while in a drain down condition. The system consists of two independent and redundant level (differential pressure) transmitters with control room indication. This system provides audible and visual annunciation in the Control Room on decreasing reactor level below a preset value. Additionally, this system provides audible and visual annunciation in the Control Room on decreasing reactor level below an adjustable value or increasing reactor water above an adjustable value. This system also provides a local audio alarm (horn) and light (located at each steam generator manway) on increasing reactor level above a preset value.

4.2.13 REFERENCES

1. Westinghouse Electric Corporation, Report Number STC-TR-85-003 dated February 8, 1985, "Structural Evaluation - Pressurizer Surge Line and Spray Line for Pressurizer/RCS Differential Temperature of 320°F," PROPRIETARY.

2. Safety Evaluation, JPE-M-85-013, dated June 13, 1985, "Increased ΔT between Pressurizer and Reactor Coolant System to 320°F for PTP Unit 3."

3. NRC Letter, from G.E. Edison (NRC) to W.F. Conway (FPL), "Turkey Point Units 3 and 4 - Generic Letter 84-04, Asymmetric LOCA Loads," dated November 28, 1988.

4. NRC Letter, from R. P. Croteau (NRC) to J. H. Goldberg (FPL), "Turkey Point Units 3 and 4 - Approval to Utilize Leak-Before-Break Methodology for Reactor Coolant System Piping (TAC Nos. M91494 and M91495)," dated June 23, 1995.

5. Westinghouse WCAP-14237, "Technical Justification for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for the Turkey Point Units 3 and 4 Nuclear Power Plants," dated December 1994.

6. Westinghouse Electric Company, LTR-REA-17-116-NP, Revision 0, Reactor Vessel Neutron Exposure Data in Support of the Turkey Point Unit 3 and Unit 4 Subsequent License Renewal (SLR) Time-Limited Aging Analysis (TLAA), December 1, 2017. C31

7. DELETED

8. Westinghouse Electric Company, Report Number MTLS-06-0131, Revision 3, “Westinghouse Supplement to EPRI PWR Primary Water Chemistry Guidelines Rev.6,” dated March 6, 2008.

9. Letter to NRC from William Jefferson, Turkey Point, dated March 18, 2009, “Update to NRC Reactor Vessel Integrity Database and Exemption Request for Alternate Material Properties Bases per 10 CFR 50.12 and 10 CFR 50.60(b)”.

10. Framatome ANP Topical Report BAW-2308, Revision 1A, "Initial RTndt of Linde 80 Weld Materials", Approved August 2005.

11. Framatome ANP Topical Report BAW-2308, Revision 2A, "Initial RTndt of Linde 80 Weld Materials", Approved March 2008.

12. Letter from Jason Paige, NRC, to Mano Nazar, FPL, “Turkey Point Units 3 and 4 - Exemption from the Requirements of 10 CFR part 50, Appendix G and 10 CFR Part 50, Section 50.61 (TAC Nos. ME 1007 and ME 1008)”, March 11, 2010.

13. EC 281319 Unit 3 Pressurizer Heater #11 Element Nozzle/Sleeve Repair and Heater Replacement. Unit 3 pressurizer modification EC 281319 replaced the #11 heater well nozzle with a half nozzle design welded to the outside of the pressurizer shell instead of the internal cladding (Ref. NRC Relief Request SER:ADAMS Access1on No.: ML 15271A325.)

14. STD-M-006, Engineering Guidelines for Fire Protection for Turkey Point Units 3 & 4.

TABLE 4.2-1

MATERIALS OF CONSTRUCTION OF TRE
REACTOR COOLANT SYSTEM COMPONENTS

Component	Section	Materials
Reactor Vessel	Pressure Plate	SA-302, Gr. B
	Shell & Nozzle Forgings	A-508 Class II
	Cladding, Stainless Weld Rod	Type 304 equivalent
	Thermal Shield and Internals	A-240, Type 304
	Instrument Tubes	Inconel
Replacement RVCHs	CRDM Housing Bodies Head & Flange Mono Forging	Inconel Alloy 690 SA-508, Class 3
	Insulation	SS-A1 Foil-SS
Steam Generator	Plate (shell course)	SA-533 Grade A Class 2
	Tube Sheet Forging	SA-508 Class 2a
	Channel Head Casting	SA-216 Grade WCC
	Support Plates	SA-240 Type 405
	Channel Head Cladding	Stainless Steel, Type 304 or equivalent
	Tube Sheet Cladding	Inconel
	Tubes	SB-163 Thermally Treated
Pressurizer	Shell	SA-302, Gr. B
	Heads	SA-216 WCC
	External Plate	SA-302, Gr. B
	Cladding, Stainless	Type 304 equivalent
	Internal Plate	SA-240 Type 304
	Internal Piping	SA-376 Type 316
Pressurizer Relief	Shell	A-285 Gr. C
Tank	Heads	A-285 Gr. C
	Internal Coating	Vinyl

Component	Section	Material
Piping	Pipes	A-376 Type 316
	Fittings	A-351, CF8M
	Nozzles	A-182 F316
Pump	Shaft	Type 304
	Impeller	A-351, CF8
	Locknut	Type 304
	Casing	A-351, CF8M
	Bearings	Stellite and graphitar
	Seals	Silicon Carbide C28
Valves	Pressure Containing Parts	A-351, CF8M and A-182 F316
	Shafts, stems	17-4PH or equivalent
	Hard surfacing	Stellite 6 or equivalent
	Bushings, bearings	Cast Stellite 6 or equivalent
	Springs	Alloy 600 or equivalent corrosion resistant material
	Misc. Fasteners and washers	410 and 416 Series

General Note: This table represents original materials. Approved equivalents may be installed as necessary to support on-going maintenance.

TABLE 4.2-2

REACTOR COOLANT WATER CHEMISTRY SPECIFICATION

(Mode 1 Normal Values)

Electrical Conductivity	Determined by the concentration of boric acid and alkali present. Expected range is <1 to 40 uMhos/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.	0.1
Chloride, ppm, max.	0.15
Fluoride, ppm, max.	0.15
Hydrogen, cc (STP)/kg H_2O	25 - 50 (15-50 no more than 2 days prior to shutdown)
Total Suspended Solids, ppm, max.	1.0
pH Control Agent (Li), ppm	0.2 - 3.50*
Boric Acid as ppm B	Variable from 0 to approximately 4000

C24

NOTES:* Lithium concentrations at hot zero power critical conditions can be up to 5.0 ppm, but should be less than 3.5 ppm once equilibrium xenon levels are reached (Reference 8) C24

TABLE 4.2-3

STEAM GENERATOR WATER (STEAM SIDE) CHEMISTRY SPECIFICATION
MODE 1 AVT NORMAL VALUES

pH at 25°C	$\geq$ 9.0
Cation Conductivity, mhos/cc	$\leq$ 0.8
Na, ppm	$\leq$ 0.020
Cl, ppm	$\leq$ 0.020
SiO_2, ppm	$\leq$ 0.3
Sulfate, ppm	$\leq$ 0.020
Blowdown Rate	As necessary to maintain steam generator chemistry. However, a continuous blow-down is recommended.

TABLE 4.2-4

FURNACE SENSITIZED RCS STAINLESS STEEL COMPONENTS

1. Reactor Vessel

   a. CRDM housings - Inconel 690 bodies with Type 304 CRDM adapters. The replacement RVCH CRDM adapters are not sensitized.

   b. Bottom instrumentation nozzles - Inconel with 304 safe ends.

   c. Six (each vessel) primary nozzle safe ends - First layer is type 309; balance of layers is type 308 weld metal deposit.

   d. Gasket monitor tubes (not under pressure) - Type 304.

2. Steam Generator

   Two primary nozzle safe ends per generator - Type 309 first layer with balance of safe end of 308L weld.

3. Pressurizer

   All nozzle safe ends in top and bottom head - Type 316, inlet nozzle-forged (A182), balance-pipe (A312).

NOTE:

A. Reactor coolant piping field welds are Type 304 (A371) with filler of 308L (A298), and pass temperature was held <350F. The piping was water quenched during manufacture.

B. Core support structure heat treatment:

   Barrel welds - 165°F, furnace cooled.

   Other assembly welds - local heating to 75°F, air cooled.

FINAL SAFETY ANALYSIS REPORT

FIGURE 4.2-1

REFER TO ENGINEERING DRAWING

5613-M-3041 , SHEET 1

REV. 13 (10/96)

FLORIDA POWER & LIGHT COMPANY **TURKEY POINT PLANT UNIT 3**
REACTOR COOLANT SYSTEM **FIGURE 4.2-1**

FINAL SAFETY ANALYSIS REPORT
FIGURE 4.2-2

PART 1

REFER TO ENGINEERING DRAWING
5610-M-400-4, SHEET 1

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FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNITS 3 & 4

ARRANGEMENT OF REACTOR
VESSEL LONGITUDINAL SECTION

FIGURE 4.2-2
PART 1

FINAL SAFETY ANALYSIS REPORT
FIGURE 4.2-2

PART 2

REFER TO ENGINEERING DRAWING
5610-M-400-5, SHEET 1

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TURKEY POINT PLANT UNITS 3 & 4

ARRANGEMENT OF REACTOR VESSEL
LONGITUDINAL SECTION

FIGURE 4.2-2
PART 2

FINAL SAFETY ANALYSIS REPORT
FIGURE 4.2-2

PART 3

REFER TO ENGINEERING DRAWINGS
5613-M-460-2, SHEET 1
5614-M-460-2, SHEET 1

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TURKEY POINT PLANT UNITS 3 & 4

SPECIFICATION DRAWING FOR
REPLACEMENT REACTOR VESSEL
CLOSURE HEAD SECTION

FIGURE 4.2-2
PART 3

FINAL SAFETY ANALYSIS REPORT
FIGURE 4.2-2

PART 4

REFER TO ENGINEERING DRAWINGS
5613-M-460-2, SHEET 2
5614-M-460-2, SHEET 2

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TURKEY POINT PLANT UNITS 3 & 4

REPLACEMENT REACTOR VESSEL CLOSURE HEAD AS-BUILT DIMENSIONS

FIGURE 4.2-2
PART 4

Security-Related Information - Withheld Under 10 CFR 2.390

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PRESSURIZER

FIGURE 4.2-3

Security-Related Information - Withheld Under 10 CFR 2.390

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TURKEY POINT PLANT UNITS 3 & 4

STEAM GENERATOR

FIGURE 4.2-4

Security-Related Information - Withheld Under 10 CFR 2.390

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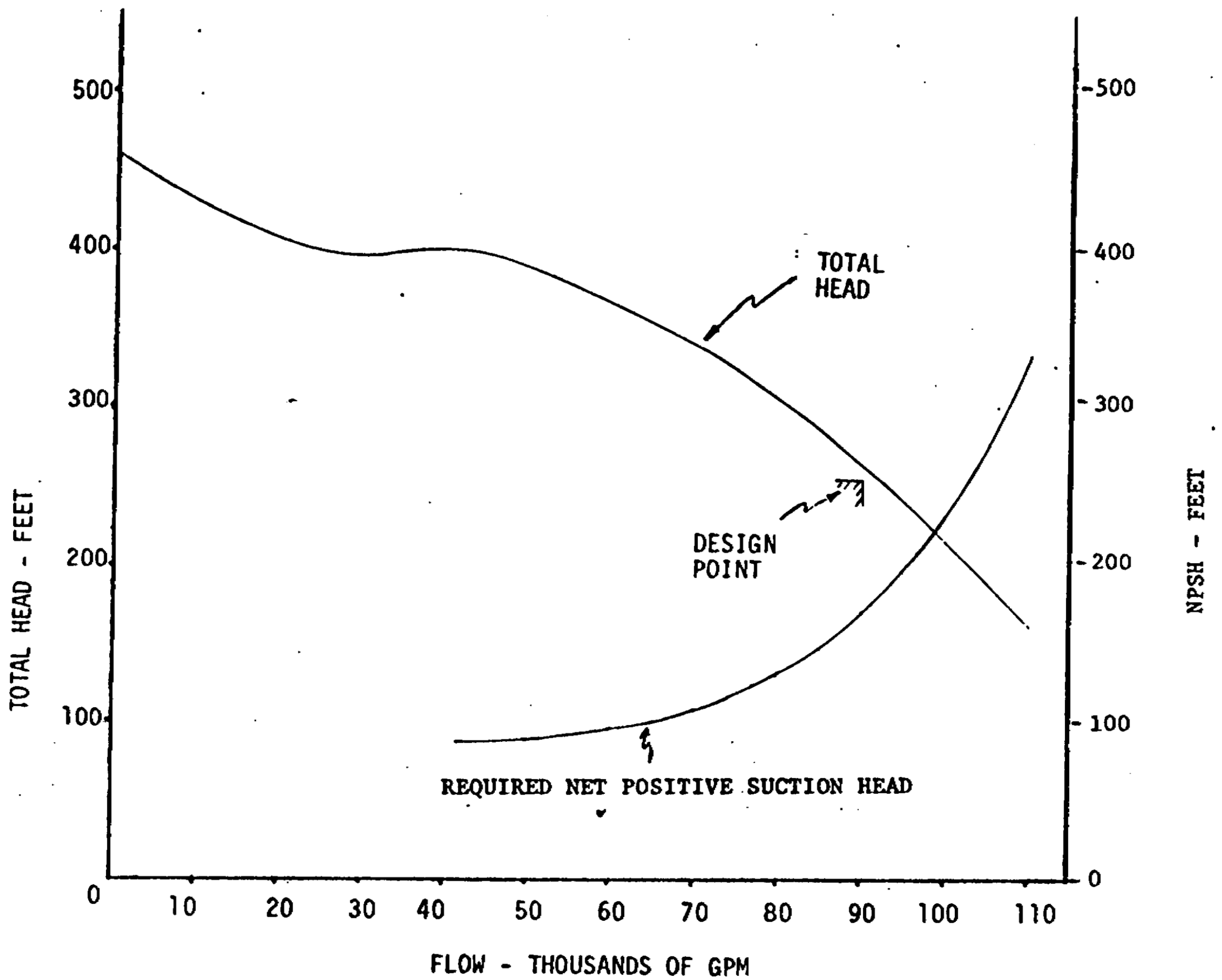
TURKEY POINT PLANT UNITS 3 & 4

REACTOR COOLANT CONTROLED
LEAKAGE PUMP

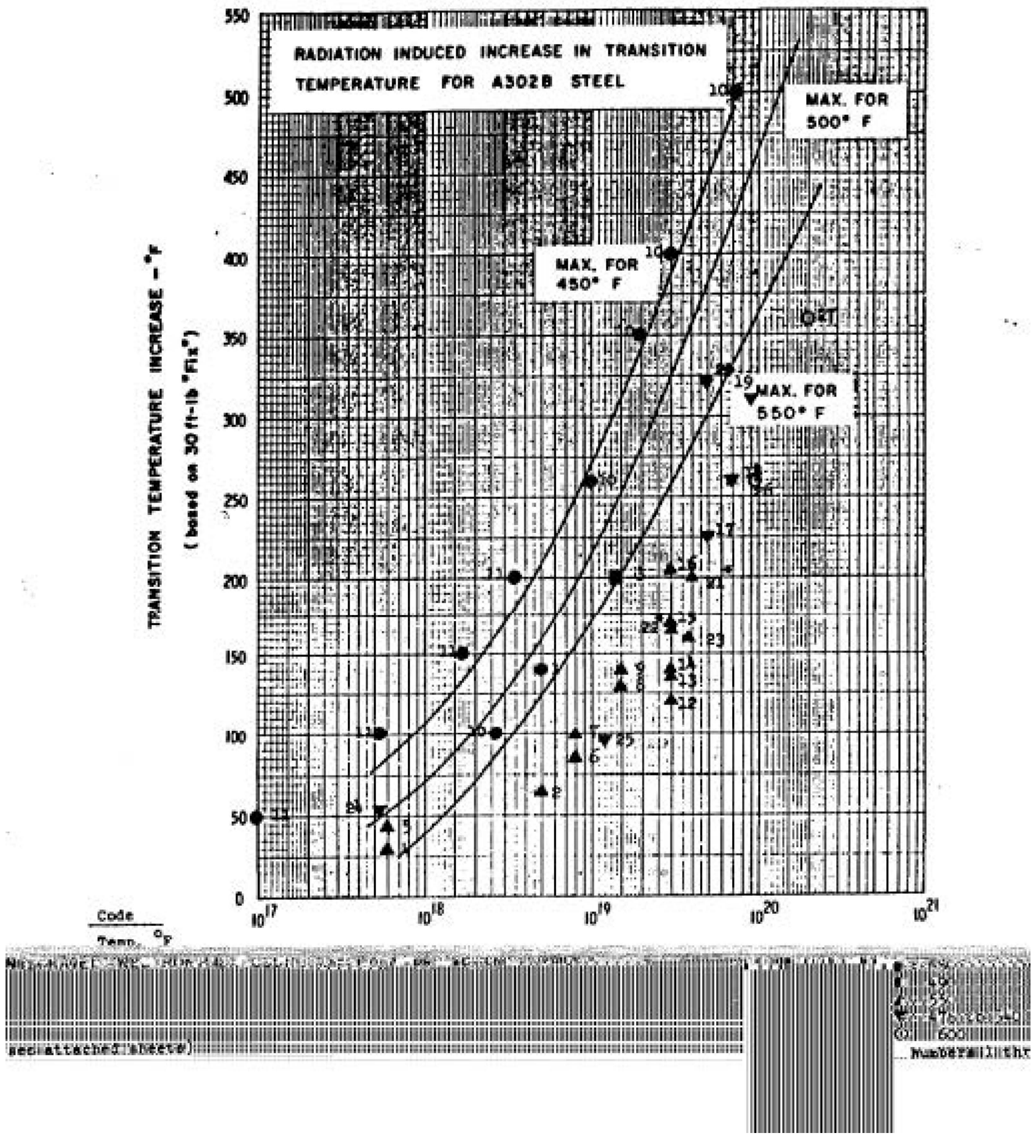
FIGURE 4.2-5



FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNITS 3 & 4

REACTOR COOLANT PUMP ESTIMATED PERFORMANCE CHARACTERISTIC

FIGURE 4.2-6



FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNITS 3 & 4

RADIATION INDUCED INCREASE IN TRANSITION TEMPERATURE for A302B STEEL

FIGURE 4.2-7
Sheet 1 of 4

References for Figure 4.2-7 Sheet 2 of 4

RADIATION INDUCED INCREASE IN TRANSITION TEMPERATURE for A302B STEEL

	References	Material	Temp. °F	Neutron Exposure n/cm^2 (<1Mev)	ΔNDT °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	6×10^{17}	30**
5.	ASTM-STP 341 Page 226	SA302B	550	6×10^{17}	45
6.	ASTM-STP 341 Page 226	SA302B	550	8×10^{117}	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

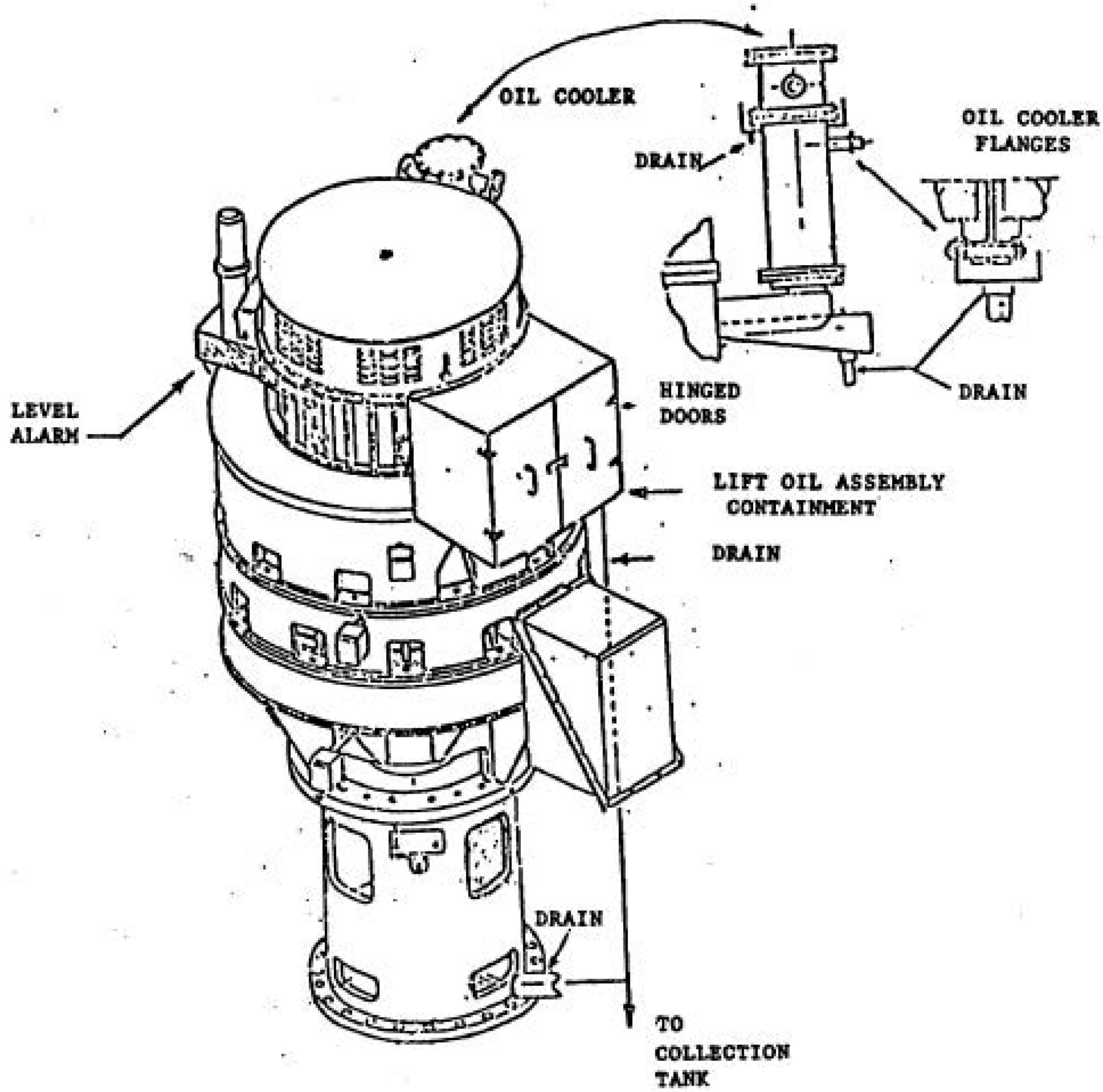
**Transverse Specimens

RADIATION INDUCED INCREASE IN TRANSITION TEMPERATURE for A302B STEEL

	References	Material	Temp. °F	Neutron Exposure n/cm^2 (<1Mev)	ΔNDT °F
13.	Quarterly Report of Progress, "Irradiation Effects on Reactor Structural Materials" 11-1-64/1-31-64	SA302B	550	3 X 10^{19}	135
14.	Quarterly Report of Progress, " Irradiation Effects on Reactor Structural Materials" 11-1-64/1-31-64	SA302B	550	3 X 10^{19}	140
15.	Quarterly Report of Progress, " Irradiation Effects on Reactor Structural Materials" 11-1-64/1-31-64	SA302B	550	3 X 10^{19}	170
16.	Quarterly Report of Progress, " Irradiation Effects on Reactor Structural Materials" 11-1-64/1-31-64	SA302B	550	3 X 10^{19}	205
17.	NRL Report 6179 Page 9	SA302B	475-540	5 X 10^{19}	225
18.	NRL Report 6179 Page 9	SA302B	475-540	7 X 10^{19}	260
19.	NRL Report 6179 Page 9	SA302B	475-540	9 X 10^{19}	310
20.	NRL Report 6179 Page 9	SA302B	475-540	5 X 10^{19}	320
21.	NRL Report 6160 Page 15	SA302B	540*	4 X 10^{18}	200
22.	NRL Report 6160 Page 15	SA302B	540*	3 X 10^{18}	165
23.	Private Communication with NRL	SA302B	550	3.8 X 10^{18}	160

RADIATION INDUCED INCREASE IN TRANSITION TEMPERATURE for A302B STEEL

	References	Material	Temp. °F	Neutron Exposure n/cm^2 (<1Mev)	ΔNDT °F
24.	Progress Report No. 1, "Irradiation Tests on Reactor Pressure Vessel Steels in Br-3 Reactor Facilities" August, 1965	SA302B	$\simeq$525	5.4×10^{18}	54
25.	"	SA302B	$\simeq$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	$\simeq$600	9.5×10^{19}	260
27.	"	SA302B	~600	2×10^{20}	360



REV. 3 7/85

FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNITS 3 & 4

REACTOR COOLANT PUMP MOTOR LUBE OIL FIRE PROTECTION

FIGURE 4.2-8

FINAL SAFETY ANALYSIS REPORT

FIGURE 4.2-9

REFER TO ENGINEERING DRAWING

5613-M-3041 , SHEET 2

REV. 13 (10/96)

FLORIDA POWER & LIGHT COMPANY **TURKEY POINT PLANT UNIT 3**
REACTOR COOLANT SYSTEM **FIGURE 4.2-9**

FINAL SAFETY ANALYSIS REPORT

FIGURE 4.2-10

REFER TO ENGINEERING DRAWING

5613-M-3041 , SHEET 3

REV. 13 (10/96)

FLORIDA POWER & LIGHT COMPANY

TURKEY POINT PLANT UNIT 3

REACTOR COOLANT SYSTEM

REACTOR COOLANT PUMPS

FIGURE 4.2-10

FINAL SAFETY ANALYSIS REPORT

FIGURE 4.2-11

REFER TO ENGINEERING DRAWING

5613-M-3041 , SHEET 4

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FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNIT 3

REACTOR COOLANT SYSTEM
PORV CONTROL

FIGURE 4.2-11

FINAL SAFETY ANALYSIS REPORT

FIGURE 4.2-12

REFER TO ENGINEERING DRAWING

5614-M-3041 , SHEET 1

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FLORIDA POWER & LIGHT COMPANY

TURKEY POINT PLANT UNIT 4

REACTOR COOLANT SYSTEM

FIGURE 4.2-12

FINAL SAFETY ANALYSIS REPORT

FIGURE 4.2-13

REFER TO ENGINEERING DRAWING

5614-M-3041 , SHEET 2

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FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNIT 4

REACTOR COOLANT SYSTEM

FIGURE 4.2-13

FINAL SAFETY ANALYSIS REPORT

FIGURE 4.2-14

REFER TO ENGINEERING DRAWING

5614-M-3041 , SHEET 3

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FLORIDA POWER & LIGHT COMPANY

TURKEY POINT PLANT UNIT 4

REACTOR COOLANT SYSTEM

REACTOR COOLANT PUMPS

FIGURE 4.2-14

FINAL SAFETY ANALYSIS REPORT

FIGURE 4.2-15

REFER TO ENGINEERING DRAWING

5614-M-3041 , SHEET 4

REV. 13 (10/96)

FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNIT 4

REACTOR COOLANT SYSTEM
PORV CONTROL

FIGURE 4.2-15

4.3 SYSTEM DESIGN EVALUATION

4.3.1 SAFETY FACTORS

The safety of the reactor vessel and all other Reactor Coolant System 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.

Reactor Vessel

A stress evaluation of the reactor vessel has been carried out in accordance with the rules of Section III of the ASME Nuclear Vessel Code. The evaluation demonstrates that stress levels are within the stress limits of the Code. Table 4.3-1 presents a summary of the results of the stress evaluation. Figures 4.3-1, 2, and 3 illustrate the areas of the pressure vessel that are analyzed in detail through systematic analytical procedures. The maximum thermal stress due to gamma ray heating occurs in the cylindrical portion of the vessel adjacent to the core and its value is about 2200 psi and is considered negligible.

A summary of fatigue usage factors for components of the reactor vessel is given in Table 4.3-2. The effect of gamma ray heating on the cumulative usage factor is negligible.

The cycles specified for the fatigue analysis are the results of an evaluation of the expected station operation coupled with experience from nuclear power plants now in service, such as Yankee-Rowe. These cycles include five heatup and cooldown cycles per year, a conservative selection when the vessel may not complete more than one cycle per year during normal operation.

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. To preclude the possibility of brittle failure the stresses allowed in the vessel in relation to operation below NDTT and DTT (NDTT+60°F) 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 the data reported (1,2), which show that if the stresses are maintained within the above limits, brittle fracture does not occur. These stress limits are maintained by prescribing operating procedures which rely upon administrative pressure and temperature control during heatup and cooldown as described in Reference 3, and by actuation of the overpressure mitigating system (OMS). The OMS is enabled/disabled by the operator during heatup and cooldown. The OMS is enabled by procedure when the RCS is less than 300°F and prior to operating without a bubble in the pressurizer. The OMS varies the setpoint of the power operated relief valve as the RCS temperature varies. Above 285°F, the OMS setpoints increase from 440 psig at 285°F to 2335 psig at 554°F as a segmented curve. Below 285°F, the setpoint is maintained at a constant 440 psig, while above 554°F (up to 750°F) the setpoint is maintained at a constant 2335 psig (Ref. 10). Technical Specifications LCO 3.4.9.3a indicates a PORV lift setting of $\leq$ 448 psig, however, the field device for OMS setpoint actuation will be set at 440 psig to provide buffer from the Technical Specifications value.

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The actual shift in RT(ndt) will be established periodically during unit 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 RT(ndt) caused by irradiation, the limits given in the unit operating manual on the pressure-temperature relationship are periodically changed to stay within the stress limits, which will be stated above during heatup and cooldown.

The vessel closure contains fifty-eight, 6-inch diameter studs. The stud material is ASTM A-540 with minimum yield strength of 104,400 psi at design temperature. Combined membrane stresses at design conditions of maximum calculated stress intensities are below the Code allowables. The membrane stress in the studs, when they are at the steady state operational conditions, is less than one half of the minimum yield strength.

The normal operating temperature always exceeds even the highest anticipated DDT during the life of the unit. Thus, the emphasis of conservative operation is placed on heatup and cooldown because long term irradiation of the vessel raises the DDT and thereby limits the heatup or cool down rates. The conservatism in setting up the temperature-pressure relationship limits stated above are:

1. Use of a stress concentration factor of 4 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 in Item 1 is not an actual stress concentration factor such as described in Article 4 Design of Section III but is a margin of conservatism based on the Fracture Analysis Diagram in ASTM E208 as well as the stress limits maintained by the prescribed operating procedures which rely upon administrative pressure and temperature control during heatup and cooldown as described in ASTM Paper No. 63-WA-100 "Reactor Vessel Design Considering Radiation Effects", L. Porse. 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 do not apply.

As part of the operator training program Westinghouse instructs supervisory and operating personnel 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 being helpful for future summation of time at power level and temperature which tends to influence the irradiated properties of the material in the core region. These instructions are incorporated in the operating manuals.

Steam Generators*

The Westinghouse analysis of the steam generator tubesheets is included as part of the Stress Report requirement for Class I Nuclear Pressure Vessels. The evaluation must be based on the stress and fatigue limitations outlined in Article-4 Design of Section III.

Calculations confirm that the steam generator tube sheet will withstand the loading (which is a quasi-static rather than a shock loading) by loss of reactor coolant. The maximum primary membrane plus primary bending stress in the tube sheet under these conditions is less than 35,550 psi per design stress. This is well below ASME Section III allowable stress of 45,000 psi at 650°F. Because the pressure in the primary channel head would drop to zero under the condition postulated, no damage will result to the channel head.

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 C26
the primary side or maximum pressure differential of 1100 psi across the tubes and tube sheet from the secondary side, respectively. Under these conditions there is no rupture of the primary to secondary boundary (tubes and tube sheet). This criterion prevents any violation of the containment boundary.

The tube sheet has been designed to accept a primary to secondary pressure differential of 1700 psi, which is the specified primary side design pressure differential. Under this pressure differential the stress criteria for design are a), the primary membrane stresses in the tube sheet ligaments, averaged across the ligament and through the tube sheet thickness, do not exceed the C26
material S_m value at the design temperature; and b), the primary membrane plus primary bending stress in the tube sheet ligaments, averaged across the ligament width at the tube sheet surface location giving maximum stress, do not exceed 150% of the material S_m value at the design temperature.

*See Appendix 4C for description replaced steam generator lower assemblies.

This use of these stress criteria for this abnormal operation is consistent with the ASME Boiler and Pressure Vessel Code, Section III Nuclear Vessels, paragraph N-414.1, N-414.2, and N-414.3 stress criteria. The stresses and stress factors in the actual tube sheet, obtained using the above stress criteria, are given in Table 4.3-3.

The tube sheet designed on the above basis meets code allowable stresses for a primary to secondary differential pressure of 1700 psi. The maximum normal operating differential pressure is 1549 psi.

The tubes have been designed to the requirements (including stress limitations) of Section III for normal operation, assuming 2485 psi 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 rate reported in Reference (4) shows "worst case" rates of 15.9 mg/dm^2 in the 2000 hour test under steam generator operating conditions. Conversion of this rate to a 80-year unit life gives a corrosion loss of less than approximately 3.00×10^{-3} inches which is insignificant compared to the nominal tube wall thickness of 0.050 inches. C31

In the case of a primary pressure loss accident, the secondary-primary pressure differential can reach 1100 psi. This pressure differential is less than the primary-secondary pressure differential capability (1549 psi) for normal operating conditions. Hence, no stresses in excess of those covered in Section III rules for normal operation are experienced on the tube sheet for this accident case. For the tubes, actual pressure tests of 3/4 in. O.D./.058 inch wall Inconel tubing show collapse under external pressure of 5700-5900 psi. Extrapolating these data to 7/8 in. O.D./.050 inch wall tubes, collapse would occur at about 2630 psi at 650°F. This gives a factor of safety of 2.4 against collapse under the 1100 psig accidental application of external pressure to tubes. The ASME Section VIII design curves for Iron-Chromium-Nickel Steel cylinders under external pressure indicate a predicted collapse pressure for the tubes of 2310 psi, which checks closely with the extrapolated value for the experimental results.

Consideration has been given to the superimposed effects of secondary side pressure loss and the maximum potential earthquake loading. The fluid dynamic forces on the internal components affecting the primary-secondary boundary (tubes) have been considered as well. For this condition the criterion is that no rupture of primary to secondary boundary (tubes and tube sheet) occurs.

For the case of the tube sheet, the maximum hypothetical earthquake 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 psi) for which the tube sheet is designed. Under horizontal shock loading of the maximum hypothetical earthquake the stresses are less than those for the 1.0g loading experienced by a steam generator when in a horizontal position, which the design can readily accept. The fluid dynamic forces on the internals under secondary 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 but boundary integrity is maintained.

The ratio of the allowable stresses on various tube sheet-head-shell components to the computed stresses for an abnormal primary to secondary pressure differential of 2485 psi are summarized in Table 4.3-4. The allowable stress limits for abnormal loading are yield and 150% of yield for membrane and membrane plus bending, respectively. C26

The steam generators were analyzed in accordance with Section III, N-415.1 of the ASME code (1965 Edition). Based upon this analysis, it was concluded that the only areas that required a fatigue analysis were the tubesheet and the mist extractor support. Results of these analyses give a fatigue usage factor of 0.41 for the mist extractor support and very low usage factors for the tubesheet, the greatest being 0.2976 at the secondary shell to tubesheet intersection.

Reactor Coolant Pumps

The casing, main flange and main flange bolts of the reactor coolant pump were analyzed in accordance with Article 4 Section III, ASME Code. The analysis included pressure, thermal and cyclic stresses.

Mathematical models of the parts were prepared and used in the analysis:

1) The design was checked against the design criteria of the ASME Code for pressure stresses. The shells were profiled to attain optimum metal distribution.

2) The interactivity forces needed to maintain geometric capability between the various components were determined at design pressure and temperature, and applied to the components along with the external loads, to determine the final stress state of the components. These were within the Code allowable values.

4.3.2 RELIANCE ON INTERCONNECTED SYSTEMS

The principal heat removal systems which are interconnected with the Reactor Coolant System are the Steam and Power Conversion, Safety Injection and Residual Heat Removal Systems. The Reactor Coolant System is dependent upon the steam generators, and the steam, feedwater, and condensate systems for stored and residual heat removal from normal operating conditions down to a reactor coolant temperature of approximately 350°F. The layout of the system ensures the natural circulation capability to permit unit cool down following a loss of all reactor coolant pumps.

The Steam and Power Conversion System is described in Section 10.2. 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 vented to the atmosphere. The auxiliary feedwater system will supply water to the steam generators in the event that the main feedwater pumps are inoperative.
The Safety Injection System is described in Section 6. The Residual Heat Removal System is described in Section 9.

Operation of a single residual heat removal loop is permitted for decay heat removal when fuel is in the reactor vessel and the refueling cavity is filled. With the reactor vessel head removed and at least 23 feet of water above the reactor pressure vessel flange, the heat sink and backup decay heat removal capability afforded by the large volume of water above the vessel flange will provide adequate time to initiate emergency procedures to cool the core in the event of a failure of the operating residual heat removal loop. The loss of this single residual heat removal loop has been evaluated to ensure that adequate natural circulation cooling can be maintained for decay heat removal. The analysis assumes that: (1) the reactor has been subcritical for at least 72 hours and (2) the reactor coolant system temperature is not more than 140°F. The analysis utilized GOTHIC thermal-hydraulic analysis software to evaluate natural circulation cooling conditions with both the reactor vessel upper internals assembly installed and with the upper internals assembly removed. In each case, stable natural circulation patterns occur such that adequate heat transfer capability is maintained to prevent fuel damage. For more detailed information regarding the analysis, see Reference 11.

For the upper internals assembly installed case, the natural circulation flow path modeled is up from the core to the vessel upper plenum to the refueling cavity via the holes in the upper support plate, the CRDM guide tubes, the head spray flow nozzles, the upper internals hold-down spring gap, and the hot leg gap, with return flow to the core via the downcomer and barrel/baffle bypass. A direct flow path for natural circulation from the core to the vessel upper plenum to the refueling cavity and back to the core via the downcomer and barrel/baffle bypass exists for the upper internals assembly removed case. The presence of these natural circulation flow paths provide assurance that, in the event of a loss of the single residual heat removal loop, the backup decay heat removal capability afforded by the 23 feet of water above the vessel flange can be credited, regardless of whether the upper internals assembly is installed or removed.

4.3.3 SYSTEM INTEGRITY

A complete stress analysis which reflects consideration of all design loadings detailed in the design specification has been prepared by the manufacturer. The analysis shows that the reactor vessel, steam generator, pump casing and pressurizer comply with the stress limits of Section III of the ASME Code. A similar analysis of the piping shows that it complies with the stress limits of the applicable USASI Code.

As part of the design control on materials, Charpy V-notch toughness test curves 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 were performed on the reactor vessel plate material. Refer to Table 4.3-5.

As an assurance of system integrity, all components in the system are hydrotested at 3107 psig prior to initial operation.

4.3.4 OVERPRESSURE PROTECTION

The Reactor Coolant System is protected against overpressure by safety valves located on the top of the pressurizer. The safety valves on the pressurizer are sized to prevent system pressure from exceeding the design pressure by more than 10 per cent, in accordance with Section III of the ASME Boiler and Pressure Vessel Code. The capacity of the pressurizer safety valves is determined from considerations of: (1) the reactor protective system, 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 system are assumed to open when the steam pressure reaches the secondary system safety valve setting.

Details of the analysis are reported in Section 14.1.8. Experience has shown that the safety valve capacity so determined is adequate for all the other transients as the results of Section 14.1 show.

4.3.5 SYSTEM ACCIDENT POTENTIAL

The potential of the Reactor Coolant System as a cause of accidents is evaluated by investigating the consequences of certain credible types of components and control failures as discussed in Sections 14.1 and 14.2. Reactor coolant pipe rupture is evaluated in Section 14.3.

4.3.6 REDUNDANCY

Each loop of the Reactor Coolant System contains a steam generator and a reactor coolant pump. Operation at reduced reactor power is possible with one loop out of service (Section 14.1.9). The normal power supply to the reactor coolant pumps is from two electrically separate buses, as shown in Figure 8.2-2.

REFERENCES

1. Kibara and Masubichi, "Effect of Residual Stress on Brittle Fracture," (Welding Journal Volume 38), April 1959.

2. Robertson, "Propagation From Brittle Fracture in Steel", (Journal of the Iron and Steel Institute), 1953.

3. L. Porse, "Reactor Vessel Design Considering Radiation Effects," ASME Paper No. 63-WA-100.

4. W.E. Berry and F.W. Fink, "The Corrosion of Inconel in High Temperature Water," Battelle Memorial Institute, April, 1958.

5. W. A. Van der Sluvs, "An Investigation of Mechanical Properties and Chemistry within a Thick MnMoNi Submerged Arc Weldment," EPRI NP-373, February 1977.

6. Westinghouse WNET-168, "Model 44F Replacement Steam Generator Stress Report for Florida Power and Light Company Turkey Point Units 3 & 4," January 1982.

7. Westinghouse WCAP-14291, "Turkey Point Units 3 and 4 Uprate Engineering Report," December 1995.

8. WCAP-15092 Revision 3, Turkey Point Units 3 and 4 WOG Reactor Vessel 60-Year Evaluation Minigroup Heatup and Cooldown Limit Curves for Normal Operation, Westinghouse, May 2000.

9. Low Temperature Overpressure Protection System Setpoints, Florida Power & Light Company, 32 EFPY and 48 EFPY for Turkey Point, Units 3 & 4. Westinghouse Electric Company LLC, June 2000.

10. PCM 00-027, Rev. 0, "Cold Overpressure Mitigation System (COMS) Setpoint Change."

11. Engineering Evaluation PTN-ENG-SENS-07-032, "RHR System Operation with the Reactor Cavity Filled and the Vessel Upper Internals In Place," Revision 0.

12. Framatome ANP Topical Report BAW-2008, Revision 2A, "Initial RTndt of Linde 80 Weld Materials", Approved March 2008. C26

TABLE 4.3-1

SUMMARY OF PRIMARY PLUS SECONDARY STRESS INTENSITY
FOR COMPONENTS OF THE REACTOR VESSEL

Area	Stress Intensity (psi)	Allowable Stress 3 S_m (psi) (Operating Temperature)
CRDM Housing	56,440	69,900
Head Flange	66,500	80,100
Vessel Flange	56,100	80,100
Closure Studs	84,700	104,400
Outlet Nozzles	46,142	80,100
Inlet Nozzles	59,686	80,100
Core Support pad(s)	21,078	80,100
Shell at Core Support Pads	35,637	80,100
Bottom head to shell juncture	34,785	80,100
Bottom instrumentation	70,300(1)	69,900
Shell to shell juncture	45,644	80,100

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(1) This value is greater than the allowable value. The ASME code year of construction specifies that the 3S_m limit can be exceeded based on the occurrence of shakedown, but does not specify a procedure. The simplified elastic plastic procedure from the 1998 Edition of ASME Code through the 2000 Addenda was used to demonstrate shakedown, and it was concluded that the Code Criteria was met.

TABLE 4.3-2

SUMMARY OF CUMULATIVE FATIGUE USAGE FACTORS FOR
COMPONENTS OF THE REACTOR VESSEL

Item	Usage Factor*
Control Rod Housing	0.73
Head Flange	0.083
Vessel Flange	0.531
Stud Bolts	0.81
Outlet nozzles	0.063
Inlet nozzles	0.066
Core support pad	0.020
Shell at Core Support Pads	0.509
Bot. head to shell juncture	0.023
Bot. instrumentation	0.002
Shell to shell juncture	0.034

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* As defined in Section III of the ASME Boiler and Pressure Vessel Code, Nuclear Vessels.

TABLE 4.3-2a

SUMMARY OF CUMULATIVE FATIGUE USAGE FACTORS FOR
PRESSURE BEARING COMPONENTS OF THE REACTOR COOLANT PUMPS

Item	Usage Factor
Casing	< 0.001
Main Flange	0.025
Main Flange Studs	0.29

C26

TABLE 4.3-3

MAXIMUM LIGAMENT STRESSES DUE TO STEAM GENERATOR TUBE SHEET DESIGN PRESSURE DIFFERENTIAL (1700 PSI)

Stress	Computed Value	Allowable Value
Primary Membrane Stress	8.9 ksi	30 ksi
Primary Membrane plus Primary Bending Stress	40.7 ksi	45 ksi

In addition to the foregoing evaluation, elasto-plastic limit analysis of the tube sheet-head-shell combination indicates an allowable limit differential pressure of 3100 psi at 650 F, which compares to the 1700 psi primary to secondary design differential pressure.

C26

TABLE 4.3-4

RATIO OF ALLOWABLE STRESSES TO COMPUTED STRESSES
FOR A STEAM GENERATOR TUBE
SHEET ABNORMAL PRESSURE DIFFERENTIAL OF 2485 PSI at 668 F

Component Part	Yield (Membrane)	150% Yield (Membrane + Bending)
Channel head	3.38	4.29
Channel head-tube sheet joint	1.48	1.14
Shell	3.91	2.82
Tube sheet	7.36	
Max. Avg. Ligament		1.59

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TABLE 4.3-5

SUMMARY OF RESULTS OF CHARPY V-NOTCH
AND DROP WEIGHT TESTS FOR REACTOR
VESSEL PLATES AND FORGINGS AND BELTLINE WELDS

Component	Grade	30ft-lb Fix Min. Curve (Temp/°F) Unit 3	Unit 4	Drop Weight NDT (°F) Unit 3	Unit 4
Replacement RVCH Mono Forging	SA508 Class 3	****	****	-50	-50
Vessel Flange	" "	-62	-36	-	-
Upper Shell	" "	20	-25	50	40
Inlet Nozzle	" "		-18	-	-
" "	" "			-	-
" "	" "	-19		-	-
Outlet Nozzle	" "	-60	-63	-	-
" "	" "	-20	-20	-	-
" "	" "	8	8	-	-
Intermediate Shell	" "	-8	42	40	50
Lower Shell	" "	-26	-22	30	40
Lower Transition Ring	" "		18	-	-
Bottom Head Dome	A302 Grade B	6	0	-10	0

Unit	Vessel Component	Drop. Wt. NDT Temp. (°F)	50 Ft.-Lb. Temp. (°F) Long.	Trans.	Upper Shelf Energy (Ft.-Lbs.) Long.	Trans.
No. 3	Upper Shell	50	20	40**	--	99**
No. 3	Core Region Weld	-60***	--	70***	--	65
No. 4	Inter. Shell	50	50	70	--	88
No. 4	Core Region Weld	-60***	--	70***	--	65

* This information represents the vessel components having the highest drop weight NDTT, the highest 50 ft.-lb. temperature and the lowest upper shelf energy level.

** Estimated

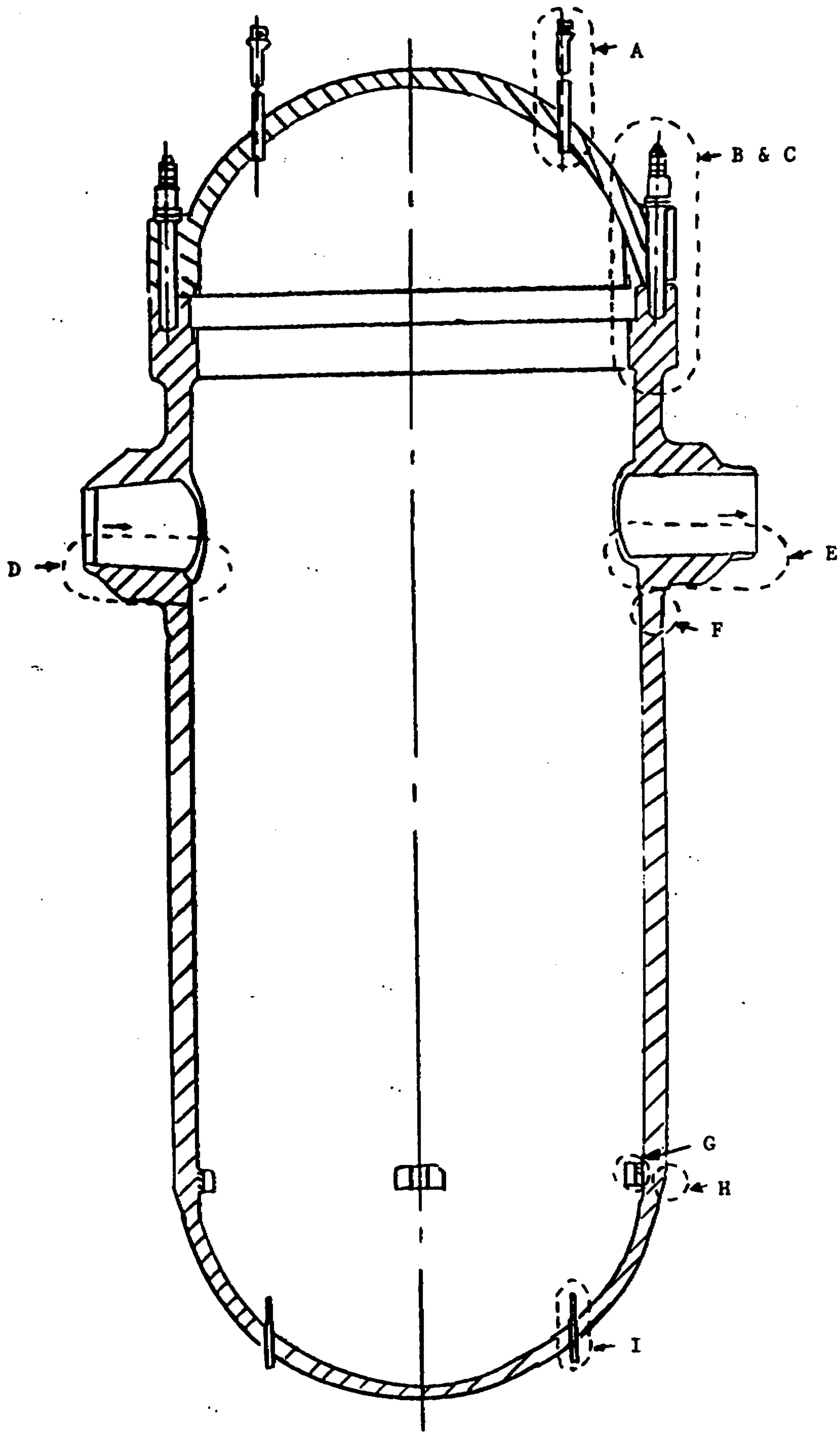
*** As per Reference 5; However, per Reference 12, Initial RT_{NDT} = -53.5°F and σ_i = 12.8.

**** From 6 Charpy Impact tests conducted at T_{NDT} + 60°F (10°F), the minimum absorbed energy was 142 ft.-lbs. Which is above the required 50ft.-lbs and the minimum lateral expansion was 79 mils. Which is above the required 35 mils. minimum. Based on these results, RT_{NDT} = -50°F.

TABLE 4.3-6

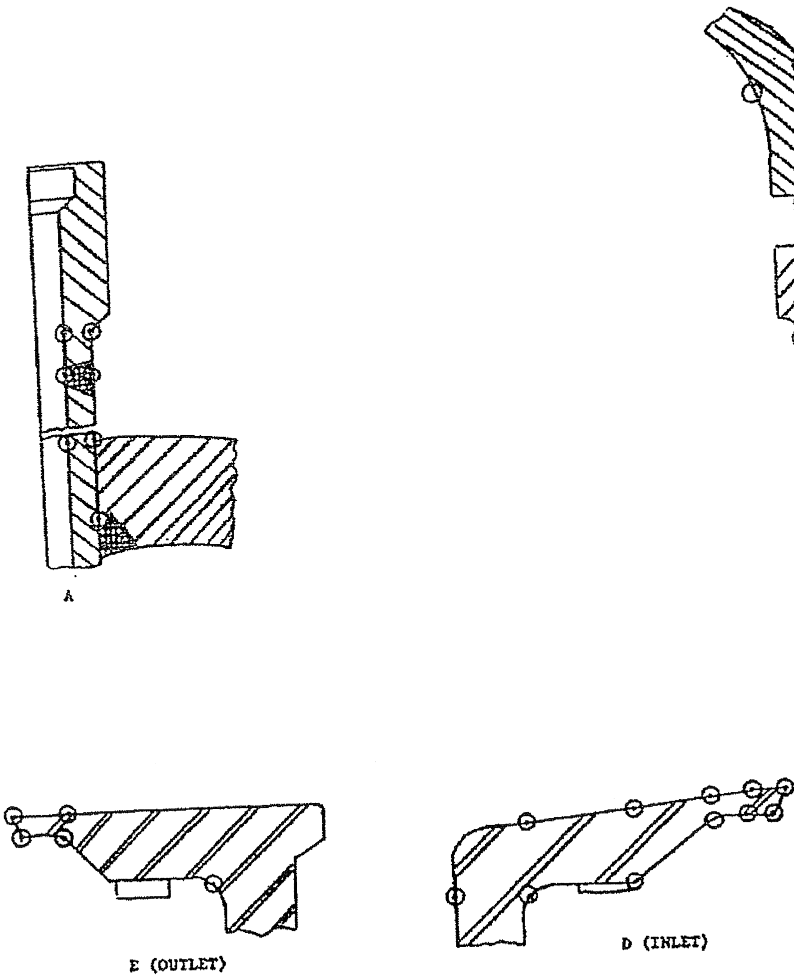
SUMMARY OF ESTIMATED STRESS INTENSITIES
FOR AREAS OF CONCERN IN THE STEAM GENERATORS

[DELETED]



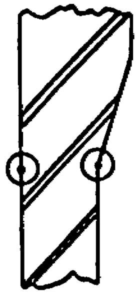
Reactor Vessel Stress Analysis: Areas Examined

FIGURE 4.3-1

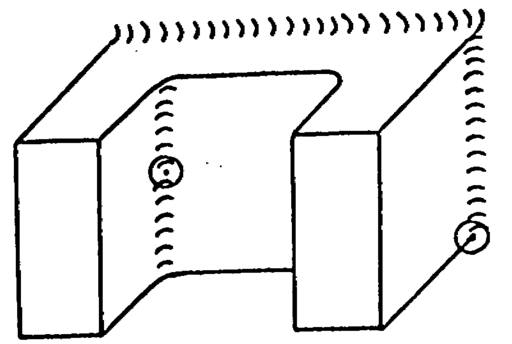


Reactor Vessel Stress Analysis: Details – Upper
Figure 4.3-2

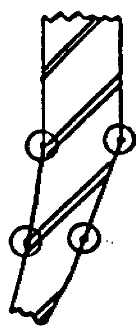
Revised 07/19/2005



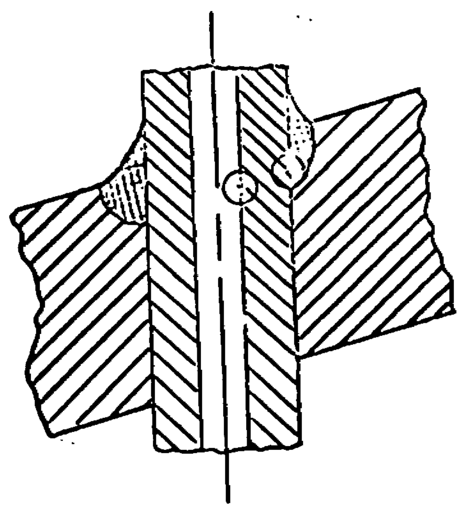
F



G



H



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NOTE:

THE POINTS CIRCLED IN THE SKETCHES REPRESENT THE GENERAL LOCATION AND GEOMETRY OF THE AREAS OF DISCONTINUITY AND/OR STRESS CONCENTRATION.

Reactor Vessel Stress Analysis: Details - Lower

FIGURE 4.3-3

4.4 TESTS AND INSPECTIONS

4.4.1 REACTOR COOLANT SYSTEM INSPECTION

Non-Destructive Inspection of Materials and Components (See Note 1)

Table 4.4-1 summarizes the quality assurance program for all Reactor Coolant System components. In this table all of the non-destructive tests and inspections which are required by Westinghouse specifications on Reactor Coolant System 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.

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 below:

1) Ultrasonic Testing - Westinghouse requires that a 100% volumetric ultrasonic test of reactor vessel plate for both shear wave and longitudinal wave be performed. Section III Class A vessel plates are required by code to receive only a longitudinal wave ultrasonic test on a 9 in. x 9 in. grid. The 100% volumetric ultrasonic test is a severe requirement, but it assures that the plate is of the highest quality.

2) Radiation Surveillance Program - This program monitors the effects of neutron irradiation on the reactor vessel beltline. Irradiation damage is based on pre- and post- irradiation testing of charpy V-notch and tensile test specimens. The program evaluates the effect of irradiation on the fracture toughness of reactor vessel steels and weldments by measuring transition temperature shift and using a fracture mechanics methodology. The program is in accordance with ASTM E185, "Recommended Practice For Surveillance Tests on Structural Material in Nuclear Reactors," required by 10 CFR 50 Appendix H.

Note 1:

The Unit 3 AND Unit 4 Reactor Vessel Closure Heads (RVCH) have been replaced. See the discussion at the end of this heading for the NDE information pertaining to the replacement RVCHs.

The original program used eight surveillance capsules in each vessel which are mounted on the thermal shield about 3 inches from the wall opposite the core midplane. Capsules are withdrawn according to a schedule and the surveillance materials are tested. The capsules contain reactor vessel steel machined specimens from forgings, welds, and heat affected zones as well as correlation monitor material. Dosimeters are included to permit the evaluation of the flux seen by the specimens. Dosimetry includes Ni, Cu, Fe, Co, Al, Cd shielded Co-Al, Cd shielded Np-237, and Cd shielded U-238. Thermal monitors of low melting point alloys are included to monitor the temperature range seen by the specimens. Wedge open loading fracture toughness specimens are included in the capsules but have not been tested because they were not yielding meaningful data. They will be tested in the future.

The circumferential girth weld of both vessels is the limiting weld material and is a high Copper (0.23%) Linde 80 flux submerged arc weld (SA1101) for both. This weld appears as surveillance material in only 3 capsules for each vessel. Since this would not supply enough data throughout life on an individual vessel basis, the surveillance programs were integrated in accordance with the provisions of Appendix H. The integrated program approach has been approved for use at PTN by the NRC. The surveillance program shown in Table 4.4-2 reflects the integrated program and meets all requirements.

There are two (2) supplemental capsules which contain the limiting weld SA 1101 being irradiated in the Babcock and Wilcox Owners' Group master integrated surveillance program. When these capsules are removed and tested, the data will be evaluated and considered as appropriate.

Table 4.4-1 summarizes the quality assurance program with regard to inspections performed on primary system components. In addition to the inspections shown in Table 4.4-1, there are those which the equipment supplier performs to confirm the adequacy of material he receives, and those performed by the material manufacturer in producing the basic material. The inspections of reactor vessel, pressurizer, and steam generator are governed by ASME code requirements. The inspection procedures and acceptance standards required on pipe materials and piping fabrication are governed by USAS B31.1 and Westinghouse requirements and are equivalent to those performed on ASME coded vessels.

Procedures for performing the examinations are consistent with those established in the ASME Code Section III and are reviewed by qualified Westinghouse 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 surfaces most subject to damage as a result of the heat treating, rolling, forging, forming and fabricating processes, receive a 100% surface inspection by Magnetic Particle or Liquid Penetrant Testing after all these operations are completed. All reactor coolant plate material is subject to shear as well as longitudinal ultrasonic testing to give maximum assurance of quality. (All forgings receive the same inspection.) In addition, 100% of the material volume is covered in these tests as an added assurance over the grid basis required in the code.

Westinghouse Quality Control engineers monitor 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 includes verification of records of material, physical and chemical properties, review of radiographs, performance of required tests and qualification of supplier personnel. Florida Power and Light Company, using Bechtel and others as consultants has reviewed the quality control methods and results of vendors and has found them to be satisfactory.

Field erection and field welding of the reactor coolant system are performed such as to permit exact fit-up of the 31" I.D. closure pipe subassemblies between the steam generator and the reactor coolant pump. After installation of the pump casing and the steam generator, measurements are taken of the pipe length required to close the loop. Based on these measurements, the 31" I.D. closure pipe subassembly is properly machined and then erected and field welded to the pump suction nozzle and to the steam generator exit nozzle.

Cleaning of RCS piping and equipment is accomplished before and during erection of various equipment. Stainless steel piping is cleaned in sections as specific portions of the systems are erected. Pipe and units large enough to permit entry by personnel are cleaned by locally applying approved solvents (acetone or alcohol), and demineralized water, and by using a rotary disc sander or 18-8 wire brush to remove all trapped foreign particles. Standards for final physical and chemical cleanliness are defined in Section 13.

Equipment specifications for fabrication require that suppliers submit the manufacturing procedures (welding, heat treating, etc.) to Westinghouse where they are reviewed by qualified Westinghouse engineers. This also is done on the field fabrication procedures to assure that installation welds are of equal quality.

Section III of the ASME B&PV Code requires that nozzles carrying significant external loads shall be attached to the shell by full penetration welds. This requirement has been carried out in the reactor coolant piping, where all auxiliary pipe connections to the reactor coolant loop are made using full penetration welds.

The Reactor Coolant System components are welded under procedures which require the use of both preheat and post-heat. Preheat requirements, non-mandatory under Code rules, are 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 post-heat of weldments both serve 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 post-heating achieves this by tempering any hard zones which may have formed due to rapid cooling.

Replacement RVCH Non-Destructive Inspection

Table 4.4-1 summarizes the quality assurance program inspections for the replacement RVCHs. In this table are identified all of the nondestructive test and inspections required by the RVCH design specification. All tests required by the applicable Code (ASME Section III, 1989 Edition, no Addenda) are included in the table as well as any additional test or more stringent acceptance criteria as may have been specified in the design specification for the RVCH.

In addition to the inspections summarized in Table 4.4-1, there are those inspections which the equipment supplier performs to confirm the adequacy of material he receives, and those performed by the manufacturer of the materials in producing the basic materials. Procedures for performing all of the examinations are consistent with those established in the ASME Code SectionIII and are reviewed by qualified FPL and Owner's Agent representatives. These procedures have been developed to provide the highest assurance of quality in the materials and fabrication. They consider not only the size of 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. The volumetric inspections (Ultrasonic Testing) of the forging were done using both the straight beam and the angle beam techniques. In addition, the surfaces most subject to damage as a result of forging, heat treating, forming, fabricating, and hydrostatic testing received 100% surface inspections by Magnetic Particle or Liquid Penetrante Testing at various stages during the processes and after final completion of the hydrostatic test of the RVCH.

The RVCH requires welding and weld cladding performed under procedures which require the use of both preheat and post heat treating. Preheat of weld areas and post-heat treating are performed on all welds on the replacement RVCH. Preheat and post-heat of weldments both serve the common purpose of producing tough, ductile metallurgical structures in the weldment. Preheating produces tough ductile welds by minimizing the formation of hard non-ductile zones whereas post weld heat-treating achieves this by tempering any hard zones which may have formed due to rapid cooling.

FPL and the Owner's Agent reviewed the manufacturer's quality control methods and results of the vendor and subvendor of the RVCH and have found them to be acceptable. FPL and the Owner's Agent Quality Control engineers monitored the supplier's work, witnessing key inspections not only in the supplier's shop but in the shops of the subvendor of the major forging. Normal surveillance includes verification of records of material, physical and chemical properties, review of radiographs, performance of the required tests and qualification of supplier personnel. FPL and the Owner's Agent reviewed the manufacturing quality control results and records of the vendor and subvendors of the RVCH and have found them to be complete and acceptable.

<u>In-Service Inspection Capability</u>

During the design phase of the Reactor Coolant System, careful consideration is given to provide access for both visual and non-destructive in-service inspection of primary loop components. The following components and areas are available for visual and/or non-destructive inspection.

1) Reactor Vessel - The entire inside surface.
2) Reactor Vessel Nozzles - The entire inside surface.
3) Closure Head - The entire inside and outside surface.
4) Reactor Vessel Studs, Nuts and Washers.
5) Field Welds between the Reactor Vessel, Steam Generators, and Reactor Coolant Pumps and the Main Coolant Piping.
6) Reactor Internals
7) Reactor Vessel Flange Seal Surface
8) Fuel Assemblies
9) Rod Cluster Control Assemblies
10) Control Rod Drive Shafts
11) Control Rod Drive Mechanism Assemblies
12) Main Coolant Pipe External Surfaces (except for the five foot penetration of the primary shield)

13) Steam Generator - The internal surface, the internal surfaces of the steam drum, and channel head.
14) Pressurizer - The internal and external surfaces.
15) Reactor Coolant Pump - The external surfaces; motor and impeller.

The design considerations which have been incorporated into the primary system design to permit the above inspections are as follows:

1) All reactor internals are completely removable. The storage space required to permit these inspections is provided.
2) The closure head is stored dry on the reactor operating deck during refueling to facilitate visual inspection.
3) All reactor vessel studs, nuts and washers are removed to dry storage during refueling.
4) Removable plugs are provided in the primary shield just above the coolant nozzles, and the insulation covering the nozzle welds may be removed.
5) Access holes are provided in the lower internals barrel flange to allow remote access to the reactor vessel internal surfaces between the flange and the nozzles without removal of the internals.
6) A removable plug is provided in the lower core support plate to allow access for inspection of the bottom head without removal of the lower internals.
7) The storage stands provided for storage of the internals allow for inspection access to both the inside and outside of the structures.
8) The station provided for changeout of control rod clusters from one fuel assembly to another is specially designed to allow inspection of both fuel assemblies and control rod clusters.
9) The control rod mechanism is designed to allow removal of the mechanism assembly from the reactor vessel head.

10) Manways are provided in the steam generator, steam drum and channel head to allow access for internal inspection.
11) A manway is provided in the pressurizer top head to allow access for internal inspection.
12) All insulation on primary system components (except the reactor vessel) and piping (except for the penetration in the primary shield) may be removed.

The use of non-destructive, direct visual and remote visual test techniques can be applied to the inspection of primary loop components other than the reactor vessel. The reactor vessel requires special consideration because of the radiation levels and remote underwater accessibility to this component. Because of these limitations on access to the reactor vessel, several steps have been incorporated into the design and manufacturing procedures in preparation for non-destructive test techniques which may be available in the future. These are:

1) Shop ultrasonic examinations are performed on all internally clad surfaces to an acceptance and repair standard to assure an adequate cladding bond to allow later ultrasonic testing of the base metal. Size of cladding bonding defect allowed is 3/4 inch, which permits subsequent UT of the base metal through the clad surface.

2) The design of the reactor vessel shell in the core area is a clean, uncluttered cylindrical surface to permit future positioning of test equipment without obstruction.

3) During the manufacturing stage, selected areas of the reactor vessel were ultrasonic tested and mapped to facilitate possible future in-service inspection.

The areas which were ultrasonic mapped include:

a) Vessel flange radius, including the vessel flange to upper shell weld.
b) Middle shell course.
c) Lower shell course above the radial core supports.
d) Exterior surface to the closure head from the flange knuckle to the cooling shroud.
e) Nozzle to upper shell weld.
f) Middle shell to lower shell weld.
g) Upper shell to middle shell weld.

The pre-operational ultrasonic testing of these areas was performed after shop hydrotest.

Plans for inservice inspection of the reactor coolant system pressure envelope are currently being developed. The applicability of ultrasonic testing
techniques is also being evaluated.

Various tests are currently underway to determine the effect of cladding surface finish on ultrasonic inspectability of vessel material.

For the Unit 3 and Unit 4 replacement RVCHs, a baseline UT examination was performed on all of the nozzle penetration to RVCH forging welds after final PT of the welds. The acceptance criteria for the PT was no indications (PT White). The baseline UT was performed using the best technique and practices available and the results are achieved for future ISI comparative reference.

TABLE 4.4-1

REACTOR COOLANT SYSTEM
QUALITY ASSURANCE PROGRAM

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			yes		
1.6.4 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)				yes	
1.6.10 Nozzle safe ends (if forgings)	yes		yes		
1.6.11 Nozzle safe ends (if weld deposit)			yes		
2. Pressurizer					
2.1 Heads					
2.1.1 Casting				yes	
2.1.2 Cladding	yes		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		
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 (if forging)	yes		yes		
2.5.5 Nozzle Safe End (if weld deposit)			yes		
2.5.6 Instrument Connections			yes		
2.5.7 Support Skirt				yes	
2.5.8 Temporary attachments after removal				yes	

TABLE 4.4-1

Component	RT*	UT*	PT*	MT*	ET*
2.5.9 All welds after hydrostatic test				yes	
3. Piping					
3.1 Fittings (Castings)	yes		yes		
3.2 Fittings (Forgings)		yes	yes		
3.3 Pipe		yes	yes		
3.4 Weldments					
3.4.1 Longitudinal	yes		yes		
3.4.2 Circumferential	yes		yes		
3.4.3 Nozzle to run pipe	yes		yes		
3.4.4 Instrument Connections	yes		yes		
4. Pumps					
4.1 Casting	yes		yes		
4.2 Forgings	yes	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		
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 Seam		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		
5.3.8 Head adapter forging to head adapter tube	yes		yes		
5.3.9 All welds after hydrotest			yes		
6. Valves					
6.1 Castings	yes	yes			
6.2 Forgings	yes	yes			

Component	RT*	UT*	PT*	MT*	ET*
7. Unit 3 and Unit 4 Replacement Reactor Vessel Closure Head					
7.1 Head Mono-block Forging					
7.1.1 After Rough Machining		yes		yes	
7.1.2 After Final Machining		yes			
7.1.3 Machined Surfaces to be Clad			yes (4)		
7.1.4 External Un-clad Surfaces				yes (1)	
7.1.5 All Clad Surfaces		yes (2&3)	yes (5)		
7.1.6 Final Machined O-Ring Groove			yes (6)		
7.2 Inconel Alloy 690 CRDM Housing		yes (10)	yes (7)		
7.3 SA-182 F304 CRDM Nozzle Adapter		yes (10)	yes (7)		
7.4 Weldment					
7.4.1 All Weld Prep Areas			yes (8)		
7.4.2 Root Pass of All Welds			yes		
7.4.3 Final Surface of All Welds	yes		yes (9)		
7.4.4 Nozzle to Forge Weld Areas		yes (11)			

1. All accessible ferritic surfaces after final hydrostatic test.
2. Sealing and bearing surfaces of the head examined for defects and bond.
3. Non-sealing and non-bearing surfaces examined for bond.
4. After machining and prior to cladding.
5. After post weld heat treatment.
6. The bottom sealing surfaces must be free of indications (PT White).
7. After final machining.
8. After final weld prep machining but prior to root pass welding.
9. Final surfaces of all CRDM nozzle attachment and vent piping welds must be free of indications (PT White).
10. After rough machining.
11. This is a baseline for future examinations.

*RT - Radiographic; UT - Ultrasonic; PT - Dye Penetrant;
MT - Magnetic Particle; ET - Eddy Current
UT of clad bond-to-base metal.

Table 4.4-2

Surveillance Capsule Withdrawal Schedule
Turkey Point Units 3 & 4[(e)]

Capsule (Unit shown as subscript)	Capsule Location (Degree)	Withdrawal EFPY(b)	Lead Factor(d)	Fluence (n/cm^2, E > 1.0 MeV)
T_3	270°	1.15	2.736	0.5990×10^{19}
T_4	270°	1.17	2.74	0.649×10^{19}
S_3	280°	3.46	1.997	1.272×10^{19}
S_4	280°	3.41	2.03	1.29×10^{19}
V_3	290°	8.06	0.891	1.223×10^{19}
X_3	270° / 50°(a)	19.85	1.129	2.897×10^{19}
X_4	270° / 50°(a)	41.5(c)	2.088	1.08×10^{20}
V_4	290°	Standby	1.015	---
U_3	30°	Standby	0.767	---
U_4	30°	Standby	0.767	---
W_3	40°	Standby	0.523	---
W_4	40°	Standby	0.523	---
Y_3	150°	Standby	0.767	---
Y_4	150°	Standby	0.767	---
Z_3	230°	Standby	0.523	---
Z_4	230°	Standby	0.523	---

(a) Capsule X_3 and Capsule X_4 were moved from the 50° location to the 270° location in 1990.

(b) Effective Full Power Years (EFPY) from plant startup.

(c) Capsule X_4 should be removed at the first refueling outage that meets or exceeds 41.5 EFPY to fulfill the requirements of the "5th Capsule" to be withdrawn. This EPFY will yield a capsule fluence that is approximately equivalent to the 80-year (72 EFPY) peak vessel fluence of 1.08×10^{20} n/cm^2 (E> 1.0 MeV). C31

(d) The lead factors listed for Capsule x_4 and the standby capsules are 48 EPFY projections and pertain to the most limiting core design case (lowest lead factor). The lowest lead factor is considered most limiting to prevent premature capsule withdrawal. Turkey Point Unit 3 and 4 operate under an integrated surveillance program. Therefore, the Unit 4 standby capsule lead factors are approximated to be equivalent to the Unit 3 standby capsule lead factors.

(e) Capsule removal changes require NRC approval per 10 CFR 50 Appendix H.

APPENDIX 4A

DETERMINATION OF REACTOR PRESSURE VESSEL
REFERENCE NIL-DUCTILITY TRANSITION TEMPERATURE (RT_{NDT})

1. MEASUREMENT OF INTEGRATED FAST NEUTRON (E > 1.0 MEV) FLUX AT THE IRRADIATION SAMPLES

The use of passive neutron sensors such as those included in the internal surveillance capsule dosimetry sets does not yield a direct measure of the energy dependent neutron flux level at the measurement location. Rather, the activation or fission process is a measure of the integrated effect that the time-dependent and energy-dependent neutron flux has on the target material over the course of the irradiation period. An accurate assessment of the average flux level and, hence, time integrated exposure (fluence) experienced by the sensors may be developed from the measurements only if the sensor characteristics and the parameters of the irradiation are well known. In particular, the following variables are of interest:

1 - The measured specific activity of each sensor
2 - The physical characteristics of each sensor
3 - The operating history of the reactor
4 - The energy response of each sensor
5 - The neutron energy spectrum at the sensor location

In this section the procedures used to determine sensor specific activities, to develop reaction rates for individual sensors from the measured specific activities and the operating history of the reactor, and to derive key fast neutron exposure parameters from the measured reaction rates are described.

Determination of Sensor Reaction Rates

The specific activity of each of the radiometric sensors is determined using established ASTM procedures. Following sample preparation and weighing, the specific activity of each sensor is determined by means of gamma-ray spectrometry utilizing a high purity germanium gamma spectrometer. In the case of the surveillance capsule multiple foil sensor sets, these analyses are performed by direct counting of each of the individual wires; or, as in the case of U-238 and Np-237 fission monitors, by direct counting preceded by dissolution and chemical separation of cesium from the sensor. C26

The irradiation history of the reactor over its operating lifetime is obtained from NUREG-0020, "Licensed Operating Reactors Status Summary Report" or from other plant records. In particular, operating data are extracted on a monthly basis from reactor startup to the end of the capsule irradiation period. For the sensor sets utilized in the surveillance capsule irradiations, the half-lives of the product isotopes are long enough that a monthly histogram describing reactor operation has proven to be an adequate representation for use in radioactive decay corrections for the reactions of interest in the exposure evaluations.

Having the measured specific activities, the operating history of the reactor, and the physical characteristics of the sensors, reaction rates referenced to full power operation are determined from the following equation:

$$R = \frac{A}{N_o F Y \sum_j \frac{P_j}{P_{ref}} C_j \left[1 - e^{-\lambda t_j}\right] e^{-\lambda t_d}}$$

where:

A	=	measured specific activity (dps/gm)
R	=	reaction rate averaged over the irradiation period and referenced to operation at a core power level of P_{ref} (rps/nucleus).
N_0	=	number of target element atoms per gram of sensor.
F	=	weight fraction of the target isotope in the sensor material.
Y	=	number of product atoms produced per reaction.
P_j	=	average core power level during irradiation period j (MW).
P_{ref}	=	maximum or reference core power level of the reactor (MW).
C_j	=	calculated ratio of $\phi(E > 1.0$ MeV$)$ during irradiation period j to the time weighted average $\phi(E > 1.0$ MeV$)$ over the entire irradiation period.
λ	=	decay constant of the product isotope (sec^{-1}).
t_j	=	length of irradiation period j (sec).
t_d	=	decay time following irradiation period j (sec).

and the summation is carried out over the total number of monthly intervals comprising the total irradiation period.

In the above equation, the ratio P_j/P_{ref} accounts for month by month variation of power level within a given fuel cycle. The ratio C_j is calculated for each fuel cycle and accounts for the change in sensor reaction rates caused by variations in flux level due to changes in core power spatial distributions from fuel cycle to fuel cycle. Since the neutron flux at the measurement locations within the surveillance capsules is dominated by neutrons produced in the peripheral fuel assemblies, the change in the relative power in these assemblies from fuel cycle to fuel cycle can have a significant impact on the activation of neutron sensors. For a single-cycle irradiation, $C_j = 1.0$. However, for multiple-cycle irradiations, particularly those employing low leakage fuel management, the additional C_j correction must be utilized in order to provide accurate determinations of the decay corrected reaction rates for the dosimeter sets contained in the surveillance capsules. C26

<u>Corrections to Reaction Rate Data</u>

Prior to using the measured reaction rates in the least squares adjustment procedure discussed above, additional corrections are made to the U-238 measurements to account for the presence of U-235 impurities in the sensors as well as to adjust for the build-in of plutonium isotopes over the course of the irradiation.

In addition to the corrections made for the presence of U-235 in the U-238 fission sensors, corrections are also made to both the U-238 and Np-237 sensor reaction rates to account for gamma ray induced fission reactions occurring over the course of the irradiation.

<u>Least Squares Adjustment Procedure</u>

Least squares adjustment methods provide the capability of combining the measurement data with the neutron transport calculation resulting in a Best Estimate neutron energy spectrum with associated uncertainties. Best Estimates for key exposure parameters such as neutron fluence (E > 1.0 MeV) or iron atom displacements (dpa) along with their uncertainties are then easily obtained from the adjusted spectrum. The use of measurements in combination with the analytical results reduces the uncertainty in the calculated spectrum and acts to remove biases that may be present in the analytical technique. In general, the least squares methods, as applied to pressure vessel fluence evaluations, act to reconcile the measured sensor reaction rate data, dosimetry reaction cross-sections, and the calculated neutron energy spectrum within their respective uncertainties. C26

Values of key fast neutron exposure parameters are derived from the measured reaction rates using the FERRET least squares adjustment code (Reference 1). The FERRET approach uses the measured reaction rate data, sensor reaction cross-sections, and a calculated trial spectrum as input and proceeds to adjust the group fluxes from the trial spectrum to produce a best fit (in a least squares sense) to the measured reaction rate data. The "measured" exposure parameters along with the associated uncertainties are then obtained from the adjusted spectrum.

In the FERRET evaluations, a log-normal least squares algorithm weights both the trial values and the measured data in accordance with the assigned uncertainties and correlations. In general, the measured values, f, are linearly related to the flux, ϕ, by some response matrix A:

$$f_i^{(s,\alpha)} = \sum_g A_{ig}^{(s)} \varphi_g^{(\alpha)}$$

where i indexes the measured values belonging to a single data set s, g designates the energy group, and α delineates spectra that may be simultaneously adjusted. For example,

$$R_i \pm \delta_{R_i} = \sum_g (\sigma_{ig} \pm \delta\sigma_{ig})(\phi_g \pm \delta_{\varphi g})$$ C26

relates a set of measured reaction rates, R_i, to a single spectrum, ϕ_g, through the multigroup dosimeter reaction cross-section, σ_{ig}, each with an uncertainty δ. The log-normal approach automatically accounts for the physical constraint of positive fluxes, even with large assigned uncertainties. C26

In the least squares adjustment, the continuous quantities (i.e., neutron spectra and cross-sections) are approximated in a multi-group format consisting of 53 energy groups. The trial input spectrum is converted to the FERRET 53 group structure using the SAND-II code (Reference 2). This procedure is carried out by first expanding the 47 group calculated spectrum into the SAND-II 620 group structure using a SPLINE interpolation procedure in regions where group boundaries do not coincide. The 620 point spectrum is then re-collapsed into the group structure used in FERRET.

The sensor set reaction cross-sections, obtained from the ENDF/B-VI dosimetry file[3], are also collapsed into the 53 energy group structure using the SAND-II code. In this instance, the trial spectrum, as expanded to 620 groups, is employed as a weighting function in the cross-section collapsing procedure. Reaction cross-section uncertainties in the form of a 53 x 53 covariance matrix for each sensor reaction are also constructed from the information contained on the ENDF/B-VI data files. These matrices include energy group to energy group uncertainty correlations for each of the individual reactions.

Due to the importance of providing a trial spectrum that exhibits a relative energy distribution close to the actual spectrum at the sensor set locations, the neutron spectrum input to the FERRET evaluation is obtained from plant specific calculations for each dosimetry location. While the 53 x 53 group covariance matrices applicable to the sensor reaction cross-sections are developed from the cross-section data files, the covariance matrix for the input trial spectrum is constructed from the following relation:

$$M_{g'g} = R_n^2 + R_g R_{g'} P_{gg'}$$

where R_n specifies an overall fractional normalization uncertainty (i.e., complete correlation) for the set of values. The fractional uncertainties R_g specify additional random uncertainties for group g that are correlated with a correlation matrix given by:

$$P_{gg'} = [1 - \theta]\, \delta_{gg'} + \theta\, e^{-H}$$

where:

$$H = \frac{(g - g')^2}{2\gamma^2}$$

The first term in the correlation matrix equation specifies purely random uncertainties, while the second term describes short range correlations over a group range γ (θ specifies the strength of the latter term). The value of δ is 1 when $g = g'$ and 0 otherwise.

The use of least squares adjustment methods in LWR dosimetry evaluations is not new. The American Society for Testing and Materials (ASTM) has addressed the use of adjustment codes in ASTM Standard E944, "Application of Neutron Spectrum Adjustment Methods in Reactor Surveillance" and many industry workshops have been held to discuss the various applications. For example, the ASTM-EURATOM Symposia on Reactor Dosimetry holds workshops on neutron spectrum unfolding and adjustment techniques at each of its bi-annual conferences.

The primary objective of the least squares evaluation is to produce unbiased estimates of the neutron exposure parameters at the location of the measurement. The analytical method alone may be deficient because it inherently contains uncertainty due to the input assumptions to the calculation. Typically these assumptions include parameters such as the temperature of the water in the peripheral fuel assemblies, by-pass region, and downcomer regions, component dimensions, and peripheral core source. Industry consensus indicates that the use of calculation alone results in overall uncertainties in the neutron exposure parameters in the range of 15-20% (1σ). C26

The application of the least squares methodology requires the following input:

1. The calculated neutron energy spectrum and associated uncertainties at the measurement location.

2. The measured reaction rate and associated uncertainty for each sensor contained in the multiple foil set.

3. The energy dependent dosimetry reaction cross-sections and associated uncertainties for each sensor contained in the multiple foil sensor set.

For a given application, the calculated neutron spectrum is obtained from the results of plant specific neutron transport calculations applicable to the irradiation period experienced by the dosimetry sensor set.

This calculation is performed using the benchmarked transport calculational methodology described in the next section of this Appendix. The sensor reaction rates are derived from the measured specific activities obtained from the counting laboratory using the specific irradiation history of the sensor set to perform the radioactive decay corrections. The dosimetry reaction cross-sections and uncertainties that are utilized in LWR evaluations comply with ASTM Standard E1018, "Application of ASTM Evaluated Cross-Section Data File, Matrix E 706 (IIB)".

The uncertainties associated with the measured reaction rates, dosimetry cross-sections, and calculated neutron spectra are input to the least squares procedure in the form of variances and covariances. The assignment of the input uncertainties also follows the guidance provided in ASTM Standard E 944.

2. CALCULATION OF INTEGRATED FAST NEUTRON (E > 1.0 MEV) FLUX AT THE IRRADIATION SAMPLES

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Calculation and Dosimetry Measurement Procedures

A generalized set of guidelines for performing fast neutron exposure calculations within the reactor configuration, and procedures for analyzing measured irradiation sample data that can be correlated to these calculations, has been promulgated by the Nuclear Regulatory Commission (NRC) in Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence" (Reference 8). Since different calculational models exist and are continuously evolving along with the associated model inputs, e.g., cross-section data, it is worthwhile summarizing the key models, inputs, and procedures that the NRC staff finds acceptable for use in determining fast neutron exposures within the reactor geometry. This material is highlighted below.

The selection of a particular geometric model, the corresponding input data, and the overall methodology used to determine fast neutron exposures within the reactor geometry are based on the needs for accurately determining a solution to the problem that must be solved and the data/resources that are currently available to accomplish this task. Based on these constraints, engineering judgment is applied to each problem based on an analyst's thorough understanding of the problem, detailed knowledge of the plant, and due consideration to the strengths and weaknesses associated with a given calculational model and/or methodology. Based on these conditions, Regulatory Guide 1.190 does not recommend using a singular calculational technique to determine fast neutron exposures. Instead, Regulatory Guide 1.190 suggests that one of the following neutron transport tools be used to perform this work.

- Discrete Ordinates Transport Calculations

1. Adjoint calculations benchmarked to a reference-forward calculation, or stand-alone forward calculations.

2. Various geometrical models utilized with suitable mesh spacing in order to accurately represent the spatial distribution of the material compositions and source.

3. In performing discrete ordinates calculations, Regulatory Guide 1.190 also suggests that a P3 angular decomposition of the scattering cross-sections be used, as a minimum.

4. Regulatory Guide 1.190 also recommends that discrete ordinates calculations utilize S_8 angular quadrature, as a minimum.

5. Regulatory Guide 1.190 indicates that the latest version of the Evaluated Nuclear Data File, or ENDF/B, should be used for determining the nuclear cross-sections; however, cross-sections based on earlier or equivalent nuclear data sets that have been thoroughly benchmarked are also acceptable. C26

- Monte Carlo Transport Calculations

A complete description of the Westinghouse pressure vessel neutron fluence methodology along with the SER documenting NRC staff approval of the method and computer codes are provided in Reference 9.

Plant-Specific Calculations

The most recent fast (E > 1.0 MeV) neutron fluence evaluations for each of the Turkey Point reactor pressure vessels was based on a 2D/1D synthesis of neutron fluxes that were obtained from a series of plant- and cycle-specific forward discrete ordinates transport calculations run in R-θ, R-Z, and R geometric models. The set of calculations, which assessed dosimetry as part of the reactor vessel surveillance program and pressure vessel neutron fluences, were conducted in accordance with the guidelines that are specified in Regulatory Guide 1.190.

3. MEASUREMENT OF THE INITIAL NIL-DUCTILITY TRANSITION (NDT) TEMPERATURE OF THE REACTOR PRESSURE VESSEL WELDMENTS, BASE PLATE AND FORGINGS MATERIAL

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The unirradiated or initial nil-ductility transition temperature of the pressure vessel weldments, base plate and forgings material was 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 nil-ductility transition (NDT) temperature is defined in ASTM E208 as "the maximum temperature where a standard drop-weight specimen breaks when tested according to the provisions of this method". Using the Charpy V-notch test, the NDT temperature was defined as the temperature at which the energy required to break the specimen is a certain "fixed" value. For SA 302B and A508 Class 2 steel the ASME III Table N-421 specified an energy value of 30 ft-lb. This value was based on a correlation with the drop weight test and referred to as the "30 ft-lb-fix". A curve of the temperature versus energy absorbed in breaking the specimen was plotted. To obtain this curve, 15 tests were performed which included three tests at five different temperatures. The intersection of the energy versus temperature curve with the 30 ft-lb ordinate was designated as the NDT temperature.

For weld materials, Framatome ANP Topical Report BAW-2308, Revisions 1A and 2A (References 6 and 7) give the initial (Linde 80) material properties (RT_{NDT} and σ_i). These properties were obtained by performing fracture toughness testing based on the application of the "Master Curve" evaluation procedure. The Master Curve evaluation procedure permits data obtained from sample sets tested at different temperatures to be combined, as the basis for redefining the initial, unirradiated material properties of Linde 80 welds. Guidelines for the application of the Master Curve evaluation methodology used in Topical Report BAW-2308, Revision 1A were given in the 1997 and 2002 Editions of ASTM Standard Test Method E 1921 (ASTM E 1921) "Standard Test Method for Determination of Reference Temperature, T_o, for Ferritic Steels in the Transition Range". Revised initial values were given in Topical Report BAW-2308, Revision 2A (supplemental). Additional guidance on the application of reference temperature values based on Master Curve evaluation to the establishment of reactor pressure vessel material properties for regulatory applications was provided by ASME Code Case N-629, "Use of Fracture Toughness Test Data to Establish Reference Temperature for Pressure Retaining Materials of Section III, Division 1, Class 1". The B&W Owner Group's motivation for pursuing this option of using a Master Curve based approach to evaluate Linde 80 welds is related to the fact that, due to their generally low Charpy V-notch upper shelf energy behavior, the testing specified in ASME Code, Section III, Paragraph NB-2331 has shown to be overly conservative when used to predict the transition from ductile to brittle failure in Linde 80 welds. C26

The available data indicate differences as great as 40°F between curves plotted through the minimum and average values respectively. The determination of the NDT temperature from the average curve was considered representative of the material and was consistent with procedures as specified in ASTM E23. In assessing the NDT temperature shift due to irradiation, the translation of the average curve was used.

As part of the Westinghouse surveillance program referred to above, Charpy V-impact tests, tensile tests, and fracture mechanics specimens were 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 NDT temperature of the base plate were:

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 were based on the averaging of data. The Charpy V-notch 30 ft-lb "fix" temperature was 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 was 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 were observed in comparing curves plotted through the minimum and average values, respectively. The value of the NDT temperature derived from the average curve was 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 RT_{NDT} 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 would be used.

There are quantitative differences between the RT_{NDT} at the surface, 1/4 thickness, or the center of a plate.

The 1/4T location is considered conservative, since the enhanced metallurgical properties of the surface are not used for the determination of RT_{NDT}. In addition, the limiting RT_{NDT} for the reactor vessel after operation will be based on the RT_{NDT} shift due to irradiation. Since the fast neutron dose is highest at the inner surface, usage of the 1/4T RT_{NDT} criterion is conservative.

To assess any possible uncertainties in the consideration of the RT_{NDT} shift for welds heat affected zone, and base metal, test specimens of these three "material types" have been included in the reactor vessel surveillance program.

REFERENCES

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2. McElroy, W. N., et. al., "A Computer-Automated Iterative Method of Neutron Flux Spectra Determined by Foil Activation," AFWL-TR-67-41, Volumes I-IV, Air Force Weapons Laboratory, Kirkland AFB, NM, July 1967.

3. RSIC Data Library Collection DLC-178, "SNLRML Recommended Dosimetry Cross-Section Compendium," July 1994.

4. RSIC Computer Code Collection CCC-650 DOORS 3.2, "ORNL RSICC CCC-650 DOORS 3.2 One, Two and Three-Dimensional Discrete Ordinates Neutron/Photon Transport Code System," Radiation Safety Information Computational Center, Oak Ridge National Laboratory, April 1998.

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5. RSICC Data Library Collection DLC-185, "BUGLE-96 Coupled 47 Neutron, 20 Gamma-Ray Group Cross-Section Library Derived from ENDF/B-VI for LWR Shielding and Pressure Vessel Dosimetry Applications," Radiation Safety Information Computational Center, Oak Ridge National Laboratory, July 1999.

6. Framatome ANP Topical Report BAW-2308, Revision 1A, "Initial RT_{NDT} of Linde 80 Weld Materials", Approved August 2005.

7. Framatome ANP Topical Report BAW-2308, Revision 2A, "Initial RT_{NDT} of Linde 80 Weld Materials", Approved March 2008.

8. Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," United States Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, March 2001.

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9. Andrachek, J. D., "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," WCAP-14040-A, Revision 4, May 2004.

APPENDIX 4B

PROCEDURE FOR PLUGGING A TUBE IN A STEAM GENERATOR

Inspection and repair of defective steam generator tubes is governed by approved plant procedures. A typical sequence is as follows:

1. The reactor is shutdown and taken to cold shutdown condition; i.e., both primary and secondary sides are depressurized and cold. Decay heat is removed via the residual heat removal system.

2. The reactor coolant level is lowered until the level is between the bottom of the steam generator and the hot leg elbow, thus maintaining the remainder of the hot leg between the elbow and the vessel full of water.

   Lowering of water to this level does not affect operation of the residual heat removal system because the residual heat removal suction line is connected to the hot leg of loop C for Unit 3 and loop A for Unit 4, and the return line is connected to the cold legs of all three loops.

3. The steam generator is entered via the two manways, one on either side of the channel head partition plate. Prior to the performance of any work, the area around the steam generator is monitored to determine the radiation level. In the event of high radiation levels, biological shielding is installed around the coolant channel head, and portable respiratory apparatus is used if required. Temporary nozzle covers are placed over the inlet and outlet reactor coolant legs to the steam generator to prevent any debris from entering the reactor coolant system.

4. The defective tube is located and plugged. Remotely operated equipment may be used to locate and plug the defective tube. Tubes with indications may be plugged or plugged and staked. Remotely operated equipment may also be used to perform an in-situ pressure test on the defective tube prior to performing the repair activity.

5. The temporary nozzle covers are removed and the manway covers are replaced thus resealing the system.

6. The reactor coolant level is raised to its normal cold shutdown level, and the air which has been introduced into the steam generator is vented n the normal manner i.e., in the same way as following a refueling shutdown.

APPENDIX 4C

REPLACEMENT STEAM GENERATOR DESIGN

In 1982 and 1983 the Unit 3 and Unit 4 steam generator lower assemblies were replaced. The new assemblies match the design performance of the original assemblies. However, several design improvements have been made. This Appendix 4C describes the design parameters of the new assemblies. The pages herein are taken from Chapter 2 of the Steam Generator Repair Report as amended, which was submitted to the NRC under FPL letter number L-77-296, dated September 20, 1977.

1.0 REPLACEMENT COMPONENT DESIGN

Westinghouse has fabricated new steam generator lower assemblies as illustrated by Figure 4C-1. The design of the lower assemblies matches the design performance of the lower assemblies being replaced. However, several design improvements that do not alter mechanical, performance and FSAR parameters are included in the design. These design features will improve flow distribution, improve tube bundle access and reduce secondary side corrosion. This section discusses the design and manufacture of the lower assemblies.

1.1 COMPARISON WITH EXISTING COMPONENT DESIGN

1.1.1 Parametric Comparison

The steam generators for the Turkey Point plants, upon completion of the repair, have physical, mechanical and thermal characteristics consistent with the original design and safety analysis as currently documented in the FSAR. The existing steam generators were built to the 1965 edition of the ASME Boiler and Pressure Vessel Code (ASME Code); the new component parts of the steam generators are designed and fabricated based upon the 1974 edition of the ASME Code, including all addenda through Summer 1976. The Stress Report is based upon the 1965 edition of the ASME Code, including all addenda through Summer 1965. The replacement lower assemblies were fabricated and analyzed to standards equivalent to the original units.

The replacement lower assembly incorporates a number of refinements in design which are discussed in Section 1.2. During 1975 several modifications were made to the installed steam generators to increase performance and promote reliability. These modifications (described and noted in the text) were retained or improved with the replacement lower assemblies. The modifications accomplished at that time consisted of removing the downcomer resistance plate, improving the moisture separators, modifying the blowdown arrangement inside the steam generators, installing tube lane blocking devices and modifications to the feedring to improve performance. These modifications increased the circulation ratio and improved the units' ability to resist sludge build-up.

Design data for the steam generators is presented in Table 4C-1 allowing comparison between the present steam generators and the repaired units. Improvements have been made for increased access to the secondary side of the steam generators incorporating six 6-inch hand holes around the bundle in the tube sheet area. The thermal data for each steam generator remains the same as the original steam generator.

Since the replacement lower assemblies have been designed to incorporate changes based on field experience, a number of minor changes in specific components have been made which could affect the thermal hydraulic performance of the unit. In order to maintain the original thermal and hydraulic conditions, adjustment of heat transfer surface parameters was necessary; changes in the support plate configuration and desire to improve the circulation ratio resulted in a decrease in the number of tubes. These modifications resulted in the reactor coolant water volume in the steam generator being reduced slightly, the secondary side volume being increased slightly, a slight decrease in the amount of heat transfer surface area, as well as a slight increase in the heat transfer coefficient. Imposing closer manufacturing tolerances on the tube wall thickness results in an increase in the overall heat transfer coefficient (approximately 2.5%) for the repaired units. This increase in heat transfer coefficient offsets the decrease in heat transfer area (approximately 2.2%) so that steam generator heat transfer remains essentially unchanged.

Materials used in the fabrication of the replacement lower assemblies were procured to the requirements of the 1974 edition of the ASME Code, including all addenda through Summer 1976. These materials are identical to those used

in the original steam generators except where specific design changes have been recently incorporated or fabrication practice has changed. Specific examples of these occurrences are enumerated as follows: plate material used in the secondary shell formation has been changed to SA-533 Grade A Class 2 from SA302 Grade B Class 1 as a result of fabrication practices; support plate material has been changed to SA-240 Type 405 from SA-285 Grade C as a result of design changes to prevent corrosion. Material changes due to design improvements do not degrade the physical, mechanical and thermal properties of the steam generators. Further discussion is provided in Section 1.2 and Table 4C-2 enumerates past and present applications of materials.

1.1.2 Physical Compatibility With Existing Steam Generators and Systems

New steam generator lower assemblies (see Figure 4C-1 were provided. These lower assemblies are designed to be identical physical replacements for the existing units. Outside overall dimensions are the same as are the location of the nozzles and support attachments. Interfaces between the steam generators and plant components and systems are maintained. Dry and wet weights of the steam generators remain approximately the same as are the center of gravity; therefore, no changes to the present supports or their configuration are necessary.

1.1.3 ASME Code Application

The present operating steam generators were designed and constructed to the requirements of the 1965 edition of the ASME Code, Section III, Summer 1965 addenda. The replacement assemblies have been fabricated to the requirements of the 1974 edition of the ASME Code including all addenda through Summer 1976. Design of the steam generators is consistent with the original design of the reactor coolant system as well as the upper shell assembly of the steam generators which were not replaced. Materials to be used in fabrication were procured to the requirements of the current codes to facilitate construction. All material certification tests were performed and recorded as required by
current versions of the code. None of the requirements imposed on the replacement assemblies inhibit the capability of the steam generators to meet performance and FSAR safety requirements.

1.1.4 Regulatory Guide Application

The compilation below addresses Regulatory Guides considered applicable to the fabrication of the replacement lower assemblies. It must be noted that these guides were issued subsequent to construction and operation of this facility. The intent was to accommodate, consistent with facility design and repair program objectives, the guidance by these regulatory guidelines.

1.26 Quality Group Classifications and Standards for Water, Steam and Radioactive-Waste-Containing Components of Nuclear Power Plants (Rev.3, February 1976).

Westinghouse utilizes the classification system ANSI N18.2A-1975 for water and steam containing components. This classification method assigns safety-related components to safety classes. Assignment of the primary side of the steam generator to Safety Class 1 and the secondary side to Safety Class 2 is consistent with the quality groupings which would result from this regulatory guide and 10 CFR 50.55a.

1.28 Quality Assurance Program Requirements (Design and Construction)

(Safety Guide 28, June 1972)

Westinghouse position on Regulatory Guide 1.28 is presented in WCAP-8370, "WRD Quality Assurance Plan". For activities which occurred during the period from January 1, 1975 to September 30, 1977, the position is presented in WCAP-8370, Revision 7A. For activities occurring on or after October 1, 1977, the position is presented in WCAP-8370, Revision 8A.

1.31 Control of Stainless Steel Welding (Rev. 1, June 1973)

The Westinghouse production weld verification program, as described in WCAP-8324-A, was approved by the NRC as a satisfactory substitute for following the recommendations of the NRC Interim Position on Regulatory Guide 1.31 (4/74). The results of the verification program support the hypothesis presented in WCAP-8324-A; these results have been summarized and documented in WCAP-8693, which has been submitted to the NRC for information.

1.34 Control of Electroslag Weld Properties (December 28, 1972)

Where electroslag welding is used, Westinghouse requires its suppliers to follow the recommendations of this guide.

1.37 Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants (March 16, 1973)

The Westinghouse position on Regulatory Guide 1.37 is presented in WCAP-8370, "WRD Quality Assurance Plan." For activities which occurred during the period from January 1, 1975 to September 30, 1977, the position is presented in WCAP-8370, Revision 7A. For activities occurring on or after October 1, 1977, the position is presented in WCAP-8370, Revision 8A.

1.38 Quality Assurance Requirements for Packaging, Shipping, Receiving Storage, and Handling for Nuclear Power Plants (Rev. 2, May 1977)

The Westinghouse position of Regulatory Guide 1.38 is presented in WCAP-8370, "WRD Quality Assurance Plan". For activities which occurred during the period from January 1, 1975 to September 30, 977, the position is presented in WCAP-8370, Revision 7A. For activities occurring on or after October 1, 1977, the position is presented in WCAP-8370, Revision 8A.

1.43 Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components (May 1973)

The Westinghouse Tampa Division uses materials made to fine-grain practice or which are not susceptible to underclad cracking. These materials do not require the controls listed in the guide.

1.44 Control of the Use of Sensitized Stainless Steel (May 1973)

All of the unstabilized austenitic stainless steels used for component parts of the reactor coolant pressure boundary are utilized in the final heat treated condition required by the respective ASME Code, Section II, material specification for the particular type or grade of alloy. Processing and fabrication are performed using established methods and techniques to avoid sensitization. Westinghouse has verified that these practices will prevent sensitization by performing corrosion tests on as-received wrought materials, as well as on production and qualification weldments. In addition, the water chemistry in the reactor coolant system is controlled to prevent intergranular attack of unstabilized stainless steels; the effectiveness of these controls has been demonstrated by both laboratory tests and operating experience.

1.48 Design Limits and Loading Combinations for Seismic Category I Fluid System Components (May 1973)

Westinghouse meets the requirements of General Design Criterion 2 and will thereby satisfy the concerns of Regulatory Guide 1.48. The loading combinations and design limits used in the code stress analysis of the steam generator are the same as those in the Turkey Point FSAR.

1.50 Control of Preheat Temperature for Welding of Low-Alloy Steel (May 1973)

Westinghouse practices are in agreement with Regulatory Positions C.1.a, C.3 and C.4. For Regulatory Position C.1.b, Westinghouse qualifies welding procedures within the preheat temperature ranges required by Section IX of the ASME Code. For Regulatory Position C.2, Westinghouse uses the methods documented in WCAP-8577-A, which has been accepted by the NRC.

1.58 Qualification of Nuclear Power Plant Inspection, Examination, and Testing Personnel (August 1973)

The Westinghouse position on Regulatory Guide 1.58 is presented in WCAP-8370, "WRD Quality Assurance Plan". For activities which occurred during the period from January 1, 1975, to September 30, 1977, the position is presented in WCAP-8370, Revision 7A. For activities occurring on or after October 1, 1977, the position is presented in WCAP-8370, Revision 8A.

1.64 Quality Assurance Requirements for the Design of Nuclear Power Plants (Rev. 1, February 1975) The Westinghouse position on Regulatory Guide 1.64 is presented in WCAP-8370, WRD Quality Assurance Plan". For activities which occurred during the period from January 1, 1975 to September 30, 1977, the position is presented in WCAP-8370, Revision 7A. For activities occurring on or after October 1, 1977, the position is presented in WCAP-8370, Revision 8A.

1.66 Nondestructive Examination of Tubular Products (October 1973) Steam generator nozzles are either radiographed or ultrasonically tested in the circumferential and axial directions in accordance with the guides' positions. Steam generator tubing receives eddy current, circumferential ultrasonic testing, and hydrostatic testing to satisfy the guides' recommendations.

1.71 Welder Qualification for Areas of Limited Accessibility (December 1973)

Westinghouse practice does not require qualification of welders for areas of limited accessibility. Shop welds are repetitive and closely supervised and the ASME Code, Sections III and IX requirements are followed.

1.83 Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes (Rev. 1, July 1975)

Westinghouse steam generators are designed to permit access to tubes for inspection and plugging. A pre-service inspection of the steam generators was conducted to establish baseline conditions.

1.84 Code Case Acceptability - ASME Section III Design and Fabrication (June 1974) (Rev. 1, April 1975) (Rev. 2, June 1975) (Rev. 3, September 1975) Rev. 4, November 1975) (Rev. 5, February 1976) (Rev. 6, May 1976) (Rev. 7, August 1976) (Rev. 8, November 1976) (Rev. 9, March 1977)

1.85 Code Case Acceptability - ASME Section III Materials (June 1974) (Rev. 1, April 1975) (Rev. 2, June 1975) (Rev. 3, September 1975) (Rev. 4, November 1975) (Rev. 5, February 1976) (Rev. 6, March 1976) (Rev. 7, August 1976) (Rev. 8, November 1976) (Rev. 9, March 1977)

1. Westinghouse controls its suppliers to:

   a. Limit the use of code cases to those listed in Regulatory Position C.1 of the applicable guide revision in effect at the time the equipment is ordered, except as allowed in item 2 below.

   b. Identify and request permission for use of any code cases not listed in Regulatory Position C.1 of the applicable guide revision in effect at the time the equipment is ordered, where use of such cases is needed by the supplier.

   c. Allow continued use of a code case considered acceptable at the time of equipment order, where such code case was subsequently annulled or amended.

2. Westinghouse seeks NRC permission for use of code cases needed by suppliers and not yet endorsed in Regulatory Position C.1 of the applicable guide revision in effect at the time the equipment is ordered and permits supplier use only if NRC permission is obtained or is otherwise assured (e.g., a later version of the regulatory guide includes endorsement)

1.88 Collection, Storage and Maintenance of Nuclear Power Plant Assurance Records (Rev. 2, October 1976)

The Westinghouse position on Regulatory Guide 1.88 is presented in WCAP-8370, Revision 7A. For activities occurring on or after October 1, 1977, the position is presented in WCAP-8370, Revision 8A.

1.123 Quality Assurance Requirements for Control of Procurement of Items and Service for Nuclear Power Plants (Revision 1, July 1977)

The Westinghouse position on Regulatory Guide 1.123 is presented in WCAP-8370, "WRD Quality Assurance Plan". For activities which occurred during the period from January 1, 1975 to September 30, 1977, the position is presented in WCAP-8370, Revision 7A. For activities occurring on or after October 1, 1977, the position is presented in WCAP-8370, Revision 8A.

2.2 COMPONENT DESIGN IMPROVEMENTS

As noted, the physical, thermal and hydraulic characteristics of the steam generators essentially duplicate those of the original units. However, design changes which do not alter FSAR safety requirements have been incorporated in the design. These changes increase the operating availability and improve resistance to corrosion of the secondary side thereby minimizing the potential for future repair efforts. Figure 4C-1 illustrates some of these improvements. It should be noted that some of these features have been installed in the in situ units (see Section 1.1.1).

Research, development and testing have been utilized to select design parameters, material and component configurations which will prevent degradation of the repaired steam generators. Confirmatory tests in model boilers and other tests on the material and component configuration are continuing.

1.2.1 <u>Design Refinements To Prevent And Inhibit Corrosion</u>

1.2.1.1 Increased Circulation Ratio

Circulation ratio is defined as the total tube bundle flow divided by the feedwater flow and is inversely proportional to the steam quality exiting the tube bundle. As the circulation ratio increases, certain parameters of the steam generator, such as lateral velocity sweeping across the tubesheet, steam quality, void fraction and number of tubes exposed to sludge, change in a favorable direction. Low steam quality in the bundle reduces tube exposure to local steam blanketing. This also reduces the number of potential areas of concentration for chemical impurities. In addition, higher circulation ratios increase the fluid velocity sweeping across the tubesheet to the center of the bundle. Specific design changes, such as the quatrefoil plates (See Subsection 1.2.1.8), modification in the tube bundle size and wrapper to shell distance, influence the circulation ratio.

1.2.1.2 Flow Distribution Baffle

A flow distribution baffle has been provided 18 inches above the tubesheet. This baffle has a cut out center section and oversized drilled tube holes. The increased circulation ratio provides a greater lateral flow across the tubesheet surface. The baffle plate will assist in directing this flow across the tubesheet then up the center of the bundle through the center cutout. The design is sized to minimize the number of tubes exposed to sludge. Consistent with this purpose, the design causes the sludge to deposit in and near the center of the bundle at the blowdown intake. The flow distribution baffle plate material is ferritic stainless steel. Figure 4C-2 illustrates the flow distribution baffle.

While the baffle will direct flow toward the center of the bundle, the average velocity around the tubes will be sufficient to prevent sludge from settling. In addition, as noted, access holes have been provided to allow sludge lancing of the baffle plate.

1.2.1.3 Improved Internal Blowdown Design

Each steam generator was designed to have two 2-inch schedule 40 Inconel internal blowdown pipes. The blowdown rate from the steam generators is varied as required by chemistry conditions in the feedwater and as monitored in the blowdown. Maintenance of the steam side water chemistry is assisted through the use of the blowdown system. Continuous blowdown of the steam generator provides a dynamic system which is constantly removing impurities from the steam generator. During hot standby and hot functional testing, blowdown is employed, as needed, to maintain the steam generator chemistry within specification. The blowdown intake location is coordinated with the baffle plate design so that the maximum intake is located where the greatest amount of sludge is expected to deposit. The improved blowdown system allows higher capacity blowdown in comparison with the present blowdown arrangement.

1.2.1.4 Tube Expansion in Tubesheet

Following insertion into the tubesheet hole, tack rolling, welding and gas leak testing, the tubes are expanded to the full depth of the tubesheet hole. Full-depth expansion prevents crevice boiling. In addition it prevents a buildup of impurities from forming in the crevice region. The present steam

generator tubes were only partially expanded in the tubesheet.

1.2.1.5 Thermally Treated Inconel 600 Tubing

Research by Westinghouse has determined that significant improvement in the stress corrosion resistance of Inconel 600 tubing can be achieved by modification of the metallurgical structure through thermal treatment. The primary objective of this treatment is to develop an improved metallurgical structure, associated with grain boundary precipitate morphology, which provides increased margin with respect to stress corrosion performance. Several benefits result from this treatment such as improved resistance to stress corrosion cracking in NaOH, resistance to intergranular attack in oxygenated environments, resistance to intergranular attack in sulphur-containing species and reduction of residual stress imparted by tube processing.

Studies conducted at Westinghouse and elsewhere have indicated that certain heat treatments can improve caustic stress corrosion resistance but result in a chromium-depleted grain boundary layer (sensitization) which is not as resistant to off-chemistry environments, should they be experienced. However, analysis of available data also indicates that there is a broad band of temperature and time within the typical sensitization range for Inconel 600 which provides improved resistance to stress corrosion cracking in both caustic and pure water environments. Thermal treatment in this time-temperature band avoids formation of the chromium depleted grain boundary layer. The thermal treatment to be used was within this time-temperature band.

1.2.1.6 Offset Feedwater Distribution

Previously, feedwater flow within the steam generators was modified so that 80 percent of the flow is directed to the hot leg side of the bundle and 20 percent of the flow is directed to the cold leg side of the bundle. This reduces the steam quality in the hot leg side of the bundle and raises the steam quality in the cold leg side of the bundle. The effect of these changes in steam quality is to shift the point of highest steam quality at the tubesheet elevation toward the center of the bundle. The point of highest steam quality has the lowest density and is, therefore, a likely region for chemical concentration and sludge deposition. This area is utilized for location of blowdown intake. Feedwater flow distribution is accomplished by providing a greater number of flow paths on the portion of the feedwater ring which traverses the hot leg side of the tube bundle. These modifications were maintained in the replacement assemblies.

1.2.1.7 Corrosion Resistant Support Plate Material

Corrosion in the crevice between the tube and tube support plate has led to denting of the tubing in that area and in some cases affected the steam generator performance in general. Alternative support plate materials have been evaluated, and SA-240 Type 405 ferritic stainless steel has been selected as the optimum material for this application. This material is ASME Code approved and is resistant to corrosion with the chemistry expected during the operation of the steam generator. In addition, SA-240 has a low wear coefficient when paired with Inconel and has a coefficient of thermal expansion similar to carbon steel. Corrosion of SA-240 results in an oxide which has approximately the same volume as the parent material, whereas corrosion of carbon steel results in oxides which have approximately two times the volume of the parent material. Type 405 also has material properties such as machinability and weldability which are comparable to carbon steel. In addition to the tube support plates, the baffle plate (discussed in Subsection 1.2.1.2) was constructed of SA-240 Type 405.

1.2.1.8 Quatrefoil Tube Support Plates

The quatrefoil tube support plate design, illustrated by Figure 4C-3, consists of four flow lobes and four support lands. The lands provide support to the tube during all operating conditions, while allowing flow around the tube. This design has a lower pressure drop than the most current circulation hole designs. This low secondary pressure drop increases the circulation ratio which, when combined with other improvements, translates into higher sweeping velocities and fewer tubes exposed to a low steam quality at the tubesheet. This design directs the flow along the tubes which limits steam formation and chemical concentrations at the tube-to-tube support plate intersections. The quatrefoil support place design results in higher average velocities along the tubes, preventing sludge deposition. The combination of higher velocities in the support plate region and corrosion resistant material will minimize the possibility of support plate corrosion.

1.2.2 Design Refinements To Improve Performance

In the course of evolution of the steam generator design, as derived from operating experience and ongoing research and development programs, certain improvements and refinements have been incorporated in recent designs to improve performance of thermal hydraulic characteristics. These improvements are included in the FPL design and are discussed below. They do not alter FSAR safety requirements.

1.2.2.1 Recessed Tube to Tubesheet Weld

The tubes on the replacement lower assemblies were recessed slightly into the tubesheet holes and then welded to the tubesheet cladding. Elimination of the protruding tube stub of the original design results in lower entry pressure losses and, therefore, a lower pressure drop in the primary loop. In addition, a possible point of crud buildup and corrosion is likely avoided with this design. This is illustrated in Figure 4C-4.

1.2.2.2 Tube Lane Blocking Device

Recirculating water exiting at the bottom of the wrapper will tend to preferentially channel to the tube lane and bypass part of the tube array. In order to prevent this tube bundle bypass, a series of plates were installed in the tube lane during prior modifications. These plates are arrayed so that there will be minimal interference with sludge lancing. These blocking devices were retained in the replacement units.

1.2.2.3 Moisture Separator Improvements

Since the circulation ratio in the steam generator has increased, the duty for the moisture separator equipment will increase. To accommodate this increase, several improvements were incorporated. New demister vanes were installed to increase the efficiency of the moisture separators. Perforated plates were installed on the face of the demister vane housing to distribute the flow evenly through the demisters and provide better moisture separating. The swirl vane barrels previously modified with optimized orifice plates were realigned. These improvements are shown in Figure 4C-5.

1.2.3 Design Changes To Improve Maintenance And Reliability

Operational experience, including necessary maintenance and repair, has resulted in certain changes in design which are directed to improving the maintainability and ultimately the reliability of the units. Other changes have been incorporated to prevent occurrences of operational problems which have been experienced. These changes are discussed below and do not alter performance or FSAR safety requirements.

1.2.3.1 Access Ports

The lower assemblies were constructed with additional access ports. Four 6-inch access ports are located slightly above the tube sheet, approximately 90 degrees apart, with two located on the tube lane. Two 6 inch access ports are located on the tube lane, between the flow distribution baffle and the first tube support plate. The addition of these access ports improve and promote inspection of the tube sheet and flow distribution baffle and assist in sludge lancing.

1.2.3.2 Wet Layup Nozzle

A 2-inch nozzle was added to the upper shell to facilitate the wet layup of the steam generators during periods of inactivity. The wet layup nozzle can be used for addition of chemicals during these periods to prevent any excursions of the water quality in the steam generator. The nozzle can also be used in conjunction with other systems to circulate water through the steam generator during periods of layup.

1.2.3.3 Primary Shell Drain

A 3/8 inch primary shell drain is included in the channel head to improve drainage of the channel head. The improved drainage will lessen downtime and facilitate any maintenance or inspection to be conducted in the channel head.

1.2.3.4 Primary Closure Rings

Closure rings were welded inside the channel head at the base of each primary nozzle so that closure plates can be installed during primary chamber maintenance. This design allows the plates to be bolted to the rings for quick installation and removal. Closure plates allow maintenance or inspection to be conducted in the channel head with the reactor cavity flooded.

1.3 SHOP TESTS AND INSPECTIONS

The tests and inspections required by the ASME Code, Section III were conducted during the fabrication of the steam generator lower assembly. In addition to the ASME requirements, further tests and inspections were conducted at the fabrication facility. The primary side of the steam generator was hydrotested at the shop in accordance with approved procedures. Each tube was individually hydrotested prior to use in fabrication. After the tube bundle installation is completed, a gas leak test was performed to demonstrate the integrity of the tube-to-tubesheet welds.

1.4 Onsite Storage Facility

A temporary storage facility provided for the storage of the steam generator lower assemblies. The lower assemblies are stored in this area until they can be shipped offsite to a licensed land burial site or decommissioned with the plant. Prior to removal from the containment, the openings in the lower assemblies were sealed to prevent the release of radioactivity during transfer and subsequent on-site storage.

The only radiological consideration associated with storage is the direct radiation from the steam generators. Shielding is provided to ensure acceptable radiation levels external to the storage facility. There are no accident considerations associated with on-site storage. Based on the above considerations, the required storage facility design criteria are:

a. Appropriate shielding for direct dose

b. Provisions for periodic surveillance of steam generator seal integrity

c. Total enclosure of the sealed steam generators is not required.

The facility is founded on engineered fill at finished grade elevation +17'-6" MLW in the area approximately 150 feet south of the ash disposal pits and 290 feet east of the Radwaste Building (Figure 4C-6). The elevation of the area ranges from +6 to +9 feet MLW. At the storage facility location, the former surface layer consisted of 4 feet of limerock fill, underlain by about 6 feet of muck. Beneath the muck, Miami limestone extends 20 feet, underlain by Key Largo limestone to about elevation -100 feet.

Prior to construction of the facility, the existing limerock fill and muck were removed from below the potential zone of influence of the building foundation, and replaced with compacted crushed limerock fill up to elevation +17'-6" MLW. The existing muck and fill was excavated to a minimum distance of 15 feet beyond the edge of the building. The building is at least 65 feet back from the top of the compacted crushed limerock fill boundary slope. This slope is 1-vertical on 3-horizontal.

The crushed limerock fill which supports the facility was quarried from local Miami and Key Largo limestone formations. Maximum size is about 6 inches, with up to 20 percent passing the No. 200 sieve. The crushed limerock was stockpiled on-site to drain effectively to the optimum moisture range between 7 percent and 14 percent. The fill was placed using a maximum loose thickness of 12 inches. The fill was compacted with a vibrating drum roller to obtain a minimum dry density of 110 lb. per cubic foot. An extensive series of laboratory tests on the crushed limestone compacted to 110 lb. per cubic foot dry density has indicated effective strength parameters of conservatively 3 kips per square foot for cohesion and 39 degrees for internal angle of friction.

The compacted crushed limerock fill provides an allowable bearing capacity over 5 kips per square foot, including a factor of safety of 3.

The Miami and Key Largo limestone formations underlying the compacted crushed limerock fill have allowable bearing capacities over 30 kips per square foot

(including a factor of safety of 3).

The fill area was designed for a 100 year flood level of +12.7 feet MLW as the design storm surge required for buildings in southern Florida. This 100 year flood level is per the Code of Metropolitan Dade County Florida.

The steam generator storage facility was designed in accordance with the following current codes and standards:

South Florida Building Code

Code of Metropolitan Dade County Florida

Building Code Requirements for Reinforced Concrete (ACI 318)

American Institute of Steel Construction Manual of Steel Construction and Specification of the Design, Fabrication and Erection of Structural Steel for Buildings

American Welding Society Structural Welding Code (AWS D1.1)

The facility was designed for a hurricane wind velocity of 120 miles per hour with application of shape factors in accordance with the American Society of

Civil Engineers Paper No. 3269.

The structure consists of 2'0" thick reinforced exterior concrete shield walls sized to maintain a direct gamma dose rate of less than or equal to 2.5 mr/hr at exterior wall surfaces. The facility is approximately 130 feet by 42 feet with centerline along its length oriented in the East to West direction. A 2'-0" thick reinforced interior concrete wall was provided through the center of the structure for the full length of the facility. The interior wall provides roof support and separation between the Unit 3 and Unit 4 steam generator lower assemblies. All walls are founded on continuous strip footings with bases at approximately elevation +15'-6" MLW, 2 feet below the finished grade elevation. A maze shielded entrance with a door through the exterior wall is provided to allow for periodic surveillance of lower assembly seal integrity. Each lower assembly weighing approximately 186 tons (Specific Gravity ≈ 1.88) with two steel support saddles is stored in the facility on reinforced concrete bearing pads. A 6 inch thick reinforced concrete floor is provided. Top of floor and pad elevation is +18'-0" MLW.

Wall footings and bearing pads were designed to maintain a maximum allowable

soil bearing pressure of 5000 pounds per square foot.

Other design loads for the structure are in accordance with the South Florida Building Code. Concrete has a minimum compressive strength of 3000 psi at 28 days. Reinforcing steel and structural steel is in accordance with ASTM A615 (Grade 60) and ASTM A36 respectively. Concrete and steel allowable stresses are in accordance with Building Code Requirements for Reinforced Concrete (ACI 318) and Specification for the Design, Fabrication and Erection of Structural Steel for Buildings (AISC).

The design of the wall thickness was determined using a point-kernal computer code which used semi-empirical methods developed by Rockwell[1] for calculating the direct gamma dose rates from a homogeneous volumetric cylindrical source through slab shields.

The values of the source terms for the analysis were based on the results of a field survey of a steam generator in a drained condition one month after shutdown. For conservatism it was assumed that all short-lived isotopes had

decayed away and the sole contributor to the measured dose rate was cobalt-60, which has the highest average gamma ray energies and is therefore the most
difficult to shield for a given curie level. The results of the conservative analysis indicate that 24-inch concrete walls are required to meet the dose criteria of 2.5 mr/hr at the exterior wall surfaces. However the dose at the exterior wall surface is expected to be at or below 0.25 mr/hr.

The skyshine analysis was performed with an industry-recognized computer code G^3[2] based upon the same field survey of a steam generator, assuming the average energy of Co^{60} as that of the source. The source strength of the isotropic point source was determined by calculating a normalization constant equal to the total photon leakage from the steam generator. The skyshine contribution, without taking credit for a shielding roof, will not increase the dose rate outside the compound over 0.25 mr/hr.

The resulting dose equivalent to an individual at the north site boundary location for a full year was calculated assuming 2.5 mr/hr at the outside surfaces of the storage compound, plus the skyshine contribution assuming no

roof on the storage facility. The calculated dose was 5.2×10^{-3} mrem which is considered an insignificant contribution of the offsite dose. The presently proposed facility location (see Figure 4C-6) was assumed for the aforementioned analysis.

(1) T. Rockwell, <u>Reactor Shielding Design Manual</u>, D. Van Nostrand Co., New York (1956).

(2) R. E. Malefant <u>G^3: A General Purpose Gamma-Ray Scattering Program</u>, Los Alamos Scientific Laboratory, LA 5176 (June 1973).

The facility roof was designed to be watertight. The major roof components consist of precast concrete roof panels with concrete topping for a thickness of approximately 1 foot. The elevation of top of roof is approximately +39'-6"MLW. The lower assemblies were lowered into the facility by cranes with subsequent installation of the roof. A center wall in the facility allows for storage of Unit 4 assemblies on one side of the facility and storage of the Unit 3 assemblies on the opposite side of the facility.

Since the steam generators are welded in addition to being in a facility having a watertight concrete roof and reinforced concrete walls, there are no potential means to transport the surface contamination from the lower assembly surfaces. Therefore a floor, sumps and/or air filtration units are not required. However, as previously stated, a floor is provided.

An evaluation was performed to determine the most man-rem effective and cost-beneficial method for disposition of the removed lower assemblies. It was concluded that the lowest cost man-rem burdens are associated with (1) long-term, on-site storage and disposal during decommissioning and (2) immediate barge shipment to a licensed land burial facility.

Figure 4C-7 is a typical decay curve of percent steam generator gamma activity versus time following reactor shutdown. The initial activities used to generate the decay curve are given in Table 4C-3. With this decay curve, the effect of lower assembly radioactive decay can be directly related to the time of ultimate steam generator disposal and to the associated man-rem exposures.

TABLE 4C-1

STEAM GENERATOR DESIGN DATA (PER STEAM GENERATOR)

	Original (44)	RSG (44F) at Uprate (2)	RSG at EPU
Power Level (MWt)	736	769.3(5)	884(6)
Design Pressure, Reactor Coolant/Steam, psig	2485/1085	N.C.(1)(4)	N.C.(1)
Reactor Coolant Hydrostatic Test Pressure (tube side), psig	3107	N.C.(1)	N.C.(1)
Hydrostatic Test Pressure, Shell side, psig	1356	N.C.(1)	N.C.(1)
Design Temperature, Reactor Coolant/Steam, ^{o}F	650/556	N.C.(1)	N.C.(1)
Steam Conditions at 100% Load and 0% Tube Plugging (per SG), Outlet Nozzle:		N.C.(1)	N.C.(1)
Steam Flow, lb per hr.	3.39×10^{6}	3.39×10^{6}(4)	3.87×106(6)
Steam Temperature, ^{o}F	522.8	516.3	514.8(6)
Steam Pressure, psig	817	772(4)	762(6)
Feedwater Temperature at 100% Load ^{o}F	443	443(4)	440(6)
Overall Height, ft-in	63-1.6	N.C.(1)	N.C.(1)
Shell OD, upper/lower, in.	166/127	N.C.(1)	N.C.(1)
Shell Thickness, upper/lower	3.5/2.63	N.C.(1)	N.C.(1)
U-tube OD, in.	0.875	N.C.(1)	N.C.(1)
Tube Wall Thickness, (nominal) in.	0.050	N.C.(1)	N.C.(1)
Number of Manways/ID, in.	4/16	N.C.(1)	N.C.(1)
Number of Handholes/ID, in.	2/6	6/6	6/6
Number of U-tubes	3260	3214(5)	3214(6)
Tube length (largest U-bend), in.	397.5	N.C.(1)	N.C.(1)
Total Heat Transfer Surface Area, ft^2	44,430	43,467(5)	43,467(6)
Reactor Coolant Water Volume, $ft.^3$	925	N.C.(1)(5)	N.C.(1)(6)
Reactor Coolant Flow, lb/hr	32.1×10^{6}	32.4×10^{6}(4)	33.0×10^{6} (6)
Secondary Side Volume, $ft.^3$	4580	4682(5)	4698(6)
Secondary Side Mass No Load, lbs	134,000	N.C.(1)	N.C.(1)
Secondary Side Mass at 100% Power, lbs	76,300	81,500(5)	81,775 (6)
Center of Gravity (from support pads), ft./in.	25/4	N.C.(1)	N.C.(1)

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NOTE:

(1) N.C. means there was No Change to design data.

(2) Reflects Replacement Steam Generator (RGS) at thermal power uprate conditions.

(3) Reflects Replacement Steam Generator (RGS) at thermal power EPU uprate conditions.

(4) According to PCWG-2779, at lowest full power operating T_{avg} of 571.2^{o}F.

(5) According to WTD-TH-79-001, Rev.5.

(6) PCWG-08-34 Case at 400^{o}F feedwater and lowest full power operating T_{avg} of 577.0^{o}F

TABLE 4C-2

STEAM GENERATOR MATERIALS

	Original	Repaired
Plate (shell courses)	SA-302 Grade B	SA-533 Grade A Class 2
Tube Sheet Forging	SA-336	SA-508 Class 2a
Channel Head Casting	SA-216 Grade WCC	SA-216 Grade WCC
Support Plates	SA-285 Grade C	SA-240 Type 405
Channel Head Cladding	Stainless Steel, Type 304 or equiv.	Stainless Steel Type 304 or equiv.
Tube Sheet Cladding	Inconel	Inconel
Tubes	SB-163	SB-163 Thermally Treated

TABLE 4C-3

ESTIMATED CORROSION PRODUCT ACTIVITIES ON STEAM
GENERATOR PRIMARY SIDE PLENUM (1) (2) (3) (4) (5) (6)

Isotope	Activity (Ci/cm2)	Isotope	Activity (Ci/cm2)

Security-Related Information - Withheld Under 10 CFR 2.390

Notes

(1) The activities are based on actual Turkey Point data.

(2) Activities listed are extrapolated to 9 years of commercial operation.

(3) For Unit 4 (approximately 7 years of commercial operation) activities are bounded by those for Unit 3 (approximately 9 years of commercial operation).

(4) The activities shown are for 90 hours after shutdown.

(5) Multiplication Factors for Isotopic Concentration for Components in the steam generator:

Component	Relative Concentration Factor	Area (cm2)
Tubes	0.12	4.1 x 10^7
Divider Plate	2.0	7.2 x 10^4
Tube Sheet	2.0	3.8 x 10^4
Rolled Tube Ends	45/tube end	6520 tube ends
Channel Head Bowl	1	1.5 x 10^5

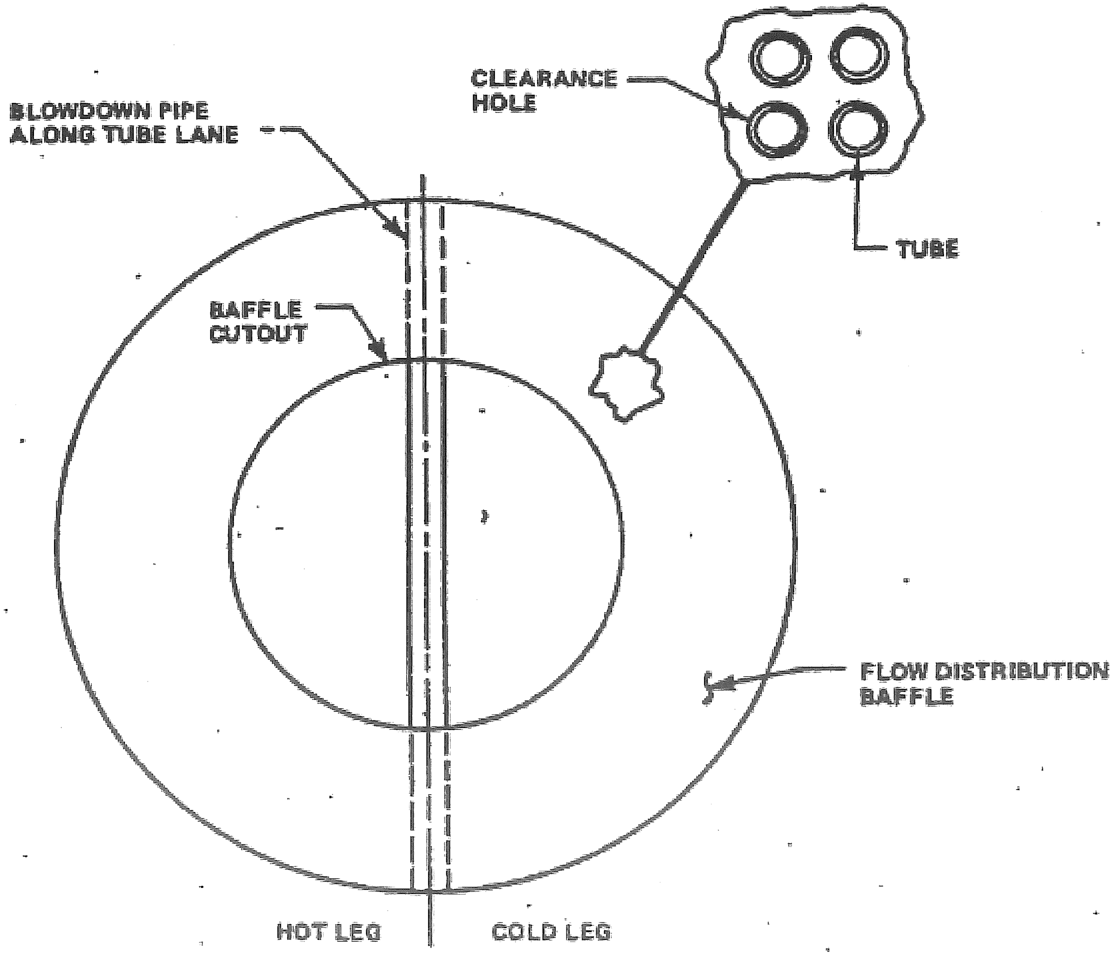
(6) The amount of each isotope, in curies, can be obtained by decaying the isotope for 80 days (the estimate for the earliest that the generators can be removed from the containment); by multiplying the surface area for each component by the primary side concentration and the relative concentration factor; and by summing for all components.

Security-Related Information - Withheld Under 10 CFR 2.390

TURKEY POINT PLANT UNITS 3 & 4
STEAM GENERATOR REPAIR REPORT

STEAM GENERATOR LOWER ASSEMBLY

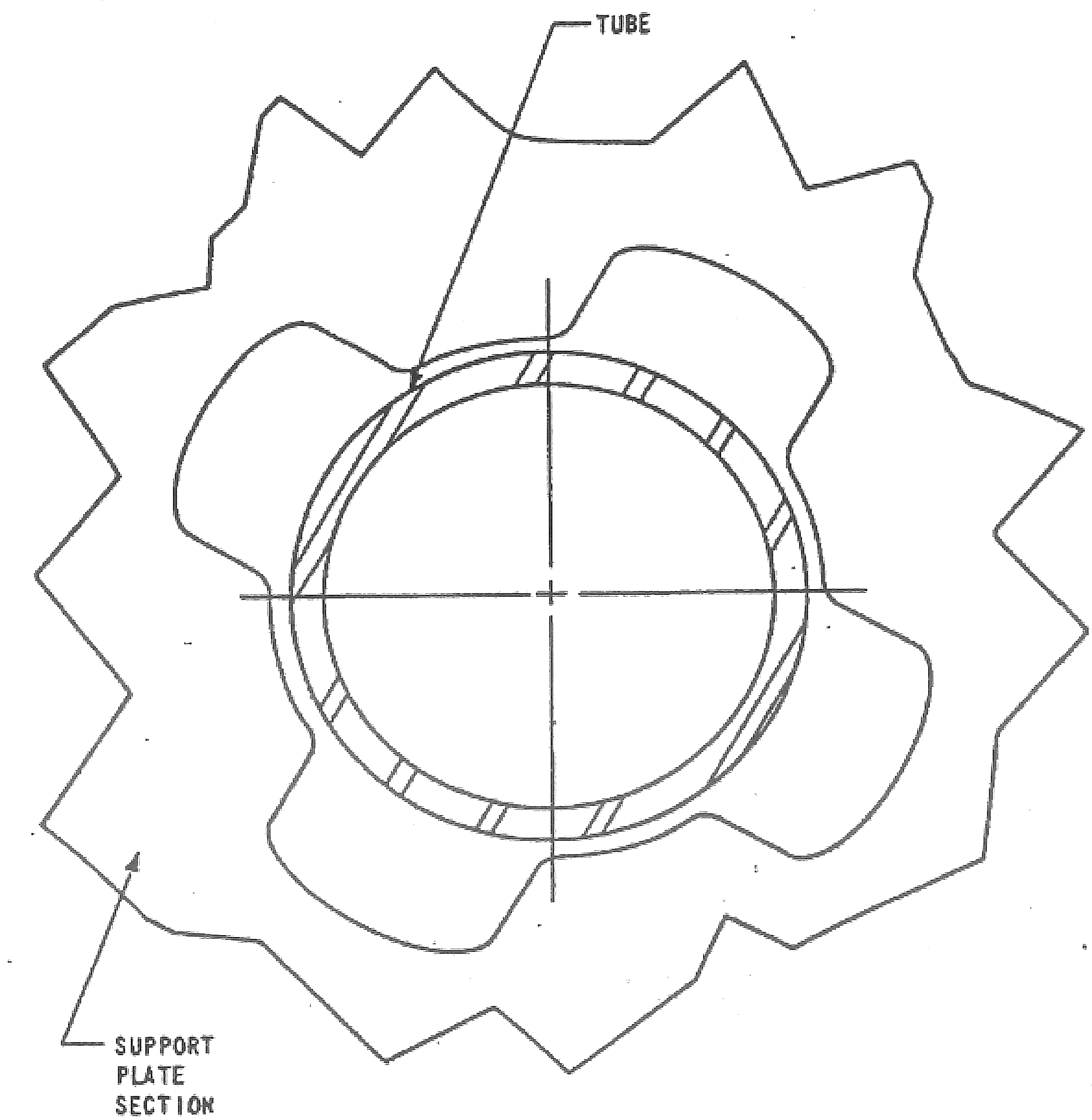
FIGURE 4.C-1



TURKEY POINT PLANT UNITS 3 & 4
STEAM GENERATOR REPAIR REPORT

FLOW DISTRIBUTION BAFFLE AND BLOWDOWN

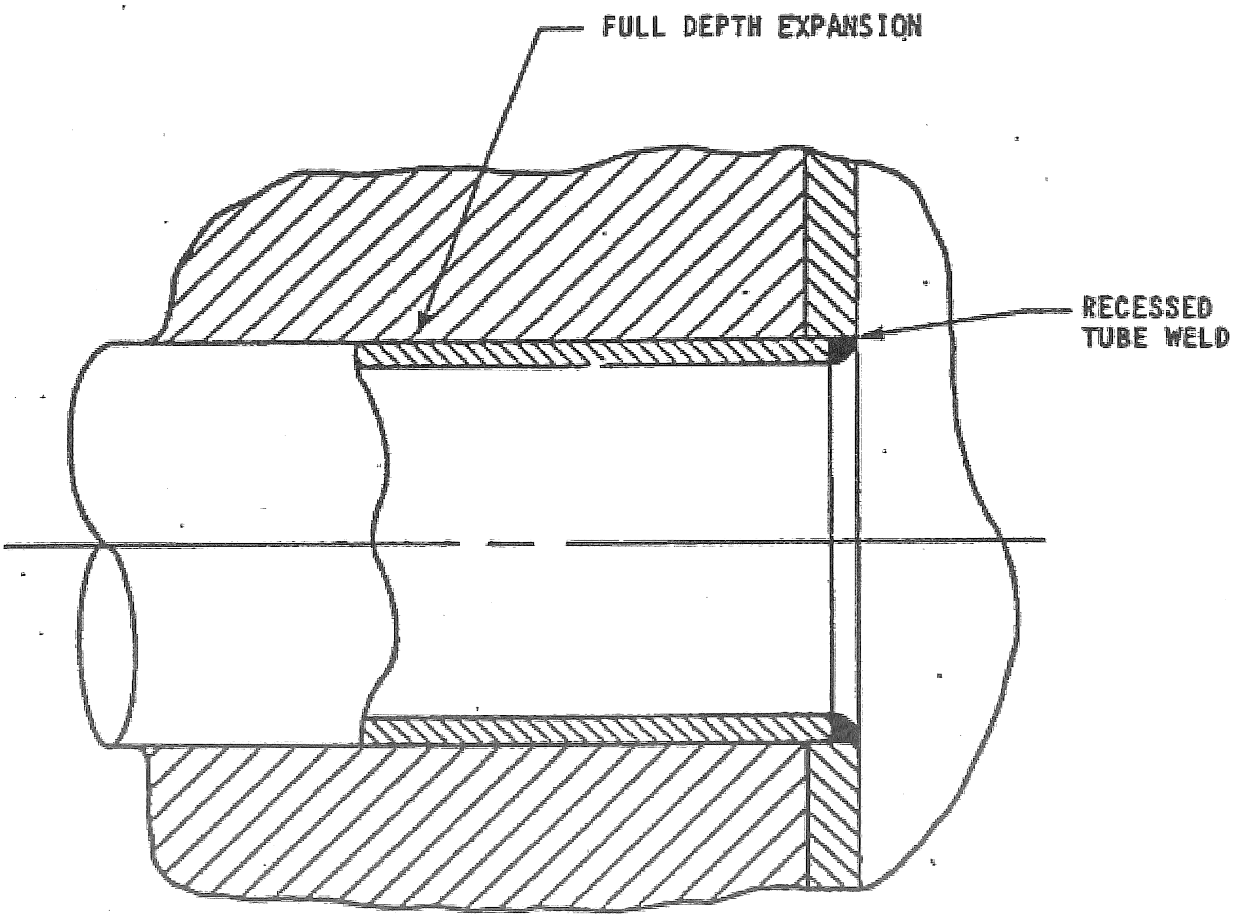
FIGURE 4.C-2



TURKEY POINT PLANT UNITS 3 & 4
STEAM GENERATOR REPAIR REPORT

QUATREFOIL TUBE SUPPORT PLATES

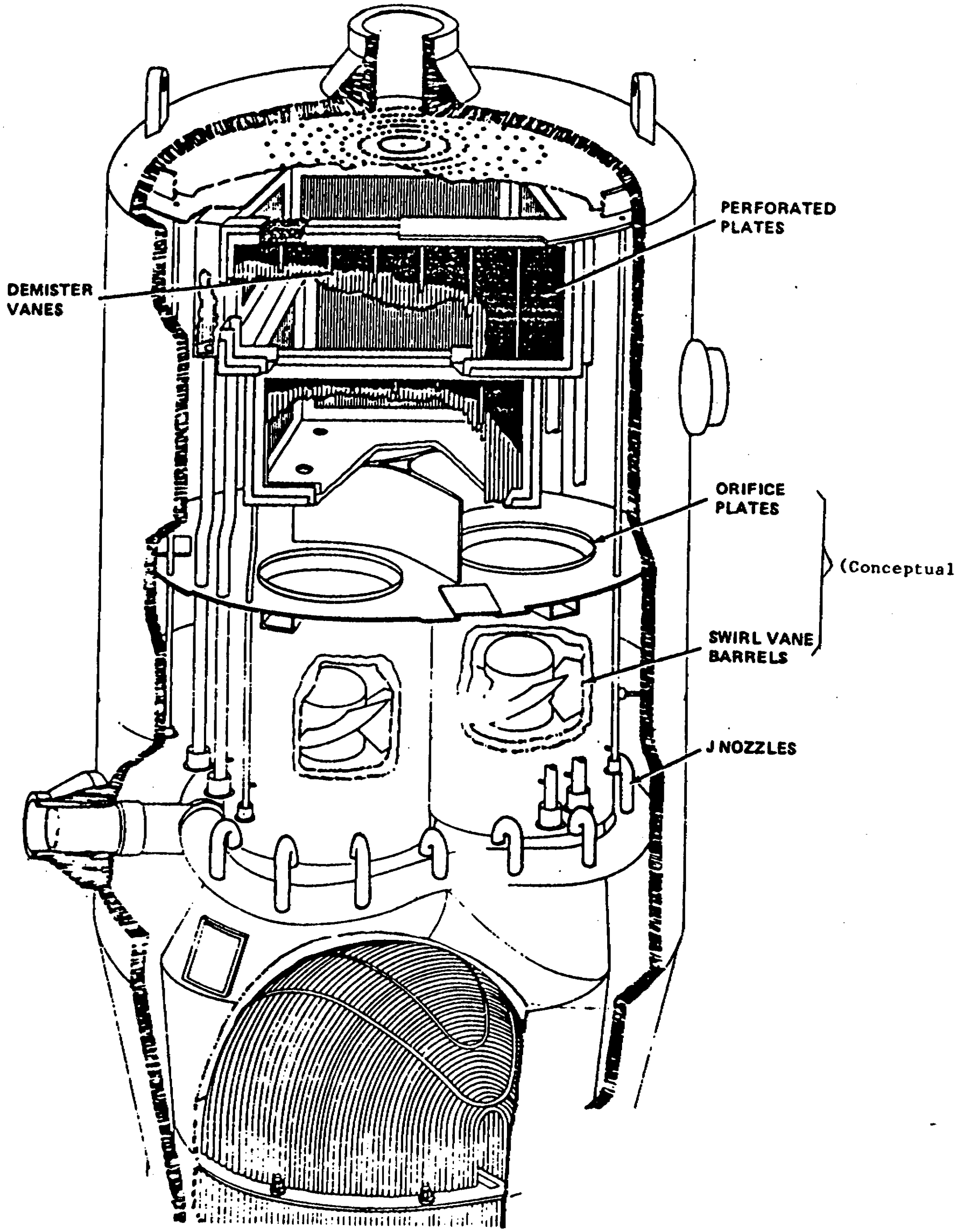
FIGURE 4.C-3



TURKEY POINT PLANT UNITS 3 & 4
STEAM GENERATOR REPAIR REPORT

TUBE TO TUBESHEET JUNCTUNE

FIGURE 4.C-4



Rev. 1-11/83

TURKEY POINT PLANT UNITS 3 & 4
STEAM GENERATOR REPAIR REPORT

MOISTURE SEPARTOR
IMPROVEMENTS

FIGURE 4.C-5

Security-Related Information - Withheld Under 10 CFR 2.390

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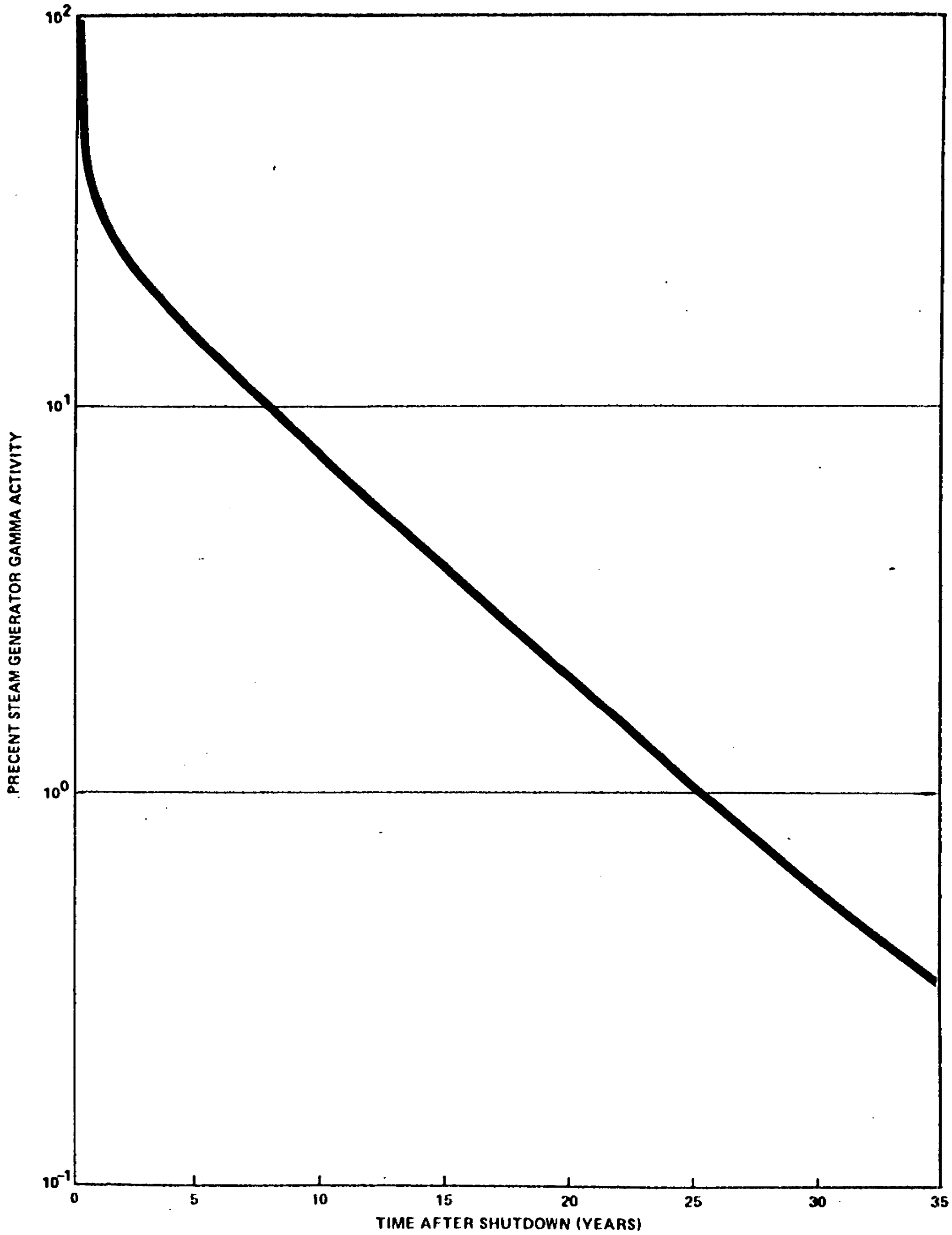
TURKEY POINT PLANT UNITS 3 & 4
STEAM GENERATOR REPAIR REPORT

STEAM GENERATOR STORAGE COMPOUND

Revision 7
March 199

FIGURE 4.C-6



Rev. 1-11/83

TURKEY POINT PLANT UNITS 3 & 4 STEAM GENERATOR REPAIR REPORT
LOWER ASSEMBLY GAMMA RADIOACTIVE DECAY VS TIME Revision April 1978
FIGURE 4.C-7