

REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

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Section 5.3.1.6.3.2	LBDCR 17-002
Section 5.3.2.1	LBDCR 17-002/17-020



Section

Cross References

Revision 310 (con't)

Section 5.3.2.2  
Section 5.3.2.3  
Section 5.3 References  
Table 5.3-3  
Table 5.3-13  
Table 5.3-14  
Section 5.4.2.1

LBDCR 17-002  
LBDCR 17-020  
LBDCR 17-002  
LBDCR 17-020  
LBDCR 17-002  
LBDCR 17-020  
LBDCR 16-037

## 5.0 REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

### 5.1 SUMMARY DESCRIPTION

The reactor is a pressurized water reactor (PWR) with two coolant loops. The Reactor Coolant System (RCS) circulates coolant (borated water) in a closed cycle, removing heat from the reactor core and internals and transferring it to a secondary (steam generating) system. In a pressurized water reactor, the steam generators provide the interface between the Reactor Coolant (primary) System and the Main Steam (secondary) System. The steam generators are vertical U-tube heat exchangers in which heat is transferred from the reactor coolant to the Main Steam System. Reactor coolant is prevented from mixing with the secondary steam by the steam generator tubes and the steam generator tube sheet. This makes the RCS a closed system which forms a barrier to the release of radioactive materials from the core of the reactor to the containment.

The arrangement of the RCS is shown in Figures 5.1-1 and 2. The major components of the system are the reactor vessel; two parallel heat transfer loops, each containing one steam generator and two reactor coolant pumps; a pressurizer connected to one of the reactor vessel outlet pipes; and associated piping. All components are located inside the containment. Effluent discharges from the pressurizer safety valves are condensed and cooled in the quench tank.

→(EC-8458, R307)

The major impacts to the RCS due to steam generator replacement were evaluated. An increase in RCS volume of approximately 4%, an increase in heat transfer capability, and a slight change in normal RCS flow rate are expected. The potential impact to specific RCS functional requirements and design bases were addressed in detail, and based on those evaluations the evaluated RCS functional and performance requirements are met following steam generator replacement.

←(EC-8458, R307)

Table 5.1-1 shows the principal pressures, temperatures and flowrates of the RCS under normal steady-state, full-power operating conditions. Instrumentation provided for operation and control of the system is described in Chapter 7.

System pressure is controlled by the pressurizer, where steam and water are maintained in thermal equilibrium. Steam is formed by energizing immersion heaters in the pressurizer, or is condensed by the pressurizer spray to limit pressure variations caused by contraction or expansion of the reactor coolant.

The average temperature of the reactor coolant varies with power level and the fluid expands or contracts, changing the pressurizer water level.

The charging pumps and letdown control valves in the Chemical and Volume Control System (CVCS) are used to maintain the programmed pressurizer water level. A continuous but variable letdown purification flow is maintained to keep the RCS chemistry within prescribed limits. Two charging nozzles and a letdown nozzle are provided on the reactor coolant piping for this operation. The charging flow is also used to alter the boron concentration or correct the chemistry of the reactor coolant.

Other reactor coolant loop penetrations are the pressurizer surge line in one reactor vessel outlet pipe; the four safety injection inlet nozzles, one in each reactor vessel inlet pipe; one outlet nozzle to the Shutdown Cooling System in each reactor vessel outlet pipe; two pressurizer spray nozzles; vent and drain connections; and sample connections and instrument connections.

Overpressure protection for the reactor coolant pressure boundary is provided by two spring-loaded ASME Code safety valves connected to the top of the pressurizer. These valves discharge to the quench tank, where the steam is released under water to be condensed and cooled. If the steam discharge exceeds the capacity of the quench tank, it is relieved to the containment atmosphere through a rupture disc.

Overpressure protection for the secondary side of the steam generators is provided by 12 spring-loaded ASME Code safety valves located in the Main Steam System upstream of the main steam isolation valves.

Components and piping in the RCS are insulated with a material compatible with the temperatures involved to reduce heat losses and protect personnel from high temperatures. All insulation material used has a low soluble chloride and other halide content to minimize the possibility of stress corrosion of stainless steel.

Principal parameters of the RCS are listed in Table 5.1-2. Table 5.1-3 lists RCS volumes.

Shielding requirements of the surrounding concrete structures are described in Section 12.3. Reactor Coolant System shielding permits limited personnel access to the containment building during power operation. The reactor vessel sits in a thick walled concrete well. This and other shielding reduces the dose rate within the containment and outside the shield wall during full power operation to acceptable levels.

#### 5.1.1 SCHEMATIC FLOW DIAGRAM

The principal pressures, temperatures, and flow rates at major components are listed in Table 5.1-1. These parameters are referenced to Figure 5.1-3, the piping and instrument diagram, by numbered locations. Instrumentation provided for operation and control of the RCS is described in Chapter 7 and is indicated on Figure 5.1-3.

#### 5.1.2 PIPING AND INSTRUMENT DIAGRAM

Figure 5.1-3 is the piping and instrument diagram of the RCS. The entire system is located within the containment. Fluid systems which are connected to the Reactor Coolant System and which include the limits of the reactor coolant pressure boundary as defined in 10CFR50.2 (v), are identified and the appropriate piping and instrument diagrams in other sections are referenced. Figure 5.1-4 is the piping and instrument diagram for the reactor coolant pump seals.

#### 5.1.3 ELEVATION DRAWING

Major components of the RCS are surrounded by concrete structures, which provide support plus shielding and missile protection. Elevation drawings, illustrating principal dimensions of the RCS in relationship to the surrounding concrete structures, are provided on Figures 1.2-17 through 1.2-22.

WSES-FSAR-UNIT-3

TABLE 5.1-1

Revision 307 (07/13)

PROCESS DATA POINT TABULATION

➔(DRN 03-2059, R14)

<u>Parameter</u>	<u>Pressurizer</u>	<u>S.G. 1-A Midpoint</u>	<u>Pump 1-B Outlet</u>	<u>Reactor Vessel Midpoint</u>	<u>Pump 1-A Outlet</u>	<u>S.G. 2-A Midpoint</u>	<u>Pump 2-A Outlet</u>	<u>Pump 2-B Outlet</u>
Data Point Figure 5.1-3	1	2	3	4	5	6	7	8
Pressure, psig	2,250	2,232.0	2,306.4	2,282.1	2,306.4	2,232.0	2,306.4	2,306.4
➔(EC-8458, R307) Temperature, °F	652.7	572.5	543	572.5	543	572.5	543	543
Mass Flowrate lbm/hr (x 10 <sup>6</sup> )	-	81.8	40.9	163.5	40.9	81.8	40.9	40.9
Volumetric Flow- rate, gpm (x 10 <sup>4</sup> )	-	21.5	10.8	43.1	10.8	21.5	10.8	10.8

←(DRN 03-2059, R14; EC-8458, R307)

→(DRN 03-2059, R14)

PARAMETERS OF REACTOR COOLANT SYSTEM

Parameter	Value
Rated Thermal Power, Mwt (core only)	3,716
Thermal Power Btu/hr (including heat addition from RCPs)	1.274 x 10 <sup>10</sup>
Design Pressure, psia	2,500
Design Temperature (except pressurizer and surge line), °F	650
Pressurizer and Surge Line Design Temperature, °F	700
→(EC-8458, R307) Coolant Flow Rate, Operating, lbm/hr ←(EC-8458, R307)	163.5 x 10 <sup>6</sup>
Cold Leg Temperature, Operating, °F	543
→(EC-8458, R307) Average Temperature, Operating, °F	572.5
Hot Leg Temperature, Operating, °F ←(DRN 03-2059, R14; EC-8458, R307)	602
Normal Operating Pressure, psia	2,250
→(EC-8458, R307) System Water Volume, ft. <sup>3</sup> (without pressurizer) ←(EC-8458, R307)	10,485
Pressurizer Water Volume, ft. <sup>3</sup> (full power)	800
Pressurizer Steam Volume, ft. <sup>3</sup> (full power)	700

## WSES-FSAR-UNIT-3

TABLE 5.1-3

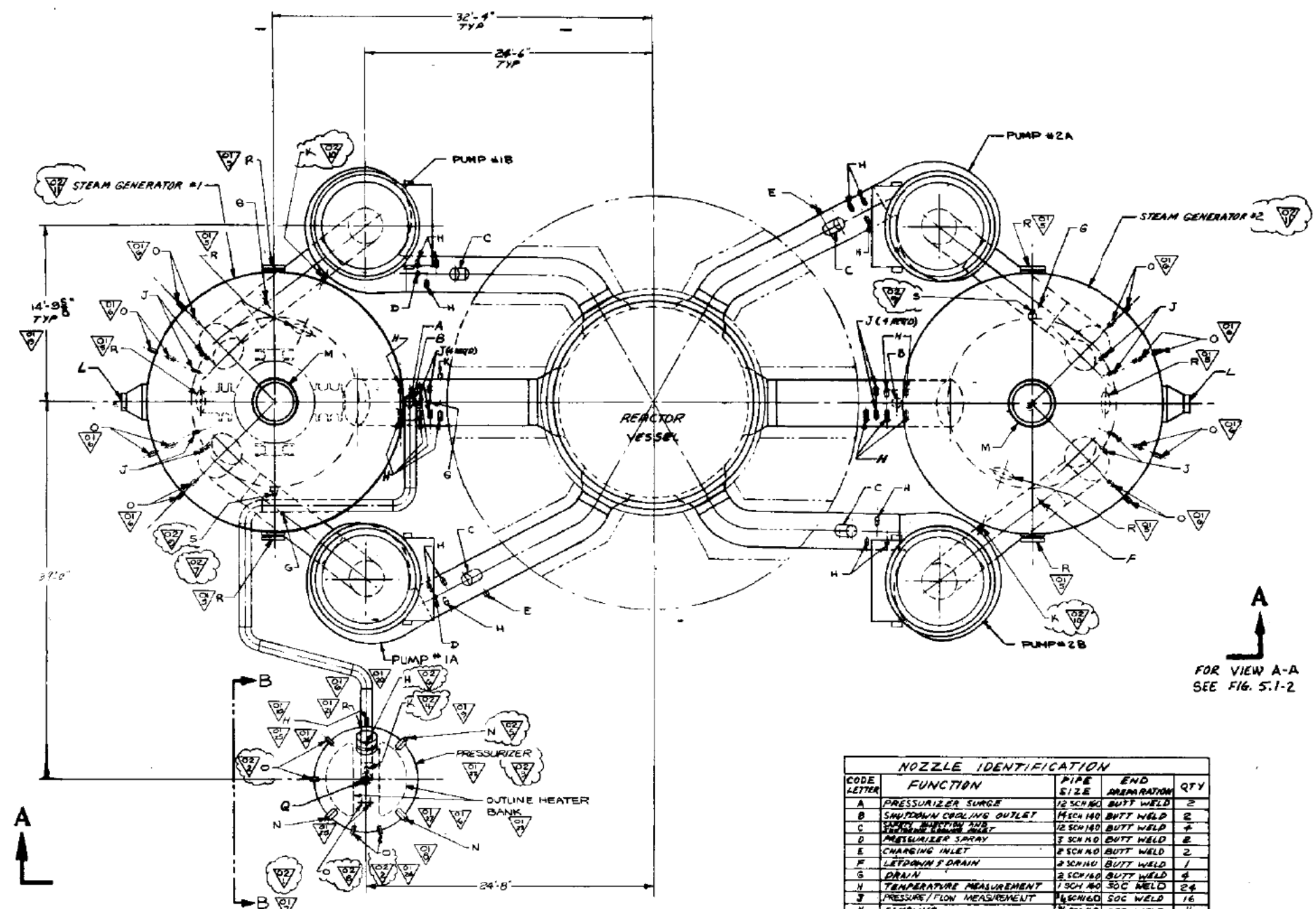
Revision 307 (07/13)

NOMINAL REACTOR COOLANT SYSTEM VOLUMES

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Component	Volume (ft. <sup>3</sup> )
→(EC-8458, R307) Reactor Vessel (including nozzles)	4,811
←(EC-8458, R307) Steam Generators (including nozzles)	2,051 each
Reactor Coolant Pumps	112 each
→(EC-8458, R307) Pressurizer	1,519
Piping	
Hot Leg ←(EC-8458, R307)	145 each
Cold Leg	198 each
→(EC-8458, R307) Surge Line ←(EC-8458, R307)	39

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A  
↑  
FOR VIEW A-A  
SEE FIG. 5.1-2

FOR VIEW B-B SEE  
FIG. 5.1-2

NOZZLE IDENTIFICATION				
CODE LETTER	FUNCTION	PIPE SIZE	END ARRANGEMENT	QTY
A	PRESSURIZER SURGE	1/2 SCH 40	BUTT WELD	2
B	SHUTDOWN COOLING OUTLET	1/2 SCH 40	BUTT WELD	2
C	SAFETY SHUTDOWN INLET	1/2 SCH 40	BUTT WELD	4
D	PRESSURIZER SPRAY	1/2 SCH 40	BUTT WELD	2
E	CHARGING INLET	2 SCH 40	BUTT WELD	2
F	LETDOWN DRAIN	2 SCH 40	BUTT WELD	1
G	DRAIN	2 SCH 40	BUTT WELD	4
H	TEMPERATURE MEASUREMENT	1/2 SCH 40	SOC WELD	24
J	PRESSURE/FLOW MEASUREMENT	1/2 SCH 40	SOC WELD	16
K	SAMPLING	1/2 SCH 40	SOC WELD	4
L	FEEDWATER	1/2 SCH 40	BUTT WELD	2
M	STEAM OUTLET	30" I.D.	BUTT WELD	2
N	SAFETY VALVE INLET	6 SCH 40	BUTT WELD	3
D	PRESSURE/LEVEL MEASUREMENT	1/2 SCH 40	SOC WELD	22
G	PRESSURIZER SPRAY	1/2 SCH 40	BUTT WELD	1
R	MANWAY	16" I.D.		3
S	BLOWDOWN	2 SCH 80	BUTT WELD	2

LOUISIANA POWER & LIGHT CO.  
Waterford Steam Electric Station

REACTOR COOLANT  
SYSTEM ARRANGEMENT PLAN

FIGURE 5.1-1

WSES-FSAR-UNIT-3

→(DRN 07-2, R15)

Figure 5.1-2 has been incorporated by reference in accordance with NEI 98-03.

Figure information can be found on Drawing 1564-22.

←(DRN 07-2, R15)



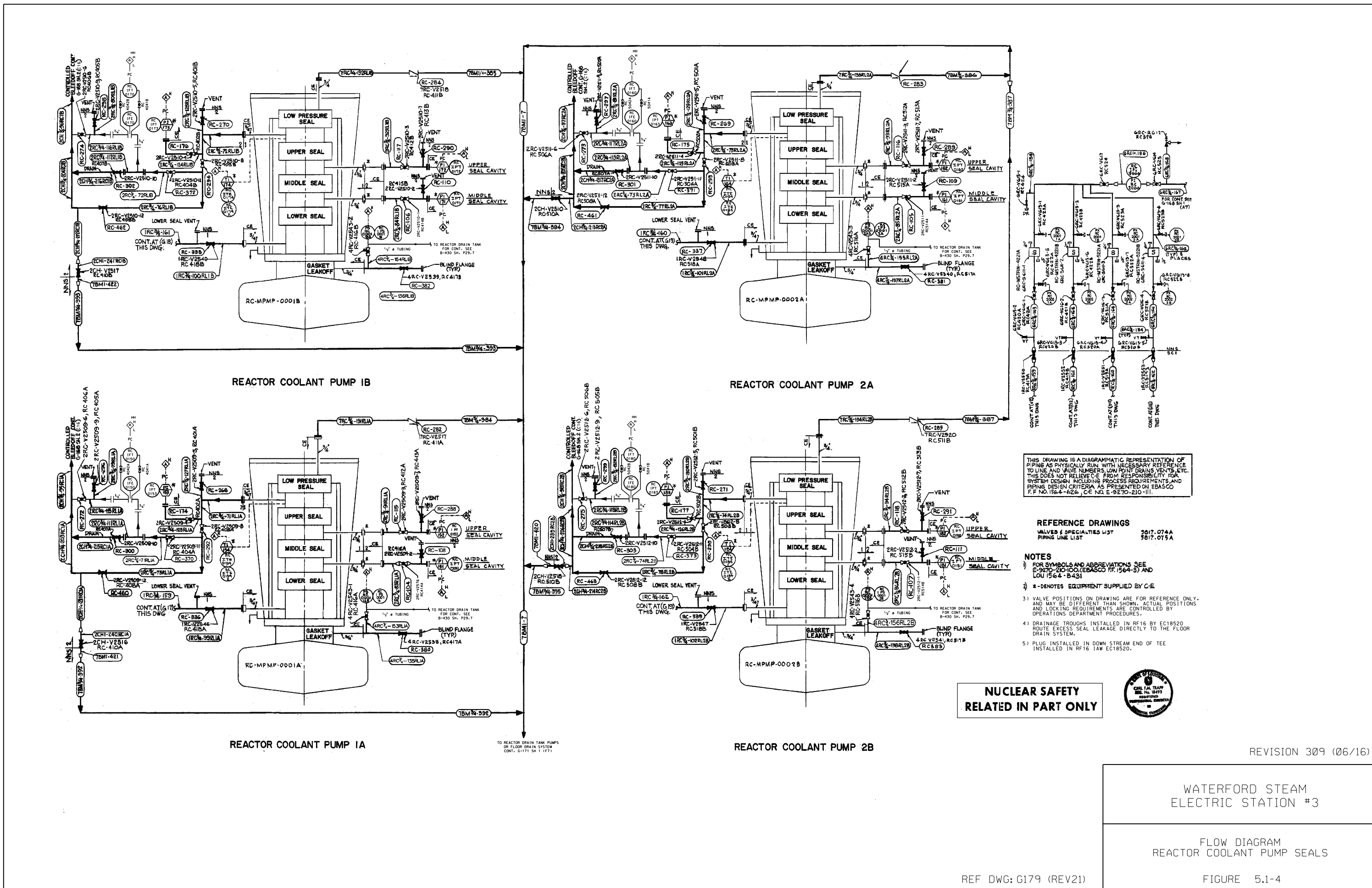
WSES-FSAR-UNIT-3

→(DRN 03-1348, R13)

Figure 5.1-3 has been incorporated by reference in accordance with NEI 98-03

Figure information can be found on Drawing G-172

←(DRN 03-1348, R13)



REACTOR COOLANT PUMP 1B

REACTOR COOLANT PUMP 2A

REACTOR COOLANT PUMP 1A

REACTOR COOLANT PUMP 2B

THIS DRAWING IS A DIAGRAMMATIC REPRESENTATION OF PIPING AS PHYSICALLY RUN WITH NECESSARY REFERENCE TO LINE AND VALVE NUMBERS, LOW POINT DRAINS, VENTS, ETC. THIS DOES NOT RELIEVE C-E FROM RESPONSIBILITY FOR SYSTEM DESIGN INCLUDING PROCESS REQUIREMENTS, AND PIPING DESIGN CRITERIA AS PRESENTED ON EBASCO F.F. NO. 1564-626, C.E. NO. E-9270-210-111.

**REFERENCE DRAWINGS**  
 VALVES & SPECIALTIES LIST 2017.074A  
 PIPING LINE LIST 2017.075A

- NOTES**
- FOR SYMBOLS AND ABBREVIATIONS SEE E-9270-210-100 (EBASCO F.F. 1564-3) AND LOU 1564-B-431
  - \* - DENOTES EQUIPMENT SUPPLIED BY C-E
  - VALVE POSITIONS ON DRAWING ARE FOR REFERENCE ONLY, AND MAY BE DIFFERENT THAN SHOWN. ACTUAL POSITIONS AND LOCKING REQUIREMENTS ARE CONTROLLED BY OPERATIONS DEPARTMENT PROCEDURES.
  - DRAINAGE TROUGHS INSTALLED IN RF16 BY EC18520 ROUTE EXCESS SEAL LEAKAGE DIRECTLY TO THE FLOOR DRAIN SYSTEM.
  - PLUG INSTALLED IN DOWN STREAM END OF TEE INSTALLED IN RF16 IAW EC18520.

**NUCLEAR SAFETY  
 RELATED IN PART ONLY**



REVISION 309 (06/16)

WATERFORD STEAM  
 ELECTRIC STATION #3

FLOW DIAGRAM  
 REACTOR COOLANT PUMP SEALS

REF DWG: G179 (REV21)

FIGURE 5.1-4

5.2 INTEGRITY OF REACTOR COOLANT PRESSURE BOUNDARY

This section discusses the measures employed to provide and maintain the integrity of the Reactor Coolant Pressure Boundary (RCPB) throughout the facility design lifetime. The RCPB is defined in accordance with 10CFR50.2(v), to include all pressure containing components such as Pressure vessels, piping, pumps, and valves which are:

- a) Part of the Reactor Coolant System (RCS), or
- b) Connected to the RCS, up to and including any and all of the following:
  - 1) The outermost containment isolation valve in system piping that penetrates the primary containment
  - 2) The second of two valves normally closed during normal reactor operation in system piping which does not penetrate the primary containment.
  - 3) The RCS primary safety valves

5.2.1 COMPLIANCE WITH CODES AND CODE CASES

5.2.1.1 Compliance With 10CFR50.55a

10CFR50.55a provides minimum dates for codes and standards applicable to the RCPB\*. This regulation became effective on July 12, 1971.

The construction permit application for Waterford 3 was filed in December, 1970. Major components for the Nuclear Steam Supply System were ordered consistent with the application date and anticipated schedule. The reactor vessel, steam generators, pressurizer and piping were ordered in March, 1971 and the reactor coolant pumps, in March, 1972. The construction permit was received in November, 1974.

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\*Components that are connected to the RCS and are part of the RCPB may be excluded from complying with 10CFR50.55a, provided:

- a) For postulated failure of the component during normal reactor operation, the reactor can be shutdown and cooled in an orderly manner assuming makeup is provided by the Chemical and Volume Control System (CVCS).
- b) The component is or can be isolated from the RCS by two valves (both closed, both open, or one closed and the other open). Each open valve must be capable of automatic actuation and its closure time must be such that for postulated failure of the component during normal reactor operation the reactor can be shutdown and cooled in an orderly manner assuming makeup is provided by the CVCS only.

The codes editions and addenda used are later than those listed in the PSAR and are listed in Table 5.2-1.

Waterford 3 meets or exceeds the in-service inspection requirements of 50.55a (g) (3) with respect to the design and access provisions as well as with respect to preservice examination requirements. The codes and addenda used for preservice examination requirements are also given in Table 5.2-1. The ability to conduct subsequent in-service inspections throughout the life of the unit, to the extent practical within the limitations of design, geometry and materials of construction of the components, is addressed in Subsection 5.2.4.

→(DRN 00-1059, R11-A)

Section 50.55a(h) requires that protection systems meet the requirements of IEEE-279 in effect on the docket date of the application for a construction permit. Protection systems at Waterford 3 meet or exceed the requirements of 50.55a(h).

←(DRN 00-1059, R11-A)

→(EC-8458, R307, LBDCR 17-002, R310)

Fracture toughness standards for materials employed in pressure-retaining components of the reactor coolant pressure boundary existed prior to the order dates of all components used at Waterford. The reactor vessel, pressurizer, primary coolant piping and pumps all meet the fracture toughness requirements in effect through the Winter 1971 Addenda to the ASME Code. Additionally, the reactor vessel lower, middle, and upper shell plates meet the fracture toughness testing requirements in effect in the 1971 edition with Summer 1973 addenda to the ASME code. The replacement steam generators meet the fracture toughness requirements in effect in the 1998 Edition with 2000 Addenda of the ASME Code. Thus, longitudinal direction Charpy impact tests were performed satisfactorily for all of these components. Charpy V-notch results are described in Subsection 5.2.3.

←(EC-8458, R307, LBDCR 17-002, R310)

→( LBDCR 17-002, R310)

The Summer 1972 Addenda and 10CFR50 Appendix G expanded testing requirements to include dropweight testing and transverse Charpy impact testing of plates, as well as dropweight testing of weld material test weld and welding procedure qualification welds and heat-affected zones (HAZs). Materials to perform such additional testing were not available, with the exception of the vessel upper, middle, and lower shell plates. Combustion Engineering ordered supplemental tests for the reactor vessel plate materials to fully meet the requirements of Appendix G and Section III NB-2331 of the Summer 1973 Addenda to the Code.

Despite incomplete dropweight data, the reactor vessel beltline and extended beltline weld materials have been evaluated using the  $RT_{NDT}$  guidelines introduced in the ASME Code 1972 Summer Addenda. To do this, C-E has made use of the generic Linde 0091 flux  $RT_{NDT}$  from 10 CFR 50.61 for submerged arc welds, sister weld  $RT_{NDT}$  for SMAW welds using filler metal heat HODA, and a bounding conservative SMAW weld  $RT_{NDT}$  for other welds without dropweight data.

←(LBDCR 17-002, R310)

→(LBDCR 17-002, R310)

To present the fracture toughness data as required by Appendix G to the maximum extent practical, the remainder of the available test data for the reactor vessel, steam generators, pressurizer and pumps have been evaluated according to Branch Technical Position MTEB 5-2 "Fracture Toughness Requirements". This approach, which was recommended by the NRC Staff at a December, 1974 meeting, results in a downgrading of the material fracture toughness properties and provides more conservatism than if the testing actually had been performed in accordance with 10CFR50, Appendix G. The available fracture toughness data are reported in Subsection 5.2.3.

←(LBDCR 17-002, R310)

→(DRN 00-1059, R11-A, LBDCR 17-002, R310)

The methods of MTEB 5-2, which allow the development of an  $RT_{NDT}$  for materials exhibiting a fracture toughness of at least 30 ft-lbs absorbed energy, were applied, with the exception of the beltline and extended beltline plate and weld materials.

←DRN 00-1059, R11-A LBDCR 17-002, R310)

→DRN 02-218, R11-A, LBDCR 17-002, R310)

Conservatism in the evaluations of Waterford 3 primary system pressure boundary ferritic materials has been confirmed by the supplemental testing performed in accordance with Appendix G of Part 50. In

addition to the Charpy impact testing conducted with test specimens prepared from longitudinal (strong direction) material, Charpy impact testing on transverse (weak direction) material and drop weight tests on base metal, welds and HAZ materials for the most limiting reactor vessel beltline material have been performed. Materials for the most limiting materials in the area beltline region were set aside and retained for purposes of performing baseline testing as part of Waterford's reactor vessel material surveillance program. LP&L elected not to wait to perform baseline testing of the limiting plate in the beltline region as is customarily done. The results of this testing, when contrasted with the results of the MTEB 5-2 evaluations, demonstrate the wide margin of conservatism in our evaluation technique.

←(DRN 02-218, R11-A, LBDCR 17-002, R310)

→(LBDCR 17-002, R310)

In summary, the components subject to 50.55a meet or exceed the design and construction requirements of that section as further discussed in the letters LP&L 8254, dated February 24, 1978 and LP&L 9992, dated November 10, 1978 and in all respects other than certain documentation and analyses requirements for valves which were promulgated subsequent to procurement of Waterford 3 components. Fracture toughness requirements of Appendix G have been satisfied either by meeting the fracture toughness testing requirements of the ASME Code Section III NB-2331 with Summer 1973 Addenda or by alternate methods of evaluation (use MTEB 5-2 and early testing of baseline surveillance materials).

←(LBDCR 17-002, R310)

Based on the above evaluations, testing and analyses, Waterford components comply with Section 50.55(a)(2)(ii) and valves, with Section 50.55(a)(2)(i).

→(EC-1020, R307)

All of the pressure retaining materials used in the fabrication of the Replacement Reactor Vessel Closure Head (RRVCH) have been tested to demonstrate compliance with the fracture toughness requirements of 10 CFR 50 Appendix G as required by the Code. All aspects of the fracture toughness (impact testing) were performed in compliance with subarticle NB-2200 and subarticle NB-2300 of the ASME Code Section III, Division 1, 1998 Edition through 2000 Addenda.

←(EC-1020, R307)

#### 5.2.1.2 Applicable Code Cases

The code cases applied to components within the RCPB are listed in Table 5.2-2.

##### 5.2.1.2.1 Regulatory Guide 1.84

Code cases applied to Waterford 3 are on the approved list except for the differences noted below:

###### a) Code Cases 1604

This code case was applied prior to the effective date of Regulatory Guide 1.84 and has been incorporated into the ASME Section III Code, Subsection NB, Paragraph NB 6223, 1974 Edition, Winter 1974 Addenda.

###### b) Code Case 1361-1

This code case is acceptable because the affected component was ordered to this specific revision prior to the specific approved version in the guide per Paragraph D.2.

##### 5.2.1.2.2 Regulatory Guide 1.85

Code Cases applied to Waterford 3 are on the approved list except as noted below:

###### a) Code Cases 1332-4, 1332-5, 1334-6, 1344-2, and 1557

These cases are acceptable because the affected components were ordered to these specific revisions prior to the specific approved version in the guide per Paragraph D.2.

###### b) Code Case 1401-1

This code case was previously approved by the guide and has since been annulled and is acceptable per Paragraph D-3 of the guide.

c) Code Cases 1459 and 1459-1

These code cases were applied prior to the effective data of the guide and have since been incorporated into ASME Code Section III.

5.2.2 OVERPRESSURIZATION PROTECTION

5.2.2.1 Design Bases

→(DRN 03-2059, R14)

The primary safety valves on the pressurizer and the secondary safety valves on the main steam lines are designed to protect the systems from overpressure, as required by ASME Code Section III. This is documented in the ASME code report on Overpressure Protection. See Appendix 5.2A.

←(DRN 03-2059, R14)

5.2.2.2 Design Evaluation

An evaluation of the functional design of the overpressurization protection system is given in Section 15.2. In this analysis, the ability of the overpressure protection system to maintain secondary and primary operating pressures within 110 percent of design is clearly demonstrated. The analytical model used in the analysis has been documented in Section 15.2.

→(DRN 03-2059, R14)

The assumptions used in the loss of load analysis are listed in Subsection 15.2.1.3 (Loss of Condenser Vacuum). These assumptions are chosen to maximize the required relieving capacity of the primary and secondary safety valves. The analysis demonstrates that sufficient relieving capacity has been provided so that, when acting in conjunction with the reactor Protective System, the safety valves will prevent the NSSS from exceeding 110 percent of the design pressure.

The pressurizer level remains below the primary safety valve inlet as demonstrated in the Feedwater Line Break analysis in Subsection 15.2.3, which produces the greatest increase in pressurizer level.

←(DRN 03-2059, R14)

5.2.2.3 Piping and Instrumentation Diagrams

The piping and instrumentation diagram showing the primary safety valves and the associated blowdown lines are given on Figure 5.1-3. The secondary safety valves are shown on Figure 10.2-4.

5.2.2.4 Equipment and Component Description

The primary safety valves are discussed in Subsection 5.4-13. A schematic drawing of the primary safety valve is given on Figure 5.4-11. The safety valve parameters are given in Table 5.4-9. The primary safety valves are designed to withstand the following transients:

→(DRN 02-524, R12)

- a) 650°F to 375°F in 50 seconds and return to 650°F in 2000 seconds for five cycles (loss of secondary pressure).

←(DRN 02-524, R12)

→(DRN 00-1059, R11-A)

- b) Temperature changes of 100°F to 400°F and a return to 100°F at a rate of 100°F/hr; and simultaneous pressure changes from 400 psig to 2250 psig and returning to 400 psig in step changes. 200 cycles of this combined transient are allowed (plant leak test).

←(DRN 00-1059, R11-A)

- c)  $\pm 10^\circ\text{F}$  step change from 653°F, 1,030,000 cycles. (Plant loading, unloading,  $\pm 10$  percent step load, normal plant variation.)

→(DRN 06-1002, R15)

d) 75°F to 653°F and return to 75°F at a rate of 200°F/hr with pressures at saturated levels for 500\* cycles. (Plant heatup and cooldown.) Heatup and cooldown are separate transients, each beginning at steady state conditions.

→(DRN 06-1002, R15)

e) Pressurize to 1.5 times set pressures at 100°F-200°F for 10 cycles plus number of hydros conducted prior to valve shipment (Hydrostatic test).

A description of overpressurization equipment and components for the Main Steam System is included in Section 10.3.

#### 5.2.2.5 Mounting of Pressure-Relief Devices

Figure 5.2-1 provides design and installation details for the pressure relief devices mounted in the secondary side of the steam generator.

Design and installation details for the primary safety valves are provided in Subsection 3.9.3.3.

#### 5.2.2.6 Applicable Codes and Classifications

The applicable codes and classification for the overpressurization protection system are contained in Table 3.2-1. Also see Subsections 5.4.11, 5.4.13, 10.3.1 and 10.3.6.

#### 5.2.2.7 Material Specification

Material specifications for the overpressure protection system are given in Subsections 5.4.13 and 10.3.6.

#### 5.2.2.8 Process Instrumentation

Figures 5.1-3 and 10.2-4 show process instrumentation for the overpressurization protection system.

#### 5.2.2.9 System Reliability

Reliability of the main steam (secondary) safety valves is discussed in Section 10.3. The primary safety valves are passive spring-actuated mechanisms which do not fail-close if setpoint pressure is exceeded. The operational reliability of the primary safety valves is assured by:

- Compliance with ASME Code Sections III and XI for safety valves
- Conservative design criteria
- Selection of a vendor with proven experience and expertise
- Accounting for thermal cycling during valve operation
- Technical Specifications

→(DRN 06-1002, R15)

\* The pressurizer is analyzed for 200 Plant heatup and cooldown cycles.

←(DRN 06-1002, R15)

#### 5.2.2.10 Testing & Inspection

Testing and inspection of the primary safety valve is governed by ASME Section XI, Subsection IWW. Testing and inspection of the main steam safety valves is discussed in Subsection 10.3.4.

### 5.2.3 REACTOR COOLANT PRESSURE BOUNDARY MATERIALS

#### 5.2.3.1 Material Specifications

A list of specifications for the principal ferritic materials, austenitic stainless steels, bolting and weld materials, which are a part of the reactor coolant pressure boundary is given in Tables 5.2-3 and 4.

To reduce sensitivity to neutron-induced changes in service, low residual requirements for copper, phosphorus, and vanadium were imposed on plate and weld materials in the reactor vessel beltline. The core beltline region, as defined by Appendix G of 10CFR50, includes the intermediate and lower shell courses and their longitudinal weld seams. Also included is the girth seam joining these two shell courses.

The chemical content of the reactor vessel beltline material as determined by chemical analysis is given in Table 5.2-5.

#### 5.2.3.2 Compatibility with Reactor Coolant

##### 5.2.3.2.1 Reactor Coolant Chemistry

Controlled water chemistry is maintained within the RCS. Control of the reactor coolant chemistry is the function of the Chemical and Volume Control System which is described in Subsection 9.3.4. Water chemistry limits applicable to the RCS are given in Subsection 9.3.4.

##### 5.2.3.2.2 Materials of Construction Compatibility to Reactor Coolant

→(DRN 00-1059)

The materials of construction used in the RCPB which are in contact with reactor coolant are designated by an "a" in Table 5.2-3. These materials have been selected to minimize corrosion and have previously demonstrated satisfactory performance in other existing operating reactor plants.

##### 5.2.3.2.3 Compatibility with External Insulation and Environmental Atmosphere

←(DRN 00-1059)

The possibility of leakage of reactor coolant onto the reactor vessel head or other part of the reactor coolant pressure boundary causing corrosion of the pressure boundary has been investigated by C-E.

Tests have shown that RCS leakage onto surfaces of the reactor coolant pressure boundary will not affect the integrity of the pressure boundary.

→(DRN 00-1059; 02-88)

The reactor vessel and closure head are insulated with stainless steel reflective insulation or Owens-Corning Fiberglas nuclear blanket type thermal insulation qualified per Reference 1, to minimize insulation contamination in the event of active solution spillage. The reactor vessel supports are not insulated. Removable panels of insulation are provided on the closure head, on the vessel lower head, and around the reactor vessel nozzles as required to allow access for in-service inspection of weld areas.

←(DRN 00-1059; 02-88)



→(DRN 00-1059, R11-A; 02-88, R11-A)

←(DRN 00-1059, R11-A; 02-88, R11-A)

In the local areas around stainless steel and nickel-based alloy nozzles in the reactor vessel head, some small plugs of mineral wool insulation encapsulated in fiber glass cloth may be used. The C-E specification for the mineral wool/fiber glass insulation limits the amount of leachable halides in accordance with Regulatory Guide 1.36 (2/23/73). The amount of mineral wool contained in these small plugs would not be sufficient to restrict the openings in the Safety Injection System sump screens.

### 5.2.3.3 Fabrication and Processing of Ferritic Materials

#### 5.2.3.3.1 Fracture Toughness

→(LBDCR 17-002, R310)

Testing and measuring equipment for fracture toughness tests for the reactor vessel, steam generators, pressurizer, and reactor coolant pumps were calibrated in accordance with Paragraph NA-4600 of the 1971 ASME Code Section III, through Summer 1971 Addenda. Testing and measurement equipment for piping fracture toughness tests were calibrated in accordance with Paragraph NB-2360 of the 1971 ASME Code Section III through Summer 1972 Addenda.

←(LBDCR 17-002, R310)

Wherever possible, the tests and acceptance requirements of 10CFR50, Appendix G, were applied to the primary system pressure boundary ferritic materials, bolting and weld materials used for fabrication of the reactor vessel, steam generators (primary side), pressurizer, and 42 in. and 30 in. reactor coolant piping.

→(EC-1020, R307; EC-2800, R307, LBDCR 17-002, R310)

These materials, except for the Replacement Reactor Vessel Closure Head (RRVCH) material, were ordered to earlier code requirements (see Table 5.2-1) and, therefore, some of the additional tests required by 10CFR50, Appendix G, were not performed. The CEDM fracture toughness requirements comply with 10CFR50, Appendix G with no application of BTP MTEB 5-2. The reactor vessel beltline and extended beltline plate and weld materials also comply with 10 CFR 50, Appendix G with no application of BTP MTEB 5-2; however, neither full Charpy transition curves nor some dropweight tests as required by Appendix G were obtained for weld materials. For the welds, adequate Charpy data was obtained to determine  $RT_{NDT}$  from a combination of sources, including generic Linde type 0091 flux  $RT_{NDT}$  of 10 CFR 50.61, sister vessel weld data, and available dropweight data. USE were determined by either conservatively taking the 10°-20°F Charpy energy as the USE or using full Charpy transition curve data.

←(LBDCR 17-002, R310)

→(LBDCR 17-002, R310)

←(LBDCR 17-002, R310)

→(DRN 00-1059, R11-A; EC-1020, R307, EC- LBDCR 17-002, R310)

To present the fracture toughness data as required by 10CFR50, Appendix G, the available test data for the remaining reactor coolant pressure boundary materials were evaluated according to Branch Technical Position MTEB 5-2, "Fracture Toughness Requirements." The available fracture toughness data are reported in Tables 5.2-6 through 5.2-9. This approach results in a downgrading of the material fracture toughness properties and provides more conservatism than if the testing were performed in accordance with 10CFR50, Appendix G. Charpy V-notch results are shown in Figures 5.2-2 through 5.2-30, except for the RRVCH forging for which results are shown in Table 5.2-6. Footnotes in Table 5.2-6 through 5.2-9 indicate the sections of MTEB 5-2 that were used in evaluating the data. The various tests were performed in accordance with the applicable ASME Code and applicable Addenda, as noted in Table 5.2-1.

←(EC-1020, R307, LBDCR 17-002, R310)

→(DRN 02-218, R11-A)

The SA 516 Gr. 70 plate material for the RCS piping meets the fracture toughness requirement of Table I-1.1, Appendix I of the Summer 1971 Addenda of the ASME Code (20 ft-lbs avg.). The methods of MTEB 5-2, which allow the development of an  $RT_{NDT}$  for materials exhibiting a fracture toughness of at least 30 ft-lbs absorbed energy were applied. Footnotes in Table 5.2-7 indicate which MTEB 5-2 sections were utilized

←(DRN 00-1059, R11-A; 02-218, R11-A)

→(DRN 00-1059, R11-A; 02-218, R11-A)

in evaluating the piping. For those plates which exhibited fracture toughness energies between 20 and 30 ft-lbs a generic basis to establish a conservative  $RT_{NDT}$  was utilized. These plates are denoted by a footnote D in Table 5.2-7. The Charpy energy absorbed vs. temperature and the mils lateral expansion vs. temperature for the Waterford 3 material and for the 25 additional heats of 516 Gr- 70 plate from Southern California Edison Co.'s San Onofre Units 2 and 3, were plotted to establish the highest temperature necessary to achieve 50 ft-lbs absorbed energy and 35 mils lateral expansion. (Figs. 5.2-29 and 5.2-30, respectively). The highest temperature necessary to achieve 50 ft-lb absorbed energy is 118°F and the highest temperature necessary to achieve 35 mils lateral expansion is 96°F. A conservative  $RT_{NDT}$  for this material is 58°F (T 50 ft-lbs - 60).

←(DRN 00-1059, R11-A; 02-218, R11-A)

Because reactor vessel beltline materials are subject to neutron induced changes in mechanical properties, 10CFR50 Appendix G Section 3C requires that additional fracture toughness tests be performed. These materials were not tested in full accordance with 10CFR50, Appendix G.

→(LBDCR 17-002, R310)

Testing of weld and weld heat affected zone (HAZ) materials had not been required by the applicable code to which the materials were ordered; however, additional base metal testing was conducted with test specimens prepared from longitudinal (strong direction) and transverse (weak direction) material. Transverse (weak direction) tests on base metal, welds, and HAZ materials for the reactor vessel beltline surveillance materials have also been made as part of the Waterford 3 reactor vessel material surveillance program, (described in Subsection 5.3.1.6).

←(LBDCR 17-002, R310)

#### 5.2.3.3.2 Control of Welding

→(DRN 06-872, R15)

##### 5.2.3.3.2.1 Avoidance of Cold Cracking

←(DRN 06-872, R15)

→(DRN 00-1059, R11-A)

Waterford 3 components comply with NRC Regulatory Guide 1.50, Control of Preheat Temperature for Welding of Low Alloy Steel, May 1973, except for Part C, Paragraphs 1.b and 2. The strict interpretation of Paragraph 1.b would imply that the qualification plates are an infinite heat sink that would instantaneously dissipate the heat input from the welding process. The procedure qualification consists of starting the welding at the minimum preheat temperature. Welding is continued until the maximum interpass temperature is reached. At this time, the test plate is permitted to cool to the minimum preheat temperature and the welding is restarted. Preheat temperatures utilized for low alloy steel are in accordance with Appendix D of Section III of the ASME Code. The maximum interpass temperature utilized is 500°F. This position applies to the steam generators, reactor vessels, 42 in. and 30 in. RCS piping and pressurizer.

←(DRN 00-1059, R11-A)

The paragraph 2 requirement is considered an unnecessary extension of present NSSS vendor procedures, which continue to produce low alloy steel welds meeting ASME Code Sections III and IX requirements. The requirements of Regulatory Guide 1.50 are met by compliance with Paragraph 4. The soundness of all welds is verified by ASME Code acceptable examination procedures.

With regard to Regulatory Guide 1.43 (May 1973), major RCS components are fabricated with corrosion resistant cladding on internal surfaces exposed to reactor coolant. The major portion of the material protected by cladding from exposure to reactor coolant is SA-533, Grade B, Class 1 plate which, as discussed in the Regulatory Guide, is immune to underclad cracking. Cladding performed on SA-508, Class 2 forging material is performed using low-heat input welding processes controlled to minimize heating of the base metal. Low-heat-input welding processes are not known to induce underclad cracking.

→(EC-1020, R307)

The Replacement Reactor Vessel Closure Head (RRVCH) is fabricated from SA-508 Grade 3 Class 1 forging material. This material is considered to be resistant to underclad cracking.

←(EC-1020, R307)

#### 5.2.3.3.2.2 Compliance with Regulatory Guide 1.34

Regulatory Guide 1.34 (December 28, 1972) addresses controls to be applied during welding using the electroslag process. The electroslag process has not been used in the fabrication of any RCPB components. Therefore, the recommendations of this guide are not applicable.

#### 5.2.3.3.2.3 Compliance with Regulatory Guide 1.71

Waterford 3 does not comply with the specific requirements of Regulatory Guide 1.71 (December 1973). Performance qualifications, for personnel welding under conditions of limited accessibility, are conducted and maintained in accordance with the requirements of ASME Code Sections III and IX. A requalification is required when (1) any of the essential variables of Section IX are changed, or (2) when authorized personnel have reason to question the ability of the welder to satisfactorily perform to the applicable requirements. Production welding is monitored for compliance with the procedures parameters and welding qualification requirements are certified in accordance with Sections III and IX. Further assurance of acceptable welds of limited accessibility is afforded by the welding supervisor assigning only the most highly skilled personnel to these tasks. Finally, weld quality, regardless of accessibility, is verified by the performance of the required nondestructive examination.

#### 5.2.3.3.2.4 Compliance with Regulatory Guide 1.66

All tubular products used for components of the RCPB (except the three components noted below) are nondestructively examined in accordance with the requirements of ASME B&PV Code, Section III, Division 1, 1974 Edition and Addenda through Summer 1974. In addition, these nondestructive examination requirements are consistent with the recommendations of Regulatory Guide 1.66. (October, 1973)

The three components not consistent with the recommendations of Regulatory Guide 1.66 were ultrasonically tested in accordance with the requirements of the following ASME B&PV Code Addenda for the 1971 Edition: reactor vessel instrument tubing, and heater sleeve tubing - Summer 1971; CEDM upper pressure housing Winter 1973. It is considered that performing the additional ultrasonic testing examination of these components will not provide additional meaningful information on material quality commensurate with safety.

### 5.2.3.4 Fabrication and Processing of Austenitic Stainless Steel

#### 5.2.3.4.1 Avoidance of Stress Corrosion Cracking

##### 5.2.3.4.1.1 Avoidance of Sensitization

#### 5.2.3.4.1.1.1 NSSS Components

Waterford 3 is consistent with the recommendations of Regulatory Guide 1.44 as described in items a through d except for the criteria used to demonstrate freedom from sensitization. The ASTM A-393 Strauss Test was used in lieu of the ASTM A-262 Practice E, Modified Strauss Test to demonstrate freedom from sensitization in fabricated, unstabilized, stainless steel.

→(EC-2800, R307)

For replacement CEDMs, the ASTM A262 Practice E test was used.

←(EC-2800, R307)

##### a) Solution Heat Treatment Requirements

All raw austenitic-stainless steel material, both wrought and cast, in the fabrication of the major NSSS components in the RCPB, is supplied in the annealed condition as specified by the pertinent ASTM or ASME Code; viz., 1850-2050°F for 1/2 to one hour per in. of thickness and water quenched to below 700°F. The time at temperature is determined by the size and type of component. For example, reactor coolant pump casings which are cast from CF8M are usually subject to more than one solution anneal and, therefore, the time at temperature is limited to 1/2-hour per in. of thickness.

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

##### b) Material Inspection Program

Extensive testing on stainless steel mockups, fabricated using production techniques, has been conducted to determine the effect of various welding procedures on the susceptibility of unstabilized 300 series stainless steels to sensitization-induced intergranular corrosion. Only those procedures and/or practices demonstrated not to produce a sensitized structure are used in the fabrication of these RCPB components. The ASTM standard A-393 (Strauss test) is the criterion used to determine susceptibility to intergranular corrosion. This test has shown excellent correlation with a form of localized corrosion peculiar to sensitized stainless steels. As such, ASTM A393 is utilized as a go-no-go standard for acceptability.

→(EC-2800, R307)

For replacement CEDMs, the ASTM A262 Practice E test was used.

←(EC-2800, R307)

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

##### c) Unstabilized Austenitic Stainless Steels

The unstabilized grades of austenitic stainless steel with carbon content of more than 0.03 percent used for components of the RCS are 304 and 316. These materials are furnished in the solution annealed condition. Exposure of completed or partially fabricated components to temperatures ranging from 800°F to 1500°F is prohibited wherever possible. Exceptions may arise where valves containing stellite seats which cannot be quenched are exposed to this temperature range during cooling from hard surfacing.

Duplex, austenitic stainless steels, containing more than five weight percent delta ferrite (weld metal, cast metal, weld deposit overlay), are not considered unstabilized since these alloys do not sensitize, that is, form a continuous network of chromium-iron carbides. Specifically, alloys in this category are:

→(DRN 06-911, R15)

CF8M Cast stainless steels  
CF8 Cast stainless steels

}Delta ferrite controlled to  
}5-25 v/o

308

}Singly and combined stainless

309

}steel weld filler metals.

312

}Delta ferrite controlled to

316

}5-18 v/o as deposited.

←(DRN 06-911, R15)

Delta ferrite of deposited weld metal or castings exposed to the temperature range of 1000°-1500°F was determined by either a magnetic measurement, chemical analysis in conjunction with the Schaeffler Diagram, or metallographic analysis.

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

d) Avoidance of Sensitization

→(DRN 00-1059, R11-A)

Exposure of unstabilized austenitic 3XX stainless steels to temperatures ranging from 800°-1500°F will result in carbide precipitation. The degree of carbide precipitation, or sensitization, depends on the temperature, the time at the temperature, and also, the carbon content. Severe sensitization is defined as a continuous grain boundary chromium-iron carbide network. This condition induces susceptibility to intergranular corrosion in oxygenated aqueous environments, as well as those containing fluorides. Such a metallurgical structure will rapidly fail the Strauss test ASTM A-393. Discontinuous precipitates (i.e., an intermittent grain boundary carbide network) are not susceptible to intergranular corrosion in a PWR environment.

←(DRN 00-1059, R11-A)

→(EC-2800, R307)

For replacement CEDMs, the ASTM A262 Practice E test was used.

←(EC-2800, R307)

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

- Weld heat input to less than 60 kj/in.
- Interpass temperature to a maximum of 350°F

→(DRN 00-1059, R11-A; 02-218, R11-A)

Homogeneous or localized heat treatment in the temperature range 800-1500°F is prohibited for unstabilized austenitic stainless steel with a carbon content greater than 0.03 used in components of the RCPB. When stainless steel safe ends are required on component nozzles or piping, fabrication techniques and sequencing require that the stainless steel piece be welded to the component after final stress relief. This is accomplished by welding an Inconel overlay on the end of the nozzle. Following final stress relief of the

←(DRN 00-1059, R11-A; 02-218, R11-A)

→(DRN 02-218)

component, the stainless steel safe end is welded to the Inconel overlay, using Inconel weld filler metal.

←(DRN 02-218)

5.2.3.4.1.1.2 Components Other Than NSSS

a) Regulatory Guide 1.44

With respect to other Class 1 components, Waterford 3 is consistent with the recommendations of Regulatory Guide 1.44 as described in Subsection 6.1.1 for the ESF components.

5.2.3.4.1.2 Avoidance of Contaminants Causing Stress Corrosion Cracking

5.2.3.4.1.2.1 NSSS Components

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

Contamination of austenitic stainless steels of the 300 type by compounds which can alter the physical or metallurgical structure and/or properties of the material was avoided during all stages of fabrication. Painting of 300 series stainless steels was prohibited. Grinding was accomplished with resin or rubber-bounded aluminum oxide or silicon carbide wheels which were not previously used on materials other than austenitic-ferrite alloys. Outside storage of partially fabricated components was avoided and, in most cases prohibited. Exceptions were made with certain structures provided they were dry, completely covered with a waterproof material, and kept above ground.

Internal surfaces of completed components, are cleaned to produce an item which was clean to the extent that grit, scale, corrosion products, grease oil wax, gum, adhered or embedded dust or extraneous materials were not visible to the unaided eye. Cleaning was effected by either solvents (acetone or isotropyl alcohol or inhibited water 30-200 ppm hydrazine or 0.5-0.75 weight percent trisodium phosphate). Water conformed to the following requirements:

Halides

Chloride (ppm)	< 0.60
Fluoride (ppm)	< 0.40
Conductivity ( pmhos/cm)	< 5.0
pH	6.0-8.0

Visual clarity                      No turbidity, oil or sediment

Prior to shipment, RCPB components were packaged in such a manner that they were protected from the weather, dirt, wind, water spray, and any other extraneous environmental conditions encountered during shipment and subsequent site storage. The environment within the package and/or component was maintained clean and dry. In some instances, use of a desiccant breather system was utilized. The shipment package was employed for site storage and was not removed until the component was installed within the containment. Once in the containment, with the shipping package removed, the component was maintained clean and dry, either by covering with a polyethylene cover, or placing in a clean area.

To prevent halide-induced, intergranular corrosion which could occur in aqueous environment with significant quantities of dissolved oxygen, solutions were inhibited via additions of hydrazine. Results of tests such as those documented in Reference I have proven this inhibitor to be completely effective. Operational chemistry specifications restrict concentrations of halide and oxygen, both prerequisites of intergranular attacks. (Refer to Subsection 9.3.4).

5.2.3.4.1.2.2 Components Other Than NSSS

Specific requirements for cleanliness and contamination protection are included in the equipment specifications for components fabricated with austenitic stainless steel. The provisions described in Subsection 6.1.1 also apply to the Class 1 components during fabrication, shipment and storage.

5.2.3.4.1.3 Characteristics and Mechanical Properties of Cold-Worked Austenitic Stainless Steels for RCPB Components

Cold-worked austenitic stainless steel is not utilized for components of the RCPB.

5.2.3.4.2 Control of Welding

5.2.3.4.2.1 Avoidance of Hot Cracking

a) NSSS Components

1) Interim Position MTEB 5-1 on Regulatory Guide 1.31

In order to preclude microfissuring in austenitic stainless steel welds, Waterford 3 is consistent with the recommendations of the Interim Position (Branch Technical Position of the Interim Position (Branch Technical Position MTEB 5-1) on Regulatory Guide 1.31, Control of Stainless Steel Welding except for the difference noted below.

→(EC-2800, R307)

The replacement CEDMs conform to Reg. Guide 1.31 Rev. 3, which supersedes BTP MTEB 5-1. See Section 1.8.

←(EC-2800, R307)

(a) Major RCPB Components, Excluding Reactor Coolant Pumps

→(DRN 00-1059, R11-A; 02-88, R11-A; 06-911, R15)

The delta ferrite content of A-7 austenitic stainless steel filler metal, except for 16-8-2, in the fabrication of major components of the reactor coolant pressure boundary has been controlled to 5-15 vol percent. Delta ferrite content was predicted by magnetic measurement or chemical analysis in conjunction with the Schaeffler or McKay Diagram, performed on undiluted weld deposits. In the case of the filler metal used with a non-consumable electrode processes, the delta ferrite content may have been predicted by chemical analysis of the rod, wire or consumable insert in conjunction with the stainless steel constitution diagram.

←(DRN 00-1059, R11-A; 02-88, R11-A; 06-911, R15)

→(DRN 00-1059, R11-A; 02-88, R11-A)

←(DRN 00-1059, R11-A; 02-88, R11-A)

The ferrite requirements was met for each heat, lot, or heat/lot combination of weld filler material.

(b) Reactor Coolant Pumps

The quality and structural adequacy of welds in the reactor coolant pumps were assured by the use of controls on materials, procedures, and personnel. These controls were selected to be pertinent to the component functional safety level required and generally, were imposed through the appropriate ASME B&PV Code referenced in Table 5.2-1.

2) Regulatory Guide 1.34

Regulatory Guide 1.34 is discussed in Subsection 5.2.3.3.2.2 and Appendix 3A.

3) Regulatory Guide 1.71

Regulatory Guide 1.71 is discussed in Subsection 5.2.3.3.2.3 and Appendix 3A.

b) Components Other Than NSSS

1) Regulatory Guide 1.31 is discussed in Subsection 6.1.1.

2) Regulatory Guide 1.34 is discussed in Subsection 5.2.3.3.2.2.

3) Regulatory Guide 1.71 is discussed in Subsection 6.1.1.

5.2.3.4.3 Nondestructive Examination

Nondestructive examination of tubular products is discussed in Subsection 5.2.3.3.

SECTION 5.2.3: REFERENCES

1) Topical Report OCF-1, Nuclear Containment Insulation System, on file with U.S. Nuclear Regulatory Commission.

5.2.4 INSERVICE INSPECTION AND TESTING OF REACTOR COOLANT PRESSURE BOUNDARY

→(DRN 99-0821; 06-872, R15)

An inservice inspection (ISI) program is provided for the examination of the Reactor Coolant Pressure Boundary (RCPB) components and supports defined as Code Class 1. The program reflects the principles and intent embodied in the ASME Boiler and Pressure Vessel Code, Section XI. Specific Code Editions and addenda required by 10CFR50.55a are referenced in the Pre-Service Inspection (PSI) and ISI programs. The purpose of the inservice inspection program is to periodically monitor the systems or components requiring inservice inspection in order to identify and to repair those indications which do not meet acceptance standards.

←(DRN 99-0821; 06-872, R15)



5.2.4.1 System Boundary Subject to Inspection

→(DRN 99-0821; 00-1059, R11-A)

The reactor pressure vessel, pressurizer, primary side of the steam generator, and associated piping, pumps, valves, bolting and-component supports are subjected to inspection. Standard exemptions as applicable are listed in the inservice inspection program.

←(DRN 00-1059, R11-A)

5.2.4.2 Arrangement of Systems and Components to Provide Accessibility

The layout and arrangement of the plant provides adequate working space and access for inspection of specific areas of Code Class 1 components of the RCPB. The Code Class 1 components of RCPB subject to inspection are those components defined by ASME Section XI.

←(DRN 99-0821)

Listed below are the provisions for access for examination of the RCPB:

a) Reactor Vessel and Closure Head

1) From Inside the Vessel:

→(EC-5000082400, R301)

All internals of the reactor vessel (which is an open structure offering insignificant impediment to access) are removable making the entire inner surface of the vessel, as well as the weld zones of the internal load-carrying structure attachments available for the required surface and volumetric inspections. Provisions are made in the plant design to allow for the removal and storage of all vessel internals (except the flow skirt) during inservice inspection. Ultrasonic testing of all reactor vessel welds will be in accordance with 10 CFR 50.55a, with the exception of ASME Exam Category B-A, Item No. B1.30 and B1.40 welds. Examinations of ASME Exam Category B-A, Item No. B1.30 and B1.40 (flange) welds shall meet or exceed the requirements of Regulatory Guide 1.150, Revision 1.

←(EC-5000082400, R301)

2) Closure Head

→(DRN 99-0821)

The closure head as available for inspection whenever it is removed, and its removal makes available the vessel closure flange, the flange-to-shell weld, bolt holes and ligaments, flange studs and nuts.

←(DRN 99-0821)

b) Reactor Coolant Piping

Biological shielding around the reactor coolant piping in the area of the reactor vessel is designed to afford access to the circumferential and longitudinal welds, as well as the transition piece to nozzle welds. The volumetric examinations are performed using ultrasonic techniques.

All reactor coolant piping, as well as major components, excluding the reactor vessel, is provided with removable insulation in the areas of all welds and adjacent base metal requiring examination.

The primary coolant piping has access at each side of the welds to manually examine the welds.

c) Steam Generators

Sufficient space is provided within the stay cylinder to permit inspection of the welds. A 12 in. diameter access opening in the steam generator support skirt is provided. The insulation in this area is removable to the extent of the full size of the access opening.

The steam generators have removable insulation and access at welds requiring examination. Manways are provided for those inspections which must be made internally on the-steam generator.

d) Other Components

All other components, including portions of the steam generators, pressurizer and primary piping, are accessible for manual examination from the outside surface.

The pressurizer has sufficient clearance around the shell weld seam for manual ultrasonic examination of these welds. The insulation is removable at each weld and access is provided for ultrasonic and visual examinations in the area of the bottom head and its nozzle penetrations of the pressurizer. A manway is provided for those inspections which must be made internally on the pressurizer.

→ (DRN 99-0821)

The reactor coolant pumps require inside visual examination only when a pump is disassembled for maintenance, repair, or volumetric examination. Access is provided to the motor flywheels for ultrasonic examination.

← (DRN 99-0821)

General provisions are made for removable insulation, removable shielding, installation of handling machinery, adequate personnel and equipment access space and lay down space for all temporarily removed or serviced components. Storage space for the removable insulation panels is also provided. Working room for a man is provided adjacent to each weld in order to examine all piping system welds manually.

5.2.4.3

Examination Techniques and Procedures

→ (DRN 99-0821)

Examinations include liquid penetrant or magnetic particle techniques when surface examination is specified, ultrasonic or radiographic techniques when volumetric examination is specified, and visual inspection techniques will be used to determine surface condition of components and for evidence of leakage. Specific techniques, procedures, and equipment varies with the contractor chosen to perform the inservice inspection, and will be defined in inservice inspection program. Alternative examination methods, a combination of methods, or newly developed techniques may be substituted for the methods specified as allowed by ASME Section XI.

← (DRN 99-0821)

5.2.4.4 Inspection Intervals

→(DRN 99-0821)

The examination program for the 120 month inspection interval is defined in the ISI Program. The ISI Program for all Code Class 1 systems and components is in accordance with the ASME Section XI, edition and addenda as specified in 10CFR50.55a and as amended by alternatives authorized by the NRC. Subsequent 120 month inspection intervals throughout the service life of the facility will comply, where practical with those requirements in the editions of the Code and addenda in effect 12 months prior to the start of each inspection interval.

←(DRN 99-0821)

5.2.4.5 Categories and Requirements

→(DRN 99-0821)

The inservice inspection program category and examination requirements for the Reactor Coolant Pressure Boundary complies with Section XI. Requests for relief are listed in the inservice inspection plan.

←(DRN 99-0821)

5.2.4.6 Evaluation of Results

The evaluation of nondestructive examination results, acceptance standards and documentation will be in accordance with Section XI.

→(DRN 99-0821)

5.2.4.7 System Leakage Tests

Code Class 1 systems and components are subjected to a system leakage test prior to startup following each reactor refueling outage. Operational limitations during heatup, cool-down, and system pressure testing, are specified in the plant Technical Specifications.

←(DRN 99-0821)

5.2.5 DETECTION OF LEAKAGE THROUGH REACTOR COOLANT PRESSURE BOUNDARY

The reactor coolant pressure boundary (RCPB) Leakage Detection System is designed to detect and identify abnormal leakage within the limits given in the Technical Specifications. The Leakage Detection System is capable of reliably:

- a) Detecting unidentified sources of abnormal leakage as low as 1.0 gpm.
- b) Identifying particular sources of abnormal leakage as low as 1.0 gpm.

→(EC-5000082424, R301)

The RCPB Leakage Detection System is consistent with the recommendations of NRC Regulatory Guide 1.45 (May 1973) with exception to regulatory position C.5; only two of the four Leakage Detection methods will meet the sensitivity requirements of Regulatory Guide position C.5. Leakage Detection System is capable of performing the functions following seismic events that do not require plant shutdown. In addition, the airborne particulate radioactivity monitoring system is designed to remain functional when subjected to the safe shutdown earthquake (SSE).

←(EC-5000082424, R301)

→(EC-19087, R305)

The guidance of Regulatory Guide 1.45 (May 2008) was used for determining the acceptability of the leakage detection instruments and monitoring program for meeting a 0.25 gpm leakage detection capability for the surge line LBB analysis under WCAP-17187-P (Reference 3).

←(EC-19087, R305)

### 5.2.5.1 Leakage Detection Methods

The means provided for leak detection consists of instrumentation which can detect general leakage from the reactor coolant pressure boundary. Through changes in liquid level, flow rate or radioactivity level, specific sources of leakage can frequently be identified. The various methods of detecting leakage (unidentified and identified) are discussed in the following paragraphs.

#### 5.2.5.1.1 Sump Level and Flow Monitoring

The collection of water in the reactor cavity containment sump indicates possible reactor coolant leakage. Reactor Building floor drains and containment fan cooling unit condensate drains are routed to the sump so that water does not accumulate in areas of the containment other than the sump.

→(DRN 00-1059, R11-A; 06-250, R14-B)

Equipment and floor drains are routed through a single eight in. diameter pipe to a measurement tank and from there to the sump. A triangular notch weir is machined on the outlet side of the measurement tank. The flow through the weir causes the level of the measurement tank to correspond to the flow of water into the tank. The measurement tank is fitted with a level transmitter. The measuring tank level is a function of the flow into the tank. The level transmitter sends 4-20 ma dc signal proportional to the tank level to the main control room for signal linearization, recording, input to the plant monitoring computer and annunciator. The alarm is set at one gpm leakage flow above normal as required by the Regulatory Guide 1.45. A second alarm is set at a higher flow rate to alert the Control Room Operator of rising leakage flow. The level transmitter is non-safety-related and capable of performing its function following seismic events up to a safe shutdown earthquake per Regulatory Guide 1.45.

←(DRN 00-1059, R11-A; 06-250, R14-B)

→(DRN 04-1221, R13-A; EC-19087, R305)

A second method of containment sump monitoring utilizes the containment sump level indication to formulate in-flow leakage rates. By maintaining level in the deep pit area of the containment sump, a change in sump level can be converted to an in-leakage flowrate. The containment sump level computer point is provided on the control room Plant Monitoring Computer (PMC) which displays data from the containment sump level transmitter (SP-ILT-6705B) to calculate the level change in the sump over a specified time period. The level change in the sump is converted to a volume change based on the deep portion of the sump pit. The change in sump volume over time is used to conservatively calculate the in-leakage flow rate. The leak rate calculation is based on 10 minutes of previous level data. The calculation is performed and displayed at the PMC scan rate of once every 1 second. Therefore, the calculated computer point is performed every second and it will display a leak rate that is obtained from 10 minutes of previous data. The PMC sump data could be delayed up to 10 minutes during a sump pump run to return sump level to its normal monitoring level or after a PMC restart. The sump level computer point on the PMC is non-seismic, however, transmitter SP-ILT-6705 B is safety-related, seismic qualified and environmentally qualified. This PMC sump level computer point meets the sensitivity requirements of 0.25 gpm unidentified leakage rate in WCAP-17187-P (Reference 3) to prevent potential surge line ruptures.

←(DRN 04-1221, R13-A; EC-19087, R305)

In order to assist the operator to detect the source of leakage, the four containment fan cooler pan drains are piped to the containment sump measuring tank inlet pipe. The presence of flow in each of the drain lines is detected by six flow switches which are monitored by the plant monitoring computer. The following are possible sources of flow in the fan coolers drain:

- a) Normal condensation from the containment air.
- b) Steam pipe rupture.
- c) Component cooling water coil rupture inside of the fan cooler enclosure.

→(DRN 00-1059, R11-A)

All of the above will be detected by the sump measuring tank input flow transmitter.

←(DRN 00-1059, R11-A)

→(DRN 04-1221, R13-A; 03-2059, R14)

#### 5.2.5.1.2 Containment Airborne Particulate Radioactivity Monitoring

The containment atmosphere radiation monitor is designed to provide a continuous indication in the main control room of the particulate, iodine and gaseous radioactivity levels inside the containment.

Radioactivity in the containment atmosphere indicates the presence of fission products due to a Reactor Coolant System leak or leakage of a contaminated secondary fluid system. This system is described in Subsection 12.3.4. High radiation level and alert status alarms are provided in the main control room. Listings of time rate of change in noble gas concentration and time for 10 percent deviation from normal are shown in Table 5.2-10 which are based on a postulated step increase in direct leakage from 0.1 gpm to one gpm at 85 percent of the original thermal rating, 0.1 percent failed fuel, at the end of a 90 day purge cycle before airborne clean-up units are operational (a 10 percent deviation is considered to be a 10 percent change in portion of the analog indicator with the total space between the position at 0.1 gpm and end of scale representing the total scale). The response times indicated represent the worst case.

←(DRN 04-1221, R13-A; 03-2059, R14)

#### 5.2.5.1.3 Primary (Pressurizer) Safety Valves

Leakage through the primary (pressurizer) safety valves is detected by a non safety grade acoustic monitoring system that provides valve position indication and an alarm in the control room. This system is described in Subsection 1.9.23. Backup methods of determining safety valve leakage are as follows:

- a) Discharge Line Temperature - Each of the primary (pressurizer) safety valve discharge lines contain a temperature detector for monitoring valve leakage. The temperature indicator (TI 107/108) and alarm for each of these temperatures are provided in the main control room. The leakage at safety valves will produce a rapidly increasing temperature indication since the discharge piping has a relatively small volume.
- b) Quench Tank Water Level - Since the safety valves discharge to the quench tank, steam leaking through the valves eventually condenses in the quench tank and causes increasing water level and temperature. Level indication (LI 116) and alarm and also temperature indication (TI-116) and alarm are provided in the main control room to detect rise in water level and temperature due to steam entry into the tank.

#### 5.2.5.1.4 Safety Injection and Shutdown Cooling System Leakage During Operation

Leakage of reactor coolant through the safety injection tank check valves (SI 215, 225, 235 and 245) can be detected by:

- a) Safety Injection Tank Water Level: Leakage of reactor coolant to the safety injection tank produces a rising water level in the tank. The level of water in each Safety Injection Tank is monitored by three level transmitters. The level monitoring instrumentation for each Safety Injection Tank, provided in the main control room, consists of three level indicators (LI 311, 312, 313), (LI 321, 322, 323), (LI 331, 332, 333), (LI 341, 342, 343) and two stage alarm to annunciate high and high-high water levels.
- b) Safety Injection Tank Pressure - Since the safety injection tank is a relatively small closed volume with a nitrogen cover gas, the rising water level due to reactor coolant inflow is accompanied by an increasing tank pressure. The pressure in each Safety Injection Tank is monitored by three pressure transmitters.

The pressure monitoring instrumentation for each Safety Injection Tank, provided in the main control room consists of three pressure indicators (PI 311, 312, 313), (PI 321, 322, 323), (PI 331, 332, 333) (PI 341, 342, 343) and two stage alarm to annunciate high and high high tank pressure.

Leakage from the RCPB to the SDCS is detected by measuring the flow from the shutdown cooling relief valves SI-486 and SI-487 (See Figure 6.3-1 Sheet 2 of 2) leakage past the RCPB valves SI-651, SI-652, SI-665 and SI-666 will pressurize the shutdown cooling lines and lift SI-486 or SI-487. The discharge from the shutdown cooling relief valves SI-486 and SI-487 is directed to the containment leak measuring tank. Flow from the containment leak measuring tank is recorded and alarmed in the main control room. Since RCPB leakage to the SDCS is released to the containment, additional leakage detection is provided by one or more of the indications listed in Subsection 5.2.5.2 and by an increased demand for RCS makeup water.

Leakage from the RCPB to the SIS is detected by the pressure transmitters located on the low pressure side of SIS line check valves SI-217, SI-227, SI-237, and SI-247 (see Figure 6.3-1 Sheet 2 of 2), and indication is provided in the main control room by PI-319, PI-329, PI-339, and PI-349. High pressure is alarmed in the main control room.

Leakage past hot leg injection check valves 1SI-V2507 or 1SI-2509 is detected by the pressure transmitters located on the low pressure side of these valves. Indication is provided in the main control room by PI-390 and PI-391. High pressure is alarmed in the main control room.

Leakage past valves SI-618, SI-628, SI-638 and SI-648 and SI-611, SI-621, SI-631 and SI-641 is detected by loss of water level in the SI tanks. Low water level in the SIT's is indicated and alarmed in the main control room.

Leakage past SIS line second check valves SI-113, SI-114, SI-123, SI-124, SI-133, SI-134, SI-143, and SI-144, and past SIS header isolation valves SI-615, SI-616, SI-617, SI-625, SI-626, SI-627, SI-635, SI-636, SI-637, SI-645, SI-646 and SI-647 is detected by HPSI and LPSI header pressure sensors. Pressure indication is provided in the main control room by PI-306, PI-307, PI-308, and PI-309. RCPB leakage to the HPSI and LPSI system will also increase the demand for PCS makeup water.

#### 5.2.5.1.5 Heat Exchanger

Leakage of reactor coolant through the letdown heat exchanger and reactor coolant pump seal heat exchanger and thermal barrier can be detected by any combination of the following:

→(DRN 00-1059)

- a) Component Cooling Water System radiation - Heat exchanger leaks will produce in-leakage of reactor coolant and fission products into Component Cooling Water System. Such in-leakage increases the normally low radiation level in the system and can be detected by the radiation detectors (Tags No RE-CC-7050A, RE-CC-7050B) in the recirculation lines from the component cooling water heat exchangers. These detectors are indicated and alarmed both locally and in the main control room. Recording is done in the main control room. All channels are seismically qualified.

←(DRN 00-1059)

Complete dispersion of only one gallon of primary coolant throughout the volume of approximately 69,000 gallons of the component cooling water system is sufficient to cause early detectable rapid change in detector reading provided there is no residual radioactivity already present in CCW fluid. In this case the limit on detection is the transport time around the Component Cooling Water System loop. The longest time a volume of coolant leakage would have to travel before reaching the detector is 3.5 minutes. The true detection time however is based both on component cooling water radiation being directly proportional to the product of percent failed fuel and leak rate, and the amount of residual radiation already in the system. For a change in leak rate from an existing 0.1 gpm to 1.0 gpm with 0.1 percent failed fuel, the elapsed time for a 10 percent change is approximately three hours.

- b) Component cooling surge tank level - Leakage of reactor coolant increases the inventory in the component cooling system, causing an increase in the surge tank level. Level switch LS-CC-7010S provides a high level alarm in the main control room. Local indication of tank water level is provided by gage glasses LG-CC-7010A and B.

#### 5.2.5.1.6 CVCS Leakage

Intersystem leakage between the RCPB and the CVCS is not monitored since the CVCS is in operation when the RCS is pressurized, and is thus processing fluid.

However, the CVCS can be used to identify any leakage from the RCS by observing makeup flowrates to the volume control tank for the purposes of identifying gross leakage over an extended period of plant operation. Leakage can also be identified through special testing in which leak rates are monitored by detecting level changes within the volume control tank; this sort of special testing is conducted in order to identify the particular source of the RCS leak. Basically, it would involve securing the makeup source to the volume control tank, securing and sampling of the RCS or CVCS, securing boration or dilution of the RCS, and recording the difference in the water inventory of the volume control tank over a set period of time.

An important means of detecting abnormal leakage from the RCS is through measurement of the net amount of makeup flow to the system. Since all normal sources of outflow from the system such as letdown flow and coolant pump controlled bleed off are collected and recycled back to the RCS by the Chemical and Volume Control System (CVCS) described in Subsection 9.3.4, the net inventory in the RCS and CVCS under normal operating conditions will be constant. Transient changes in letdown flow rate or RCS inventory can be accommodated by changes in the volume control tank level. The net makeup to the system under zero leakage steady state conditions should be essentially zero. The makeup flow rates from CVCS is continuously monitored and recorded. Analysis of the makeup flow record over a period of steady state operation can provide detection of abnormal leakage. Any increasing trend in the amount of makeup required indicates a leak which is increasing in rate. Suddenly occurring leaks are indicated by a step increase in the amount of makeup which does not decrease as would be the case for a purely transient condition.

The maximum capacity of the Reactor Coolant Makeup System is 132 gpm (three 44 gpm charging pumps) which gives a ratio of maximum allowable leakage to makeup of 1/132.

Numerous methods for identifying intersystem leaks for the CVCS are available. These methods are exemplified below:

- a) Decrease in volume control tank level via LIC226; control room alarm and indication is provided for this measurement channel.
- b) Increase in charging flow to maintain pressurizer level; charging flow is monitored by F1212 which provides control room indication; pressurizer level is monitored by LRC110X and LRC110Y in the main control room and alarm annunciation is also provided in the main control room.
- c) Regenerative heat exchanger (RHX) and letdown heat exchanger (LHX) interfaces may show increase in temperature, pressure or activity; CVCS-related instruments for RHX leakage monitoring include TIC221 (control room alarm and indication), PI212 (control room alarm and indication); CVCS-related instruments for LHX leakage monitoring include TIC223 (control room indication), TIC224 (control room alarm and indication) and PIC201 (control room indication). Increase in activity within the CCW system (interface with the LHX) is detectable by monitor within that cooling system.

#### 5.2.5.1.7 Reactor Coolant Pump Seals

Instrumentation is provided to detect abnormal seal leakage. The reactor coolant pumps are equipped with three stages of seals plus a vapor or back-up seal as described in section 5.4. During normal operation, the Reactor Coolant System operating pressure is decreased through the three seals to approximately CVCS volume control tank pressure. The vapor or backup seal prevents leakage to the containment atmosphere and allows sufficient pressure to be maintained to direct the controlled seal leakage to the volume control tank. The vapor or backup seal is designed to withstand full Reactor Coolant System pressure in the event of failure of any or all of the three primary seals.

The following conditions are postulated to exist prior to the unlikely event of a vapor (backup) seal failure:

- a) The lower, middle and upper seal have failed;
- b) The excess flow check valve has closed;
- c) The reactor coolant pump has been stopped;
- d) The pressure at the vapor seal is Reactor Coolant System pressure.

→(EC-6256, R302)

In the event of a failure, and an excess flow condition exists through the vapor seal, with resultant pressure decrease downstream of the middle seal because of seal differential pressure. The reactor coolant pump seal pressure gives this indication. The seal temperature indicator also shows an increase in temperature and increase in seal leak-off flow to the reactor drain tank or to the containment sump via the floor Drain System and/or increase in controlled bleed-off flow to the volume control tank is indicated. Abnormal seal leakage also is indicated by an increased temperature of the component cooling water from the reactor coolant pump seal. An alarm in the controlled bleed-off line is provided for high temperature.

←(EC-6256, R302)



5.2.5.1.8 Steam Generator Tube Leakage

→(DRN 01-3692, R12)

An increase in radioactivity indicated by the condenser vacuum pump exhaust radiation monitors, the steam generator blowdown radiation monitors, and the main steam line N-16 Sodium Iodide monitors will indicate reactor coolant leakage to the secondary side. Routine analysis of steam generator water samples would also indicate increasing leakage of reactor coolant.

←(DRN 01-3692, R12)

5.2.5.1.9 Reactor Vessel Head Closure Leakage

The space between the double O-ring seal is monitored to detect an increase in pressure, which indicates a leak past the inner O-ring. Alarm of this condition is available in the main control room.

5.2.5.1.10 Reactor Coolant Pump Flange Closure Leakage

→(DRN 02-317, R12)

The Reactor Coolant Pump case and pump cover / driver mount is sealed by an inner and outer gasket. Reactor Coolant Pump leak-off into the annulus between these two gaskets may be aligned to pressure switches, the Reactor Drain Tank, or isolated from the pressure switches or the Reactor Drain Tank.

←(DRN 02-317, R12)

→(EC-19087, R305)

5.2.5.1.11 Control Room Leakage Monitoring

Waterford has implemented RCS unidentified leakage monitoring and action levels in accordance with the guidance of WCAP-16465, (Reference 4). The PWR Owners Group concluded that leak rate measurements can reveal small leaks (< 0.1 gpm) when data is recorded for a sufficient period of time. WCAP-16465 established RCS unidentified leakage trending and action levels for three conditions during normal plant operation. This includes monitoring absolute unidentified leak rate (in gpm), deviation from the baseline mean (in gpm), and total integrated unidentified leakage (in gallons). The absolute unidentified leak rate action levels which a direct indication of RCS unidentified leakage are established at:

- One seven (7) day rolling average of daily unidentified RCS leak rates > 0.1 gpm.
- Two consecutive daily unidentified RCS leak rates > 0.15 gpm.
- One daily unidentified RCS leak rate > 0.3 gpm.

Waterford trends RCS normal unidentified leakage at levels below 0.1 gpm. The action level of 0.1 gpm is one tenth of the TS Limit for unidentified leakage which ensures that early detection of changes in RCS unidentified leakage will be identified and addressed prior to TS limiting conditions for operation are reached.

←(EC-19087, R305)

5.2.5.2 Indication in Main Control Room

The primary indications of reactor coolant leakage are:

- a) High containment sump flow alarm
- b) Very high containment -sump flow alarm
- c) Containment airborne radioactivity monitor indication (particulate and iodine and gaseous)
- d) High containment particulate radioactivity alarm
- e) Deleted
- f) Deleted

→(DRN 04-1221, R13-A)

←(DRN 04-1221, R13-A)

Other main control room instrumentation that indicates significant reactor coolant leakage includes:

→(DRN 00-1059, R11-A)

- a) Temperature detectors downstream of primary (pressurizer) safety valves (M-107/108)
- ←(DRN 00-1059, R11-A)
- b) Primary safety valves acoustic position monitors
- c) Safety injection tank level indication (LI-311/321, LI-331/341)
- d) High and high-high safety injection tank levels alarm
- e) Safety injection tank pressure indication and high pressure alarm
- f) CCW Radiation indication
- g) CCW Surge Tank water level indication (LI-CC7010A, LI-CC7010B)
- h) Steam generator radiation indication
- i) Condenser vacuum pumps exhaust radiation indication
- i) Safety injection check valve leakage pressure indication and alarm
- k) Safety injection header high pressure

#### 5.2.5.3 Limits for Reactor Coolant Leakage

The limits for both total and unidentified leakage are described in the Technical Specifications.

#### 5.2.5.4 Unidentified Leakage

→(EC-19087, R305)

The anticipated normal total unidentified Reactor Coolant System leakage is <0.1 gpm as discussed in Section 5.2.5.1.11.

←(EC-19087, R305)

There is no practical analytical method available by which a leak rate can be correlated with crack size. Use of mathematical models to relate reactor coolant leakage to crack size requires assumptions regarding crack geometry and the number of leak sources. If it is assumed that the total leakage is from a single source, and that the crack can be treated, for example, as a square edged orifice, then the methods of references (1) and (2) would show that a through wall crack having an equivalent diameter of approximately 0.04 to 0.05 in. would result in a one gpm leak rate at operating pressure which is the maximum allowable leakage rate from unidentified sources.

For reactor coolant piping, the material defect acceptance criteria per NB-2532.1, Section III of the ASME Code, permits an indication of up to three in. It is thus conceivable that a crack up to three in. in length could exist beneath such a laminar condition and remain undetected.

By the methods of fracture mechanics it can be shown that a through wall crack three in. in length would be approximately 12 percent of the critical crack length for an axial crack and about eight percent of the critical crack length for a circumferential crack.

#### 5.2.5.5 Maximum Allowable Total Leakage

The maximum allowable leakage rate from unidentified sources will be limited to one gpm as specified in the Technical Specifications.

→(EC-19087, R305)

The basis for the proposed one gpm leakage rate from unidentified sources in the reactor coolant system is that this rate can be readily detected and appropriate action taken prior to constituting a potential safety hazard.

←(EC-19087, R305)

The maximum allowable total leakage rate from an identified and evaluated leak will be limited to 10 gpm as specified in the Technical Specifications. This is well within the 44 gpm capacity of one charging pump. The 10 gpm leakage rate is based upon the ability of one charging pump to makeup reactor coolant leakage and still maintain a reasonable makeup margin (34 gpm)-

5.2.5.6 Differentiation Between Identified and Unidentified Leaks

RCS leakage is categorized as identified and unidentified leakage. Identified leakage is:

- a) Leakage into closed systems such as pump seal, safety valve, and valve packing leaks that are captured and directed so that their flowrates are known.
- b) Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of the unidentified leakage monitoring systems or not to be from flows in the RCPB.

→(EC-5000082424, R301)

All other leakage is unidentified leakage. Since identified leakage is known, its affect upon the various leakage detection systems also is known. An increase in leakage, resulting from unidentified leakage, is detected by the leakage detection systems. At least two of the systems are capable of responding to a one gpm leakage in one hour or less.

←(EC-5000082424, R301)

The containment air particulate and radioactive gas monitors provide the primary means of remotely identifying the source of leakage within the reactor building. If sump flow indicators detect leakage above normal without a corresponding increase in airborne activity level, the indicated source of leakage probably is a nonradioactive system.

In order to identify leaks from the RCPB to the secondary side of steam generator and to locate the general area of the leak, each steam generator has a sampling system. The sampling system is tapped off the blowdown line of each steam generator. Specimens from each steam generator are analyzed for radioactivity and chemistry to determine the integrity of the primary to secondary boundary within the steam generators.

Leakage from the RCS into the Component Cooling Water System is detectable by an increase in water level in the component cooling water surge tanks.

5.2.5.7 Testing and Inspection

Preoperational testing consists of calibrating the instruments, testing the automatic controls for activation at the proper set points and checking the operability and limits of alarm functions. Radiation detectors can be remotely checked against a standard source during normal operation.

Normal leakage rates will be identified at the early stages of plant operation by the makeup water data. The normal operating levels will be compared with the identified leakage and used to verify the sensitivity of the instrumentation.

Table 5.2-11 indicates the inservice inspection that will be performed on all valves in the HPSI, LPSI and RHR systems which form the pressure boundary for the RCS.

5.2.5.8 Leakage Checks During Shutdown

Leakage of reactor coolant is checked during shutdowns in the following manner:

- a) Prior to reactor startup following each refueling outage, pressure retaining components of the reactor coolant pressure boundary will be visually examined for evidence of reactor coolant leakage while the system is under a test pressure of not less than the nominal system operating pressure at rated power.
  - These examinations, which need not require removal of insulation, will be performed by inspecting the exposed surfaces and joints of insulation, and the floor areas, or equipment directly underneath these components.
  - ←
  -
- b) During the conduct of these examinations, particular attention will be given to the insulated areas of components constructed of ferritic steel to detect evidence of boric acid residues resulting from reactor coolant leakage which may have accumulated during the service period preceding the refueling outage.
  - ←
  -
- c) These examinations will be performed in accordance with ASME Section XI.
  - ←
  -
- d) The source of any reactor coolant leakage detected by these examinations will be located by the removal of insulation where necessary and the following corrective measures applied:
  - ←
  - 1) Normally expected leakage from component parts (e.g., valve stems) will be minimized by appropriate repair and maintenance procedures. Where such leakage may reach the surface of ferritic components of the reactor coolant pressure boundary, the leakage will be suitably channeled away from ferritic components.
  - 2) Leakage from through wall flaws in the pressure retaining membrane of a component shall be eliminated, either by corrective repair or by component replacement.
  -
- e) If boric acid residues are detected by these examinations, insulation from ferritic steel components will be removed to the extent necessary for examination of the component surface wetted by reactor coolant leakage to detect evidence of corrosion and an evaluation of the effect of any corroded area upon the structural integrity of the component will be performed in accordance with Article IWA-5250 of ASME Section XI.
  - ←
  -
- f) Repairs or replacements will be performed in accordance with Article IWA-4000 of ASME Section XI.
  - ←

SECTION 5.2.5: REFERENCES

- (1) Flow of Fluids, Technical Paper No. 410, Crane Co. 1957.
- (2) The Discharge of Saturated Water Through Tubes, H.K. Fauske, Chemical Engineering Progress Symposium Series, Heat Transfer Cleveland, No. 59, Vol. 61.
- (EC-19087, R305)  
(3) WCAP-17187-P, "Technical Justification for Eliminating Pressurizer Surge Line Rupture as the Structural Design Basis for Waterford Steam Electric Station, Unit 3, Using Leak-Before-Break Methodology Revision 0", February 2010.
- (4) WCAP-16465, "Pressurized Water Reactor Owners Group Standard RCS Leakage Action Levels and Response Guidelines for Pressurized Water Reactors", Revision 0, September 2006.
- ←(EC-19087, R305)  
→(LBDCR 17-020, R310)  
(5) C-PENG-ER-004, Rev 0, "Phase II Final Report for the Waterford 3 Reactor Pressure Vessel Plates, Forgings, Welds, and Cladding." October, 1995.
- (6) CDCC58004, "Louisiana Power & Light Waterford Steam Electric Station Unit No. 3 Evaluation of Baseline Specimens, Reactor Vessel Materials Irradiation Surveillance Program." January, 1978.
- ←(LBDCR 17-020, R310)

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TABLE 5.2-1 (Sheet 1 of 2) Revision 310 (12/17)

CODES AND ADDENDA APPLIED

TO THE REACTOR COOLANT PRESSURE BOUNDARY

<p>→(EC-1020, R307, LBDCR 16-007, R309)                  Reactor vessel (except for the reactor vessel closure head), pressurizer                  ←(EC-1020, R307, LBDCR 16-007, R309)</p>	<p>1. ASME Boiler and Pressure Vessel Code, Section III, Class 1, through Summer 1971 Addenda</p> <p>2. ASME Boiler and Pressure Vessel Code, Section XI, Design Access and Pre-service Inspection, through Summer 1974 Addenda</p>
<p>→(LBDCR 17-002, R310)</p>	<p>3. ASME Boiler and Pressure Vessel Code, Section III, Class 1 through Summer 1973 Addenda (beltline and extended beltline plate And weld material RT<sub>NDT</sub>)</p>
<p>←LBDCR 17-002, R310)</p>	
<p>→(EC-1020, R307)                  Reactor vessel closure head</p>	<p>1. ASME Boiler and Pressure Vessel Code, Section III Class 1, 1998 Edition through Summer 2000 Addenda</p> <p>2. ASME Boiler and Pressure Vessel Code, Section XI, Rules for Inservice Inspection of Nuclear Power Plant Components</p>
<p>←(EC-1020, R307)                  →(EC-8458, R307)                  Steam Generators (primary side)</p>	<p>1. ASME Boiler and Pressure Vessel Code, Section III, Class 1 1998 Edition through 2000 Addenda</p> <p>2. ASME Boiler and Pressure Vessel Code, Section XI, Design Access and Pre-service Inspection, 2001 Edition through 2003 Addenda</p>
<p>←(EC-8458, R307)                  Reactor coolant pump fly-wheels</p>	<p>1. ASME Boiler and Pressure Vessel Code, Section III, Class 1, through Winter 1971 Addenda (Ultrasonic testing only)</p> <p>2. NRC Safety Guide 14 (Reg Guide 1.14 - October 1971)</p> <p>3. ASME Boiler and Pressure Vessel Code, Section XI, Design Access and Pre-service Inspection, through Summer 1974 Addenda</p>
<p>Reactor coolant pump casing</p>	<p>1. ASME Boiler and Pressure Vessel Code, Section III, Class 1, through Winter 1971 Addenda</p> <p>2. ASME Boiler and Pressure Vessel Code, Section XI, Design Access and Pre-service Inspection, through Summer 1974 Addenda</p>
<p>RCS Piping</p>	<p>1. ASME Boiler and Pressure Vessel Code, Section III, Class 1, through Winter 1971 Addenda</p> <p>2. ASME Boiler and Pressure Vessel Code, Section XI, Design Access and Pre-service Inspection, through Summer 1974 Addenda</p>

→(DRN 99-0821)

\*In-service inspection will be in accordance with the Waterford 3 Steam Electric Station Inservice Inspection Plan.

←(DRN 99-0821)

TABLE 5.2-1 (Sheet 2 of 2) Revision 309 (06/16)

CODES AND ADDENDA APPLIED

TO THE REACTOR COOLANT PRESSURE BOUNDARY

Valves (NSSS)	<ol style="list-style-type: none"> <li>1. ASME Boiler and Pressure Vessel Code, Section III, Class 1, through Winter 1971 Addenda and through Summer 1972 Addenda</li> <li>2. Draft ASME Code for Pumps and Valves for Nuclear Power, Class I, through March 1970 Addenda</li> <li>3. ASME Boiler and Pressure Vessel Code, Section XI, Design Access and Pre-service Inspection, through Summer 1974 Addenda</li> </ol>
Valves (Non-NSSS)	<ol style="list-style-type: none"> <li>1. ASME Boiler and Pressure Vessel Code, Section III, Class 1, through Winter 1972 Addenda</li> <li>2. ASME Boiler and Pressure Vessel Code, Section XI, Design Access and Pre-service Inspection, through Summer 1974 Addenda</li> <li>3. ASME Boiler and Pressure Vessel Code, Section III, Class 1, 1974 through Summer 1975 Addenda</li> </ol>
→(LBDCR 15-021, R309)	
←(LBDCR 15-021, R309)	
→(EC-2800, R307)	
Control element drive mechanisms	<ol style="list-style-type: none"> <li>1. ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components, Class 1, 1998 Edition and 2000 Addenda</li> <li>2. ASME Boiler and Pressure Vessel Code, Section XI, Design Access and Pre-service Inspection, 2001 Edition through 2003Addenda</li> </ol>
←(EC-2800, R307)	
Bolting (studs and nuts)	<ol style="list-style-type: none"> <li>1. ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components, Class 1, Summer 1971 Addenda</li> <li>2. ASME Boiler and Pressure Vessel Code, Section XI, Design Access and Pre-service Inspection, through Summer 1974 Addenda</li> </ol>

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TABLE 5.2-2 (Sheet 1 of 3)

Revision 307 (07/13)

APPLICABLE CODE CASES

<u>COMPONENT</u>	<u>CODE CASE</u>	<u>SUBJECT</u>	
Reactor Vessel	1141-1	Foreign Produced Steel.	
	1332-5	Requirements for Steel Forgings, Section III and VIII, Division 2.	
	1344-2	Nickel-Chromium. Age-Hardenable Alloys, (Alloy X-750) Section iii.	
	→(EC-1020, R307) ←(EC-1020, R307)	2142-2	F-Number Grouping for Ni-Cr-Fe Filler Metals Section IX
	←(EC-1020, R307)	1401-1	Welding, Repairs to Cladding of Section III Components After Final Post Weld Heat Treatment.
	→(EC-1020, R307)	1492	Post Weld Heat Treatment Section I, III and VIII, Division I and 2.
	←(EC-1020, R307)	N-698	Design Stress Intensities and Yield Strength Values for UNS N06690 with a Minimum Specified Yield Strength of 35 ksi (240 MPa)
Steam Generators	1557	Steel Products Refined by Secondary Remelting.	
	1332-4, 5	Requirements for Steel Forgings, Section III and VIII, Division 2.	
	1459-1	Welding Repairs to Base Metal of Section III Components After Final Post Weld Heat Treatment	
Pressurizer	→(DRN 00-1631) ←(DRN 00-1631)	474-2	Design Stress Intensities and Yield Strength Values for UNS N06690
	←(DRN 00-1631)	1361-1	Socket Welds, Section III.
		2142-1	F-Number Grouping for Ni-Cr-Fe, Classification UNS N06052 Filler Metal
		2143-1	F-Number Grouping for Ni-Cr-Fe, Classification UNS W86152 Welding Electrode



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TABLE 5.2-2 (Sheet 2 of 3)

Revision 307 (07/13)

APPLICABLE CODE CASES

<u>COMPONENT</u>	<u>CODE CASE</u>	<u>SUBJECT</u>
Reactor Coolant	1604	Hydrostatic Testing of
Pump (casing) → (DRN 00-1631)		Class 1 Pumps.
Piping (Main RCS loops)	474-2	Design Stress Intensities and Yield Strength Values for UNS N06690
←(DRN 00-1631)	1332-6	Requirements for Steel Forgings, Section III and V Division 2
	1401-1	Welding Repairs to Cladding of Section III Components After Final Post Weld Heat Treatment.
	1459	Welding Repairs to Base Metal of Section III Components After Final Post Weld Heat Treatment.
→(DRN 00-1631)	2142-1	Number Grouping for Ni-Cr-Fe, classification UNS N06052 Filler Metal
←(DRN 00-1631) →(EC-2800, R307)	N-4-12	Special Type 403 Modified Forgings or Bars, Class 1 and CS, Section III, Division 1.
	2142-2	F-Number Grouping for Ni-Cr-Fe, Filler Metals, Section IX
←(EC-2800, R307) →(EC-14300, R303)	N-282	Nameplates for Valves, Section III, Division 1, Class 1, 2 and 3 Construction.
Valves	N-24 (1516-2)	Welding of Seats or Minor Internal Permanent Attachments in Valves for Section III Applications
←(EC-14300, R303)	N-242-1	Material Certification, Section III, Division 1, Class 1, 2, 3, MC and CS Construction.
Piping and Supports		

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→(DRN 06-552, R15)

TABLE 5.2-2 (Sheet 3 of 3)

Revision 303 (06/09)

APPLICABLE CODE CASES

COMPONENT

CODE CASE

SUBJECT

N-316

Alternate Rules for  
Fillet Weld Dimensions  
for Socket Welded  
Fittings, Section III,  
Division I, Class 1, 2  
and 3.

N-122

Procedure for Evaluation of the  
Design of Rectangular Cross  
Section Attachments on Class 1  
Piping, Section III, Division 1

N-318

Procedure for Evaluation of the  
Design of Rectangular Cross  
Section Attachments on Class 2  
or 3 Piping, Section III, Division  
1

N-391

Procedure for Evaluation of the  
Design of Hollow Circular Cross  
Section Welded Attachments on  
Class 1 Piping, Section III,  
Division 1

N-392

Procedure for Evaluation of the  
Design of Hollow Circular Cross  
Section Welded Attachments on  
Classes 2 and 3 Piping, Section  
III, Division 1

←(DRN 06-552, R15)

REACTOR COOLANT SYSTEM MATERIALS

<u>Component</u>	<u>Material Specification</u>
Reactor vessel	
Shell →(DRN 05-1400, R14-A; EC-1020, R307)	SA-533 Grade B, Class I Steel
Forgings (except for closure head)	SA-508 Class I or II
Closure Head <sup>(a)</sup>	SA-508, Grade 3, Class I
Cladding	Weld deposited austenitic stainless steel with greater than 5% delta ferrite or NiCrFe alloy
←(DRN 05-1400, R14-A)	
Reactor vessel head <sup>(a)</sup>	SB-166
CEDM Nozzles	
Instrument nozzles <sup>(a)</sup> ←(EC-1020, R307)	SB-166 and SA-182, F-304
Control element drive mechanism housings	
→(DRN 05-1400, R14-A; EC-2800, R307)	
Lower <sup>(a)</sup>	Type 403 Modified Stainless Steel per Code Case N-4-12 Condition 2, with end fittings to SB-166 and upper end fittings to SA-182, F348.
Upper <sup>(a)</sup>	SA-213 Type 316 stainless steel with end fittings of SA-479 Type 316, vent valve seal of ASTM A276 Type 440C stainless steel seat
←(DRN 05-1400, R14-A; EC-2800, R307)	
Closure head bolts	SA-540 B24
Pressurizer	SA-533 Grade B Class I
→(DRN 05-1400, R14-A)	
Shell Cladding <sup>(a)</sup>	Weld deposited austenitic stainless steel with greater than 5 percent delta ferrite or NiCrFe alloy
←(DRN 05-1400, R14-A)	
→(DRN 00-1631)	
Shell <sup>(a)</sup>	A gap exists between the original Inconel 600 and replacement Inconel 690 materials on the repaired instrument nozzles and heater sleeves.
←(DRN 00-1631)	
Forged nozzles	SA-508 Class II
Instrument nozzles <sup>(a)</sup> →(DRN 00-1631; 05-892, R14)	SB-166
Surge and safety valve nozzle safe ends	SA-351, Grade CF8M
Heater sleeves <sup>(b)</sup> →(DRN 05-1400, R14-A)	SB-167 / SB-166
Heater sleeve plug / cap ←(DRN 00-1631; 05-892, R14; 05-1400, R14-A)	SB-167 / SA479 TP304
→(DRN 05-892, R14; 06-911, R15)	
(a) Materials exposed to reactor coolant	
(b) Half-sleeve repair. A remnant of the original sleeve is left in-place	
←(DRN 05-892, R14; 06-911, R15)	

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TABLE 5.2-3 (Sheet 2 of 4) Revision 307 (07/13)

REACTOR COOLANT SYSTEM MATERIALS

<u>Component</u>	<u>Material Specification</u>
→(DRN 06-911, R15) ←(DRN 06-911, R15)	
→(DRN 05-1400, R14-A) Studs and nuts	SA-540 Grade B24 and SA-193 Grade B7
→(EC-8458, R307) Steam generator Primary head	SA-508 Grade 3 Class 2 (forging)
←(DRN 05-1400, R14-A) Primary nozzles and safe ends	SA 508 Grade 3 Class 2
Primary head cladding <sup>(a)</sup>	Weld deposited Stainless Steel with less than 0.10% Cobalt
→(DRN 05-1400, R14-A) Tubesheet	SA-508 Grade 3 Class 2
Structural Divider Plate	SG-168 Alloy UNS N06690
Tube Support Plates	Type 405 Ferritic Stainless Steel
Tubesheet cladding <sup>(a)</sup>	Weld deposited Alloy 690 with less than 0.10% Cobalt
←(DRN 05-1400, R14-A) Tubes <sup>(a)</sup>	SB-163 Alloy 690 TT
→(DRN 05-1400, R14-A) Secondary shell and head	SA 508 Grade 3 Class 2
←(DRN 05-1400, R14-A) Secondary nozzles	SA 508 Grade 3 Class 2
Secondary instrument nozzles	SA-508 Grade 1A
Studs and nuts	SA-193 Grade B7 or SA-194 Grade 7
Hydranuts	SA-540 Grade B23 Class 3
→(EC-8458, R307) Reactor coolant pumps Casing <sup>(a)</sup>	SA-351 GR CF8M
Pump Cover (Lower Flange of Driver Mount)	SA-105
Cladding <sup>(a)</sup>	Austenitic Steel Wire Electrodes Conforming to Requirements of ASME/AWS SFA/A-5.4 and SFA/A-5.9 Type 308 or 309.
→(DRN 05-1400, R14-A) Bolts	SA 540 Gr B23 Class 4 SA-564, Type 630, H-1100 (For seal cartridge and seal heat exchanger)
←(DRN 05-1400, R14-A)	

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TABLE 5.2-3 (Sheet 3 of 4) Revision 14-A (03/06)

REACTOR COOLANT SYSTEM MATERIALS

<u>Component</u>	<u>Material Specification</u>
Nuts	SA 194 Gr 7 SA 564, Type 630, H-1100 (For seal cartridge and seal heat exchanger)
→(DRN 05-1400, R14-A) Heat Exchanger Flange	SA 240 Tp 304 Annealed or SA-182 F304
<b>Reactor Coolant Piping</b>	
Piping (30" and 42")	SA-516 Grade 70* (SA-264 Clad Plate)*
←(DRN 05-1400, R14-A) Cladding <sup>(a)</sup>	SA-240 - 304L
Surge Line (12") <sup>(a)</sup>	SA-351 - CF8M
<b>Piping<sup>(a)</sup></b>	
Pressurizer spray	SA-376, TP-304
Shutdown Cooling Return	SA-376, TP-304
Reactor coolant drain	SA-376, TP-316 or TP-304
→(DRN 05-1400, R14-A) Charging line	SA-376, TP-304
←(DRN 05-1400, R14-A) Safety injection	SA-376, TP-304
Letdown line	SA-376, TP-316 or TP-304
Shutdown cooling bypass	SA-358, TP-304
<b>Piping nozzles and safe ends<sup>(a)</sup></b>	
→(DRN 05-1400, R14-A) Piping safe ends (30")	SA-351 – Grade CF8M
←(DRN 05-1400, R14-A) Surge nozzle forging	SA-105 Grade II
→(DRN 05-1400, R14-A) Surge nozzle safe end	SA-351 – Grade CF8M
←(DRN 05-1400, R14-A) Shutdown cooling outlet nozzle forgings	SA-105-Grade II
→(DRN 05-1400, R14-A) Shutdown cooling outlet nozzle safe ends	SA-351 – Grade CF8M

\*Filler metal used for Field Welds P1OW1 and P1OW2 have been rated with a strength level of 65 ksi per CE Analytical Evaluation Report CENC-1460.

←(DRN 05-1400, R14-A)

REACTOR COOLANT SYSTEM MATERIALS

<u>Component</u>	<u>Material Specification</u>
→(DRN 05-1400, R14-A) Safety injection nozzle forgings	SA-182 - F1
←(DRN 05-1400, R14-A) Safety injection nozzle safe ends	SA-351 – Grade CF8M
Charging inlet nozzle forging	SA-182 - F1
Charging inlet nozzle safe end	SA-182 - F316
Spray nozzle forgings	SA-105 Grade II
Spray nozzle safe ends	SA-182 - F316
Letdown and drain or drain nozzle forgings	SA-105 Grade II
Letdown and drain or drain nozzle safe ends	SA-182 - F316
Sampling or pressure measurement nozzles	SB-166
→(DRN 05-1400, R14-A) Sampling or pressure measurement nozzle safe ends	SA-182 - F316
RTD nozzles	SB-166 and SA-182 F316
Sampling nozzle (surge line)	SA-182-F316
Valves <sup>(a)</sup>	
Body	SA-182 F316, SA-479 Type 316 and SA-351 Grade CF8M
Bonnet	SA-105 Grade II, SA-351 Grade CF8M, SA-479 Type 316, SA-240 Type 316 and SA-182 F316
Disc or Poppet	SA-637 Grade 688, SA-240 Type 316, SA-479 Type 316, SA-182 F316, SA-351 Grade CF8M, SA-351 Grade CF3 and SA-564 Grade 630
←(DRN 05-1400, R14-A)	
→(DRN 03-1707, R13; 06-720, R15)	
←(DRN 03-1707, R13; 06-720, R15)	

TABLE 5.2-4 (Sheet 1 of 2) Revision 307 (07/13)

WELD MATERIALS FOR REACTOR COOLANT PRESSURE BOUNDARY COMPONENTS

➔(DRN 05-1400, R14-A)

<u>Material Specification</u>	<u>Base Material</u>	<u>Weld Material</u>
1. SA-533 Grade B Class 1	SA-533 Grade B Class 1	a. SFA 5.5, (b), E-8018, C3 b. MIL-E-18193, B-4
2. SA-508 Class 2	SA-533 Grade B Class 1	a. SFA 5.5, E-8018, C3 b. MIL-E-18193, B-4
3. SA-508 Class 1	SA-508 Class 2	SFA 5.5, E-8018, C3
4. SA-516 Grade 70	SA-516 Grade 70	SFA 5.1, E-7018 (c)
5. SA-182 F1	SA-516 Grade 70	SFA, 5.1, E-7018
6. SA-105 Grade II	SA-351 CF8M	SFA 5.11, ENiCrFe-3
7. SA-182 F1	SA-351 CF8M	SFA 5.11, ENiCrFe-3
8. SA-105 Grade II	SA-182 F316	SFA 5.11, ENiCrFe-3
9. SB-166	SA-182 F316	Root SFA 5.14, ERNiCr-3 Remaining SFA 5.11, ENiCrFe-3
10. SB-167	SA-182 F304	Root SFA 5.14, ERNiCr-3 Remaining SFA 5.11, ENiCrFe-3
11. SA-516 Grade 70	SA-351 CF8M	SFA 5.11, ENiCrFe-3
12. SA-182 F1	SA-182 F316	SFA 5.11, ENiCrFe-3
13. SB-166	SA-533 Grade B Class 1	SFA 5.11, ENiCrFe-3

➔(EC-2800, R307)

14. SA-182	SB-166	SFA 5.14, ERNiCrFe-7A
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Code Case N-4-12

←(DRN 05-1400, R14-A; EC-2800, R307)

- b) Special weld wire with low residual elements of copper and phosphorus is specified for the beltline region.
- c) Filler metal used for Field Welds PIOW1 and PIOW2 have been rated with a strength level of 65 ksi per CE Analytical Report CENC-1460.

TABLE 5.2-4 (Sheet 2 of 2) Revision 307 (07/13)

WELD MATERIALS FOR REACTOR COOLANT PRESSURE BOUNDARY COMPONENTS

<u>Material Specification</u>	<u>Base Material</u>	<u>Weld Material</u>
→(DRN 05-1400, R14-A)		
→(EC-1020, R307)		
15. SA-516 Grade 70	SA-508 Class 2	a. SFA 5.1, E-7018 b. MIL-E-18193, B-4
←(EC-1020, R307)		
16. Austenitic stainless steel cladding		SFA 5.9, ER-308 SFA 5.9, ER-309 SFA 5.9, ER-312
17. Inconel	Inconel	SFA 5.11, ENiCrFe-3 SFA 5.14, ERNiCr-3
18. SA-182 F-316	SA-508 Class 2	SFA 5.11, ENiCrFe-3
19. SA-351 CF8M	SA-508 Class 2	SFA 5.11, ENiCrFe-3
←(DRN 05-1400, R14-A)		
→(EC-1830, R302)		
20. Inconel	Varies(d)	SFA 5.14, ERNiCrFe-7A
←(EC-1830, R302)		
→(EC-2800, R307)		
21. SA-182 Code Case N-4-12	SA-182 F348	SFA 5.14, ERNiCrFe-7A
22. SA-479	SA-213 F316	SFA 5.9, ER316L
←(EC-2800, R307)		
→(EC-1020, R307)		
23. SA-508 Grade B Class 1	SB-166 UNS N06690	SFA 5.14 ERNiCrFe-7A
←(EC-1020, R307)		

→(DRN 00-331, R11)

When welding SB-166 N06690 or SB-167, SB-06690 base materials, ERNiCrFe-7 and EniCrFe-7 weld materials may be substituted for ERNiCr-3 and EniCrFe-3.

←(DRN 00-331, R11)

b) Special weld wire with low residual elements of copper and phosphorus is specified for the beltline region.

→(DRN 05-1400, R14-A)

c) Filler metal used for Field Welds P1OW1 and P1OW2 have been rated with a strength level of 65 ksi per CE Analytical Report CENC-1460.

←(DRN 05-1400, R14-A)

→(EC-1830, R302)

d) Filler metal used for structural weld overlays of Reactor Coolant System dissimilar metal welds.

←(EC-1830, R302)





WSES-FSAR-UNIT-3

TABLE 5.2-6 (Sheet 1 of 2)

Revision 310 (12/17)

WATERFORD UNIT 3 REACTOR VESSEL FRACTURE TOUGHNESS DATA

Piece Number	Drawing Number	Code Number	Material	Vessel Location	Drop Weight NDTT(°F)	RT <sub>NDT</sub> <sup>(A)</sup> (°F)	Charpy 30 ft-lb. Fix Temp.(°F) Long.	Charpy 50 ft-lb. Fix Temp (°F) Long.	35 Mills Lateral Expansion Temp. (°F) Long	Charpy Upper Shelf Energy (ft-lb) Long.
→EC-1020, R307, LBDCR 17-002, R310) N/A	N149327	N/A	SA508 Gr. 3,CL.1	Closure Head Forging	-44	-44 <sup>(B)</sup>	-94	-75	-75	282 <sup>(C)</sup>
←EC-1020, R307)										
126-101	741701 6103	M-1001-1	SA508 CL-II	Vessel Flange	20	20	-20	-5	2	154
131-102A	741701 6103	M-1013-1	SA508 CL-I	Safe End	10	10	-35	0	-15	148
131-102D	741701 6103	M-1013-2	SA508 CL-I	Safe End	10	10	-35	0	-15	148
131-102C	741701 6103	M-1013-3	SA508 CL-I	Safe End	10	10	-27	0	-32	149
131-102B	741701 6103	M-1013-4	SA508 CL-I	Safe End	10	10	-27	0	-30	149
131-101A	741701 6103	M-1012-1	SA508 CL-I	Safe End	10	10	0	25	-5	146
131-101B	741701 6103	M-1012-2	SA508 CL-I	Safe End	10	10	0	25	-5	146
128-301	741701 6103	M-1011-1	SA508 CL-II	Outlet Nozzle	-20	-20	-37	0	5	99
128-101	741701 6103	M-1010-1	SA508 CL-II	Inlet Nozzle	20	20	-37	10	0	135
128-101	741701 6103	M-1010-2	SA508 CL-II	Inlet Nozzle	20	20	-50	-35	-40	140
128-101	741701 6103	M-1010-3	SA508 CL-II	Inlet Nozzle	10	10	-70	-47	-42	133
128-101	741701 6103	M-1010-4	SA508 CL-II	Inlet Nozzle	30	30	-40	-20	-30	140
128-301	741701 6103	M-1011-2	SA508 CL-II	Outlet Nozzle	0	0	-30	-10	-12	188
124-102	741701 6103	M-1003-1	SA533-B CL-I	Intermediate Shell Plate	-30	-25.1 <sup>(E)</sup>	3.2 <sup>(D)</sup>	34.9 <sup>(D)</sup>	28.2 <sup>(D)</sup>	108 <sup>(D)</sup>
124-102	741701 6103	M-1003-2	SA533-B CL-I	Intermediate Shell Plate	-40	-20 <sup>(E)</sup>	10 <sup>(D)</sup>	40 <sup>(D)</sup>	40 <sup>(D)</sup>	132 <sup>(D)</sup>
124-102	741701 6103	M-1003-3	SA533-B CL-I	Intermediate Shell Plate	-30	-20 <sup>(E)</sup>	10 <sup>(D)</sup>	40 <sup>(D)</sup>	40 <sup>(D)</sup>	111 <sup>(D)</sup>
122-102	741701 6103	M-1002-1	SA533-B CL-I	Upper Shell Plate	-20	-15.4 <sup>(E)</sup>	14.5 <sup>(D)</sup>	44.6 <sup>(D)</sup>	40 <sup>(D)</sup>	104 <sup>(D)</sup>
122-102	741701 6103	M-1002-2	SA533-B CL-I	Upper Shell Plate	-20	-1.4 <sup>(E)</sup>	34.9 <sup>(D)</sup>	58.60 <sup>(D)</sup>	52 <sup>(D)</sup>	95 <sup>(D)</sup>
122-102	741701 6103	M-1002-3	SA533-B CL-I	Upper Shell Plate	-20	-20 <sup>(E)</sup>	10 <sup>(D)</sup>	40 <sup>(D)</sup>	40 <sup>(D)</sup>	120 <sup>(D)</sup>
154-102	741701 6103	M-1007-1	SA533-B CL-1	Bottom Head Torus	-80	-80	-72	-62	-60	174

TABLE 5.2-6 (Sheet 2 of 2)

Revision 310 (12/17)

WATERFORD UNIT 3 REACTOR VESSEL FRACTURE TOUGHNESS DATA

Piece Number	Drawing Number	Code Number	Material	Location Vessel	Drop Weight NDTT(°F)	RT <sub>NDT</sub> <sup>(A)</sup> (°F)	Charpy 30 ft-lb Fix Temp. (°F) Long.	Charpy 50 ft-lb Fix Temp. (°F) Long.	35 Mills Lateral Expansion Temp. (°F) Long.	Charpy Upper Shelf Energy (ft-lb) Long.
152-101	741701 6103	M-1008-1	SA533-B CL-I	Bottom Head Dome	-40	-40	-35	-10	-15	141
→(EC-1020, R307) ←(EC-1020, R307)										
142-101	741701 6103	M-1004-1	SA533-B CL-I	Lower Shell Plate	-40	-37.6 <sup>(E)</sup>	-8.3 <sup>(D)</sup>	22.4 <sup>(D)</sup>	12 <sup>(D)</sup>	135 <sup>(D)</sup>
142-101	741701 6103	M-1004-2	SA533-B CL-1	Lower Shell Plate	-0 <sup>(F)</sup>	0 <sup>(E,F)</sup>	-20.3 <sup>(D)</sup>	47 <sup>(D)</sup>	40 <sup>(D)</sup>	141 <sup>(D)</sup>
142-101	741701 6103	M-1004-3	SA533-B CL-I	Lower Shell Plate	-20	-20 <sup>(E)</sup>	10 <sup>(D)</sup>	10 <sup>(D)</sup>	10 <sup>(D)</sup>	118 <sup>(D)</sup>
←(LBDCR 17-002, R310) →(EC-1020, R307) ←(EC-1020, R307)										

(A)—MTEB Position 5-2 "Fracture Toughness Requirements," Paragraph 1.1(3)(b).

→(EC-1020, R307)

(B) – RT<sub>NDT</sub> per NB-2300 of Section III of the ASME B&PV Code, 1998 Edition through 2000 Addenda.

(C) – Charpy Upper Shelf Energy (Transverse) = 263 ft-lb.

←(EC-1020, R307)

→(LBDCR 17-020, R310)

(D)—Transverse Charpy Result—supplemental C-E fracture toughness testing (Ref 5)

(E)—RT<sub>NDT</sub> per NB-2300 of Section III of the ASME B&PV Code, 1971 Edition through Summer 1973 Addenda

(F)—NDTT for longitudinal specimen used to calculate RT<sub>NDT</sub> for conservatism

←(LBDCR 17-020, R310)

TABLE 5.2-7 (Sheet 1 of 2)

WATERFORD UNIT 3 PIPING MATERIALS FRACTURE TOUGHNESS DATA

Piece Number	Drawing Number	Code Number	Material	Location	Drop Weight NDTT(F)	RT <sub>NDT</sub> °(F)	Test Temp °(F)	Charpy Energy (ft-lbs)				Lat- Exp- (Mils)			
								1	2	3	Avg	1	2	3	Avg
722-108	74470-761-001	M2804-3	SA516GR70	Straight Seg.	NA	30 <sup>B</sup>	+10	40	40	40	40	32	33	33	32.7
722-108	74470-761-001	M2804-5	SA516GR70	Straight Seg.		30 <sup>B</sup>	+10	40	40	40	40	32	33	33	32.7
722-108	74470-761-001	M2804-6	SA516GR70	Straight Seg.		30 <sup>B</sup>	+10	40	40	40	40	32	33	33	32.7
722-108	74470-761-001	M2804-8	SA516GR70	Straight Seg.		30 <sup>B</sup>	+10	40	40	40	40	32	33	33	32.7
742-108	74470-761-001	M2808-1	SA516GR70	Elbow Seg.		58 <sup>D</sup>	+10	22	20	28	23	19	20	26	21.7
742-108	74470-761-001	M2808-2	SA516GR70	Elbow Seg.		30 <sup>B</sup>	+10	46	40	28	38	34	30	40	34.7
742-108	74470-761-001	M2808-3	SA516GR70	Elbow Seg.		58 <sup>D</sup>	+10	24	26	21	23.7	26	17	20	21
742-104	74470-761-001	M2806-1	SA516GR70	Elbow Seg.		30 <sup>B</sup>	+10	31	52	35	39.3	32	34	49	38.3
722-108	74470-761-001	M2804-1	SA516GR70	Straight Seg.		30 <sup>B</sup>	+10	40	40	40	40	32	33	33	32.7
722-102	74470-761-001	M2801-3	SA516GR70	Straight Seg.		30 <sup>B</sup>	+10	37	33	44	38	43	36	35	38
722-102	74470-761-001	M2801-4	SA516GR70	Straight Seg.		30 <sup>B</sup>	+0	37	33	44	38	43	36	35	38
722-102	74470-761-001	M1406-1	SA516GR70	Straight Seg.		10 <sup>A</sup>	+10	62	52	105	73	46	55	82	61
722-102	74470-761-001	M2801-1	SA516GR70	Straight Seg.		58 <sup>A</sup>	+10	24	22	21	22.3	20	20	20	20
722-102	74470-761-001	M2801-2	SA516GR70	Straight Seg.		58 <sup>D</sup>	+10	24	22	21	22.3	20	20	20	20
742-102	74470-742-001	M2805-1	SA516GR70	Elbow Seg.		58 <sup>D</sup>	10	30	23	32	28.3	26	31	30	29
742-106	74470-742-001	M2807-1	SA516GR70	Elbow Seg.		58 <sup>D</sup>	+10	25	28	28	27	21	22	22	21.7
742-108	74470-761-001	M2808-4	SA516GR70	Elbow Seg.		58 <sup>D</sup>	+10	25	26	20	23.7	26	21	27	24.7
722-108	74470-761-001	M2804-2	SA516GR70	Straight Seg.		58 <sup>D</sup>	+10	32	28	24	28	32	22	28	27.3
722-108	74470-761-001	M2804-4	SA516GR70	Straight Seg.		58 <sup>D</sup>	10	32	28	24	28	32	22	28	27.3
722-108	74470-761-001	M2804-7	SA516GR70	Straight Seg.		58 <sup>D</sup>	+10	32	28	24	28	32	22	28	27.3
722-108	74470-761-001	M2809-1	SA516GR70	Elbow Seg.		58 <sup>D</sup>	+10	25	25	31	27	23	22	28	24.3
722-104	74470-761-002	M2802-2	SA516GR70	Straight Seg.		58 <sup>D</sup>	+10	38	30	16	28	38	18	30	28.7

A MTEB Position 5.2, "Fracture Toughness Requirements" Paragraph 1.1 (4) Minimum of 3 tests at a single temp. > 45 ft-lbs.  
 B-MTEB Position 5.2, "Fracture Toughness Requirements" Paragraph 1.1 Minimum of 3 tests at a single temp. > 30 45<ft-lbs.  
 C-MTEB Position 5.2, "Fracture Toughness Requirements" Paragraph 1.1 (3)(b).  
 D-Subsection 5.2.3.3.1

WSES-FSAR-UNIT-3

TABLE 5.2-7 (Sheet 2 of 2)

WATERFORD UNIT 3 PIPING MATERIALS FRACTURE TOUGHNESS DATA

Piece Number	Drawing Number	Number	Material	Location	Drop Weight NDTT(°F)	RT <sub>NDT</sub> (°F)	Test Temp (°F)	Charpy Energy (ft-lbs)				Lat. Exp.(Mils)			
								1	2	3	Avg.	1	2	3	Avg.
722-106	74470-761-002	M2803-2	SA516GR70	Straight Seg.	NA	+58 <sup>D</sup>	+10	38	30	16	28	38	18	30	28.7
722-104	74470-761-002	M2802-3	SA516GR70	Straight Seg.		+30 <sup>B</sup>	+10	52	56	43	50.3	50	48	42	46.7
722-106	74470-761-002	M2803-3	SA516GR70	Straight Seg.		+30 <sup>B</sup>	+10	52	56	43	50.3	50	48	42	46.7
722-104	74470-761-002	M2802-4	SA516GR70	Straight Seg.		+58 <sup>D</sup>	+10	26	33	30	29.7	31	34	33	32.7
722-106	74470-761-002	M2803-4	SA516GR70	Straight Seg.		+58 <sup>D</sup>	+10	26	33	30	29.7	31	34	33	32.7
722-104	74470-761-002	M2802-1	SA516GR70	Straight Seg.		+30 <sup>B</sup>	+10	47	41	47	45	44	44	38	42
722-106	74470-761-002	M2803-1	SA516GR70	Straight Seg.		+30 <sup>B</sup>	+10	47	41	47	45	44	44	38	42
728-301	D-728-003-06	M2810-2	SA182F1	Safety Injection Nozzle Forg.		20 <sup>A</sup>	-20	74	70	90	78	54	51	66	57
728-301	D-728-003-06	M2810-3	SA182F1	Safety Injection Nozzle Forg.		20 <sup>A</sup>	-20	74	70	90	78	54	51	66	57
728-301	D-728-003-06	M2810-4	SA182F1	Safety Injection Nozzle Forg.		0 <sup>B</sup>	-20	50	38	67	51.7	40	33	53	42
728-201	C-728-002-00	M2813-1	SA105-2	Shutdown Cooling Outlet Nozzle		+30 <sup>B</sup>	+10	36	55	39	43.3	40	56	41	45.7
728-201	C-728-002-00	M2813-2	SA105-2	Shutdown Cooling Outlet Nozzle		+30 <sup>B</sup>	+10	35	57	35	42.3	41	56	40	45.7
728-501	D-728-005-00	M2811-1	SA182F1	Charging Inlet Nozzle Forg.		-20 <sup>A</sup>	-20	70	83	75	76	53	59	56	56
728-501	D-728-005-00	M2811-2	SA182F1	Charging Inlet Nozzle Forg.		-20 <sup>A</sup>	-20	70	83	75	76	53	59	56	56
728-301	D-728-003-00	M2810-1	SA182F1	Safety Injection Nozzle Forg.		-20 <sup>A</sup>	-20	55	67	74	65.3	43	53	54	50
728-401	C-728-004-00	M2814-1	SA105-2	Spray Noz. Forg-		+30 <sup>B</sup>	+10	227	43	42	101.	89	41	39	56.3
728-401	C-728-004-00	M2814-2	SA105-2	Spray Noz.Forg.		+30 <sup>B</sup>	+10	227	43	42	101.	89	41	39	56.3
738-601	C-728-006-01	M2815-1	SA105-2	Letdown and Drain Nozzle Forg.		+30 <sup>B</sup>	+10	227	43	42	101.	89	41	39	56.3
728-601	C-728-006-01	M2815-2	SA105-2	Letdown and Drain Nozzle Forg.		+30 <sup>B</sup>	+10	227	43	42	101.	89	41	39	56.3
728-601	C-728-006-01	M2815-3	SA105-2	Letdown and Drain Nozzle Forg.		+30 <sup>B</sup>	+10	227	43	42	101.	89	41	39	56.3
728-601	C-728-006-01	M2815-4	SA105-2	Letdown and Drain Nozzle Forg.		+30 <sup>B</sup>	+10	227	43	42	101.	89	41	39	56.3
728-701	C-728-007-01	M2816-1	SA105-2	Drain Nozzle Forg.		+30 <sup>B</sup>	+10	227	43	42	101.	89	41	39	56.3

A-MTEB Position 5.2 "Fracture Toughness Requirements" Paragraph 1.1 (4), Greater Than 45 ft-lbs.

B-MTEB Position 5.2 "Fracture Toughness Requirements" Paragraph 1.1 Less Than 45 ft-lbs, More Than 30 ft-lbs.

C-MTEB Position 5.2 "Fracture Toughness Requirements" Paragraph 1.1 (3)(b)

D-Subsection 5.2.3.3.1

WSES-FSAR-UNIT-3

TABLE 5.2-8

**WATERFORD 3 PRESSURIZER MATERIALS FRACTURE TOUGHNESS DATA**

Piece Number	Drawing Number	Code Number	Material	Location	Drop Weight NDTT (°F)	RT <sub>NDT</sub> (°F)	Test Temp (°F)	Charpy Energy (ft-lbs.) at 0 Position				Charpy Energy (ft-lbs.) at 180 Position				Lat. Exp. (mils) at 0 Position				Lat. Exp. (Mils) at 180 Position			
								1	2	3	Avg.	1	2	3	Avg.	1	2	3	Avg.	1	2	3	Avg.
658-101	E661-002-03	M2601-1	SA508CLII	Surge Nozzle	-50	-30 <sup>B</sup>	+10	110	104	101	105	106	126	99	110	82	71	74	75.7	73	84	73	76.7
608-101	E661-002-03	M2602-1	SA508CLII	Spray Nozzle	-50	0 <sup>B</sup>	-20	28	34	35	32.3	51	30	25	35.3	29	31	33	31	37	29	25	30.3
608-201	E661-002-03	M2603-1	SA508CLII	Safety Valve Noz	-50	-30 <sup>C</sup>	+10	135	120	97	117	88	119	140	115	89	78	72	79.7	69	82	94	81.7
608-201	E661-002-03	M2603-2	SA508CLII	Safety Valve Noz	-50	-30 <sup>C</sup>	+10	135	120	97	117	88	119	140	115	89	78	72	79.7	69	82	94	81.7
656-101	E661-002-03	M2611-1	SA508CLII	Support Forging	-50	-30 <sup>C</sup>	+10	130	131	121	127	118	97	100	105	96	84	88	89.3	84	76	79	79.7
								Charpy Energy Absorbed (ft-lbs)				Lateral Expansion (Mils)											
								1	2	3	Avg.	1	2	3	Avg.								
642-101	E661-002-03	M2606-2	SA533BCLI	Shell Plate(Lower)	-50	-30 <sup>C</sup>	+10	78	79	73	76.7					53	53	47	51				
236-200	E661-002-03	M2610-1	SA533BCLI	Top Head	+30	+30 <sup>B</sup>	+10	42	36	50	42.7					46	46	36	42.7				
236-200	E661-002-03	M2610-2	SA533BCLI	Bottom Head	+30	+30 <sup>B</sup>	+10	44	35	34	37.7					42	34	83	36.3				
673-102	E661-002-03	M2637-8	SA516GR70	Supp. Ring Flange	+30	+30 <sup>B</sup>	+10	39	34	56	43					38	44	33	38.3				
673-104	E661-002-03	M2638-1	SA516GR70	Supp. Ring Segment	-50	-30 <sup>C</sup>	+10	77	85	81	81					65	65	60	63.3				
676-102	E661-002-03	C3529-1	SA516GR70	Manway Cover	-50	-30 <sup>C</sup>	+10	51	65	51	55.6					73	68	64	68.3				
622-102	E661-002-03	M2605-1	SA533BCLI	Upper Shell Plate	+10	+30 <sup>B</sup>	+10	55	43	44	47.3					36	27	26	29.9				
622-102	E661-002-03	M2605-2	SA533BCLI	Upper Shell Plate	-50	-30 <sup>B</sup>	+10	58	59	59	58.7					41	40	39	40				
642-102	E661-002-03	M2606-1	SA533BCLI	Lower Shell Plate	+30	+30 <sup>B</sup>	+10	40	43	48	43.7					26	27	32	28.3				

A-MTEB Position 5.2, "Fracture Toughness Requirements" Paragraph 1.1 (4), Minimum of 3 tests at a single temperature >45 ft-lbs.

B-MTEB Position 5.2, "Fracture Toughness Requirements" Paragraph 1.1 (4), Minimum of 3 tests at a single temperature >30 <45 ft lbs.

C-MTEB Position 5.2, "Fracture Toughness Requirements" Paragraph 1.1 (3)b

WSES-FSAR-UNIT-3

TABLE 5.2-9

Revision 307 (07/13)

WATERFORD UNIT 3 STEAM GENERATOR MATERIALS FRACTURE TOUGHNESS DATA

➔(EC-8458, R307)

Material Spec.	Material ID	Material Type	Heat Number	NDT (F)	NDT (C)	TCv50 ft lbs(F)	TCv50 ft lbs(C)	TCv35 mils (F)	TCv35 mils (C)	RT <sub>NDT</sub> (F)	RT <sub>NDT</sub> (C)	YS (ksi)	YS (MPa)	TS (ksi)	TS (MPa)	Elongation (%)	Reduction In Area (%)
SA-508 Grade 2 Class 2	F1	Forging	4584	0	-18	145	63	160	71	100	38	88.3	608.8	105.5	727.4	21.7	62.4
SA-508 Grade 2 Class 2	F2	Forging	5387	20	-7	115	46	90	32	55	13	83.2	573.7	99.3	684.7	20.5	57.0
SA-508 Grade 2 Class 2	F3	Forging	5389	50	10	130	54	105	41	70	21	89.1	614.3	105.9	730.2	20.4	59.2
SA-508 Grade 2 Class 2	FHAZ1	HAZ - SAW	4585/4109	-30	-34	15	-9	20	-7	-30	-34	93.0	641.2	111.6	769.5	25.7	49.1
SA-508 Grade 2 Class 2	FHAZ2	HAZ-SMAW	4585/3993, 4004 & 4009	-10	-23	50	10	40	4	-10	-23	93.9	647.4	105.2	725.4	21.8	38.9
SA-508 Grade 2 Class 2	FHAZ3	HAZ - SAW	5387/4109	-20	-29	15	-9	35	2	-20	-29	97.2	670.2	112.2	773.6	33.6	47.5
SA-508 Grade 2 Class 2	FHAZ4	HAZ - SAW	5389/4109	-10	-23	-5	-21	20	-7	-10	-23	101.9	702.6	117.8	812.2	35.0	55.0
SA-533 Type A, Class 2	P1	Plate	2864	30	-1	20	-7	20	-7	30	-1	82.3	567.5	102	703.3	22.8	---
SA-533 Type A, Class 2	P2	Plate	2899	-20	-29	-25	-32	-30	-34	-20	-29	79.6	548.8	101.4	699.2	26.1	---
SA-533 Type A, Class 2	P3	Plate	3272	10	-12	15	-9	15	-9	10	-12	74	510.2	94.5	651.6	25.6	---
SA-533 Type A, Class 2	P4	Plate	3312	-20	-29	-15	-26	-25	-32	-20	-29	69.4	478.5	89.5	617.1	26.4	---
SA-533 Type A, Class 2	SAW1	Weld - SAW	4336/4098	-80	-62	25	-4	15	-9	-35	-37	89.9	619.9	105.8	729.5	23.6	---
SA-533 Type A, Class 2	SAW2	Weld - SAW	4335/4098	-50	-46	40	4	40	4	-20	-29	90.9	626.8	106.5	734.3	56.2	---
SA-533 Type A, Class 2	SAW3	Weld - SAW	3742/3881	-60	-51	35	2	25	-4	-25	-32	81.6	562.6	98.9	681.9	56.8	---
SA-533 Type A, Class 2	PHAZ1	HAZ - SAW	4335/4098	-30	-34	0	-18	0	-18	-30	-34	97.4	671.6	114.2	787.4	48.8	---
SA-533 Type A, Class 2	PHAZ2	HAZ - SAW	4336/4113	-90	-68	-20	-29	-20	-29	-80	-62	93.9	647.4	113.8	784.7	49.4	---
SA-533 Type A, Class 2	PHAZ3	HAZ - SAW	3742/4113	-60	-51	0	-18	15	-9	-45	-43	84.4	581.9	100.5	692.9	56.4	---
SA-533 Type A, Class 2	GTAW1	Weld - GTAW	D4603/32J7	-50	-46	60	16	55	13	0	-18	103.9	716.4	120.6	831.5	48.2	59.8
SA-533 Type A, Class 2	GTAW2	Weld - GTAW	B1035/A070	-90	-68	-25	-32	-30	-34	-85	-65	109.2	752.9	123.2	849.5	58.8	68.2
SA-533 Type A, Class 2	GTAW3	Weld - GTAW	B1035/B481	-100	-73	-5	-21	-10	-23	-65	-54	101.4	699.2	116.5	803.3	55.7	68.4
SA-508 Grade 3, Class 2	F4	Forging	7341	-10	-23	165	74	67	19	105	41	81.1	559.2	99.7	687.4	22.4	58.6
SA-508 Grade 3, Class 2	F5	Forging	7431	80	27	175	79	92	33	115	46	92.5	637.8	111.1	766.0	18.3	53.8
SA-508 Grade 3, Class 2	FHAZ5	HAZ - SAW	7341 HAZ	-20	-29	80	27	85	29	25	-4	99.3	684.7	119.1	821.2	38.8	52.9
SA-508 Grade 3, Class 2	FHAZ6	HAZ - SAW	7431 HAZ	-60	-51	30	-1	40	4	-20	-29	101.3	698.5	121.3	836.4	39.9	58.0
SA-508 Grade 3, Class 2	F6	Forging	---	---	---	---	-37 to -43	---	-49 to -52	---	-25	---	510 to 570	---	650 to 700	---	---

⬅(EC-8458, R307)

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TABLE 5.2-10 (Sheet 1 of 2)

Revision 15 (03/07)

REACTOR COOLANT LEAK DETECTION SENSITIVITY

<u>Leakage Source</u>	<u>Detection Instrumentation</u>	<u>Instrument Range</u>	<u>Normal Reading</u>	<u>Average Rate of Change for 1.0 gpm leak</u>	<u>Time for Scale to Move 10% from Normal Reading for 1.0 gpm leak</u>
1. Direct →(DRN 04-1221, R13-A)  ←(DRN 04-1221, R13-A)	Sump input flow measurement system	0-20 gpm***** (Nonlinear)	0	1.55 min first 25 percent change	Approximately 1 min.
	Sump level flow measurement system	gpm	NA	10 min	NA
	Containment Radiation	10-10 <sup>6</sup> cpm	35,000 cpm*	94 cpm	1.0 hrs. approximately
2. Safety Valves	Valve Position Monitors	0-100%	Flow-No Flow	NA	NA
	Discharge Line Temperature	0-300 °F	Operating temp 120 °F	NA	NA
	Quench Tank Water Level	0-100%	NA	NA	NA
→(DRN 06-885, R15) 3. S.I. Tank Check Valves  ←(DRN 06-885, R15)	S.I. Tank Water Level	0-100%	NA	NA	NA
	S.I. Tank Pressure	350-750 psig	600 psig	NA	NA
4. Heat Exchangers	CCW Radiation	10-10 <sup>6</sup> cpm	200,000 cpm**	300 cpm	2.0 hrs. approximately
	CCW Surge Tank Water Level	0-100%	72.80%	1 gpm leak takes 3-1/2 hrs for level to rise from normal to high 72.80% to high 98.70%	1 1/2 hrs.
5. Steam Generator Tubing	Blowdown Line Radiation	10-10 <sup>6</sup> cpm	20,000 cpm***	300 cpm	2.0 hrs. approximately
	Condenser vacuum pumps exhaust Radiation	10-10 <sup>6</sup> cpm	70-100 cpm*****	NA	NA
6. Reactor Vessel Closure Head	O-Ring Space Pressure	0-3000 psig	0	NA	NA



WSES-FSAR-UNIT-3

TABLE 5.2-10 (Sheet 2 of 2)

REACTOR COOLANT LEAK DETECTION SENSITIVITY

<u>Leakage Source</u>	<u>Detection Instrumentation</u>	<u>Instrument Range</u>	<u>Normal Reading</u>	<u>Average Rate of Change for 1.0 gpm leak</u>	<u>Time for Scale to Move 10% from Normal Reading for 1.0 gpm leak</u>
7. Reactor Coolant Pump Closure Cover	Flange Gasket leak of pressure	0-3000 psig	0	NA	NA

\* Based on 0.1 gpm and 0.1% failed fuel, noble gas monitor faster sensibility can be achieved with the particulate monitor for lower leakages or lower percentages of failed fuel. However particulate monitor will be surveyed at this level. Step advance on filter paper will allow determination of leakage increase by taking readings of identical duration.

\*\* Assumes leak from one of the potentially very radioactive components (i.e., 0.1% failed fuel content). If leak is from lower activity component, leakage change can be detected more rapidly.

\*\*\* Variable as a function of blowdown rate, assumed rate < 8500 lbm/hr.

\*\*\*\* Based on Xe 133 which represents bulk of activity released.

\*\*\*\*\* Linearization performed within plant analog control system (PAC) by a function generator before signal is used as input to computer and flow recorder.

ISI FOR VALVES WHICH FORM THE PRESSURE  
BOUNDARY OF THE RCS

VALVE NUMBER	CLASS	VALVE CATEGORIES, PER ASME CODE SECTION XI, IWB				SIZE (INCHES)	VALVE TYPE (1)	ACTUATOR TYPE (2)	FAILURE POSITION (3)	TEST REQUIREMENTS (4)	CLARIFICATION (5)	TESTING_ ALTERNATIVES	LEAK RATE TEST VALUE (GPM)	POSITION (6)
		A	B	C	D									
→(EC-935, R302) 1501 B (SI-665) SI-405B	1	X				14	GA	PP	FC	CS, LT	3	-	5	LC
←(EC-935, R302) →(DRN 06-897, R15) →(EC-14765, R305) SI-4052B	1	X				3/4	GL	S	FC	CS, LT	3	-	5	LC
←(EC-14765, R305) 1502 B (SI-666) SI-401B	1	X				14	GA	M	-	CS, LT	3	-	5	LC
→(EC-935, R302) 1503 A (SI-651) SI-405A	1	X				14	GA	PP	-	CS, LT	3	-	5	LC
←(DRN 06-897, R15; EC-935, R302) →(EC-14765, R305) SI-4052A	1	X				3/4	GL	S	FC	CS, LT	3	-	5	LC
←(EC-14765, R305) 1504 A (SI-652) SI-401A	1	X				14	GA	M	-	CS, LT	3	-	5	LC
1509 RL1A (SI-217) SI-335A	1	X	X			12	CH	-	-	CS, LT	1	-	1	C
1510 TK1A (SI-215) SI-329A	1	X	X			12	CH	-	-	CS, LT	1	-	1	C
1511 RL1B (SI-227) SI-335B	1	X	X			12	CH	-	-	CS, LT	1	-	1	C

ISI FOR VALVES WHICH FORM THE PRESSURE  
BOUNDARY OF THE RCS

VALVE NUMBER	CLASS	VALVE CATEGORIES PER ASME CODE SECTION XI, IWB				SIZE (INCHES)	VALVE TYPE (1)	ACTUATOR TYPE (2)	FAILURE POSITION (3)	TEST REQUIREMENTS (4)	CLARIFICATION (5)	TESTING ALTERNATIVES	LEAK RATE TEST VALUE (GPM)	POSITION (6)
		A	B	C	D									
1512 TK1B (SI-225) SI-329B	1	X	X			12	CH	-	-	CS, LT	1	-	1	C
1513 RL2A (SI-237) SI-336a	1	X	X			12	CH	-	-	CS, LT	1	-	1	C
1514 TK2A (SI-235) SI-330A	1	X	X			12	CH	-	-	CS, LT	1	-	1	C
1515 RL2B (SI-247) SI-336B	1	X	X			12	CH	-	-	CS, LT	1	-	1	C
→(DRN 06-897, R15) 1516 TK2B (SI-245) SI-330B	1	X	X			12	CH	-	-	CS, LT	2	-	1	C
←(DRN 06-897, R15) 1517 RL1A (SI-114) SI-143B	1	X	X			8	CH	-	-	CS, LT	1	-	1	C
1518 RL1B (SI-124) SI-142B	1	X	X			8	CH	-	-	CS, LT	1	-	1	C

WSES-FSAR-UNIT-3

TABLE 5.2-11 (Sheet 3 of 5)

Revision 305 (11/11)

ISI FOR VALVES WHICH FORM THE PRESSURE  
BOUNDARY OF THE RCS

VALVE NUMBER	CLASS	VALVE CATEGORIES (PER ASME CODE SECTION XI, IWB				SIZE (INCHES)	VALVE TYPE (1)	ACTUATOR TYPE (2)	FAILURE POSITION (3)	TEST REQUIREMENTS (4)	CLARIFICATION (5)	TESTING ALTERNATIVES	LEAK RATE TEST VALUE (GPM)	NORMAL POSITION (6)
		A	B	C	D									
1519 RL2A (SI-134) SI-143A	1	X	X			8	CH	-	-	CS, LT	1	-	1	C
1520 RL2B (SI-144) SI-142A	1	X	X			8	CH	-	-	CS, LT	1	-	1	C
1522 RL1A (SI-113) SI-241	1	X	X			3	CH	-	-	CS, LT	1	-	1	C
1523 RL1B (SI-123) SI-242	1	X	X			3	CH	-	-	CS, LT	1	-	1	C
1524 RL2A (SI-133) SI-243	1	X	X			3	CH	-	-	CS, LT	1	-	1	C
1525 RL2B (SI-143) SI-244	1	X	X			3	CH	-	-	CS, LT	1	-	1	C
2506 SI-510A	1	X	X			3	CH	-	-	LT	1	-	1	C
2507 SI-512A	1	X	X			3	CH	-	-	LT	1	-	1	C
2508 SI-510B	1	X	X			3	CH	-	-	LT, CS	1	-	1	C
2509 SI-512B	1	X	X			3	CH	-	-	LT, CS	1	-	1	C

ISI FOR VALVES WHICH FORM THE PRESSURE  
BOUNDARY OF THE RCS

Notes:

1) Valve Type

GA - Gate  
CH - Check  
GL - Globe

2) Actuator Type

HP - Hydraulic, Pneumatic  
M - Motor  
DP - Diaphragm, Pneumatic

→(EC-935, R302)

PP – Piston, Pneumatic

←(EC-935, R302)

→(EC-14765, R305)

S – Solenoid

←(EC-14765, R305)

3) Failure Position

FC - Fail Closed

4) Test Requirements

Q - Exercise valve (full stroke) for operability every three months

LT - Valves are leak tested per Section XI, Subsection IWV

MT - Stroke time measurements are taken and compared to the stroke time function every three months

CV - Exercise check valves to the position required to fulfill their function every three months.

SRV - Safety and relief valves are tested per Section XI, Subsection IWV

DT - TEST Category D valves per Section XI, Subsection IWV

CS - Exercise valve for operability every cold shutdown

RR - Exercise valve for operability every reactor refueling

5) Clarification

→(DRN 06-897, R15)

1. Exercising and leak testing will be performed once per refuel outage, prior to unit start-up of reduced pressure.

←(DRN 06-897, R15)

ISI FOR VALVES WHICH FORM THE PRESSURE  
BOUNDARY OF THE RCS

→(DRN 06-897, R15)

2. Exercising and leak testing will be performed once per refuel outage prior to unit start-up. Leak rate will be measured by rise in water level in the Safety Injection Tank.
3. Exercising and leak testing will be performed once per refuel outage prior to unit start-up at full pressure.

←(DRN 06-897, R15)

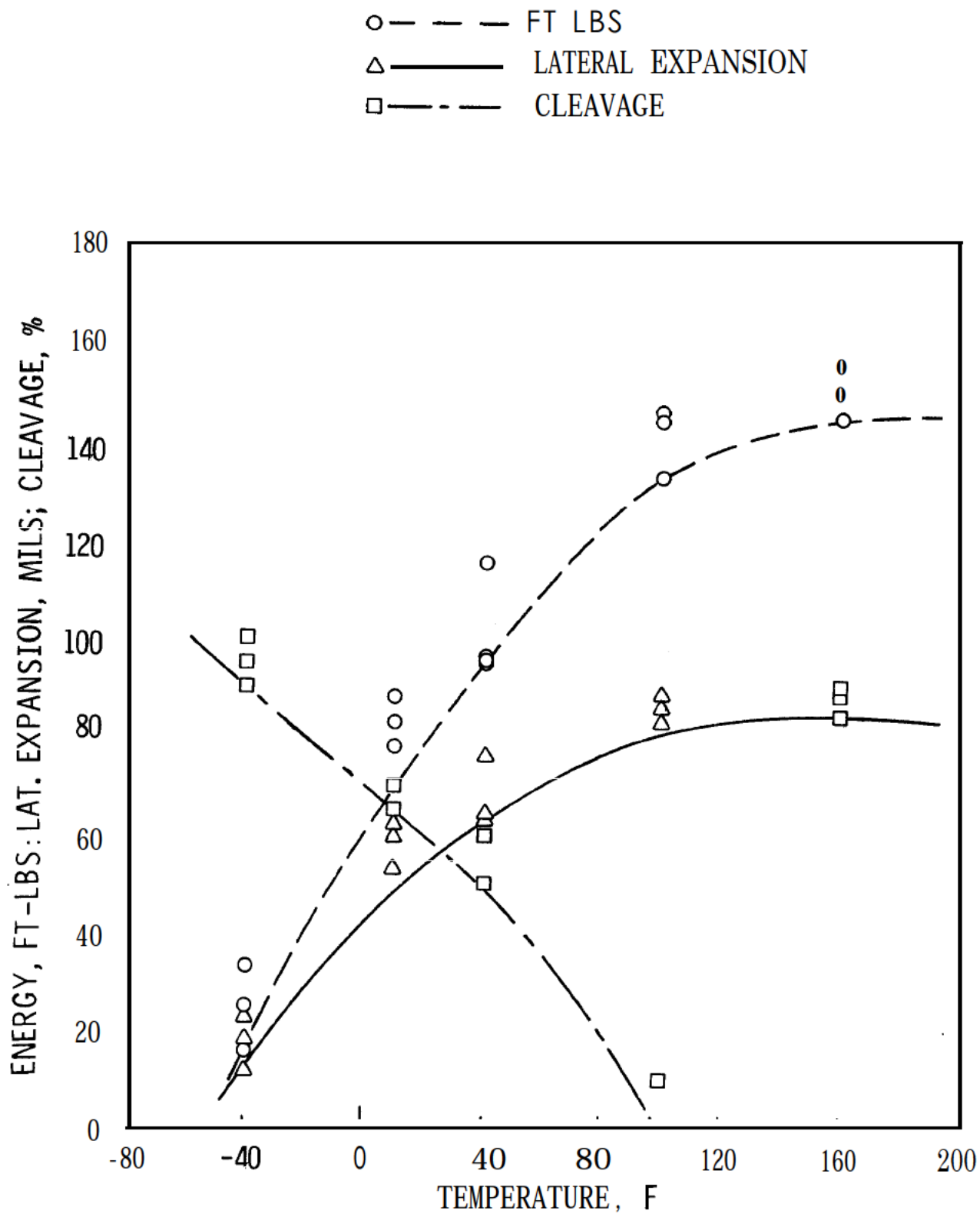
4. Leak testing performed at reduced pressure.

General: Provisions have been made for part stroke exercising of Class 1 check valves in the SIS and RHR Systems during normal operation. However, it is considered that probable system upset during part stroke exercising may do more damage than benefit of partial exercising. Thus, these valves will be full stroke exercised during refueling . This follows the requirements of ASME Section XI, Subsection IWV.

6) Normal Position

LC - Locked Closed  
C - Closed



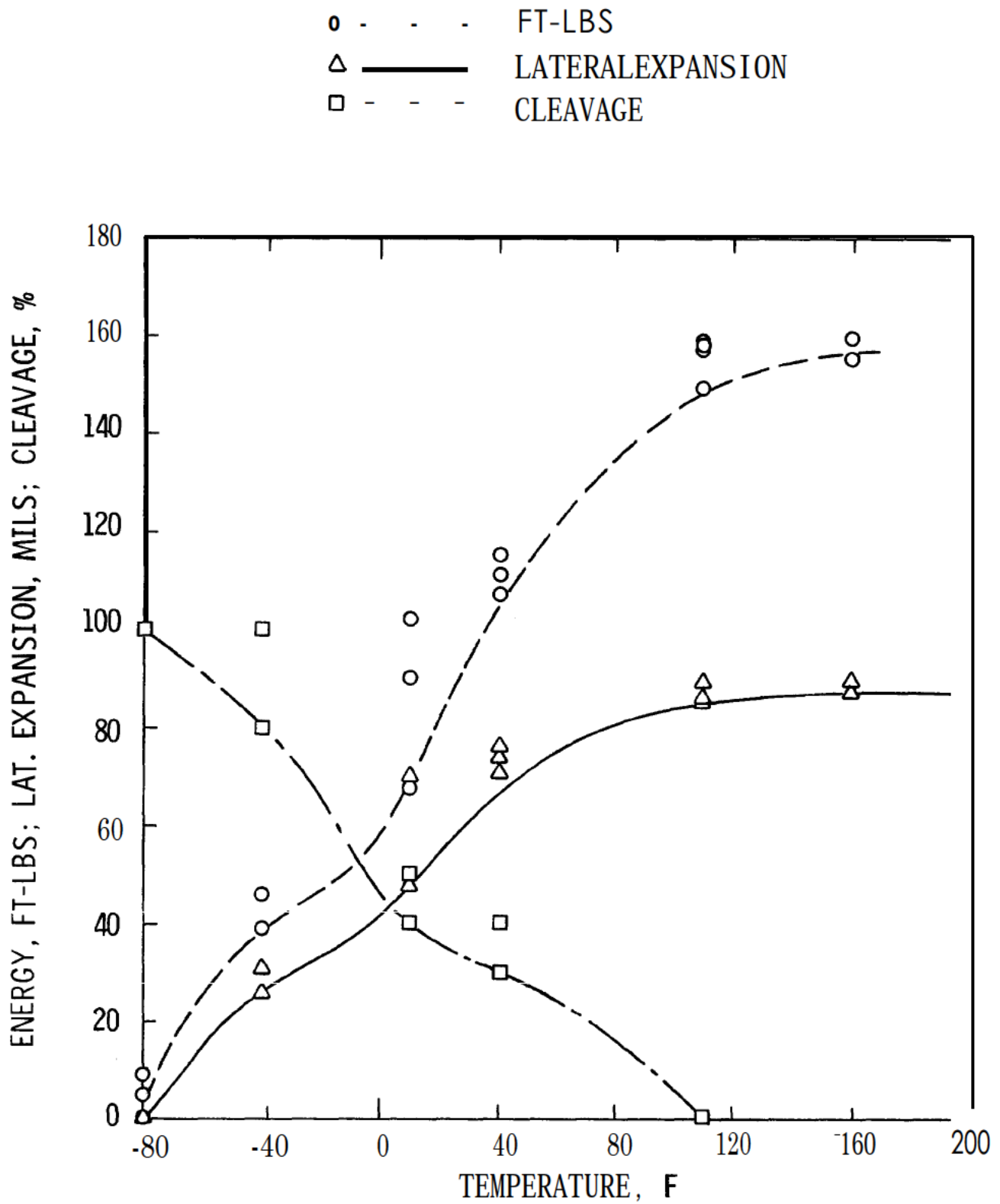


LOUISIANA  
POWER & LIGHT CO.  
Waterford Steam  
Electric Station

CHARPY TEST RESULTS  
INTERMEDIATE SHELL PLATE M-1003-1

Figure  
5.2-2



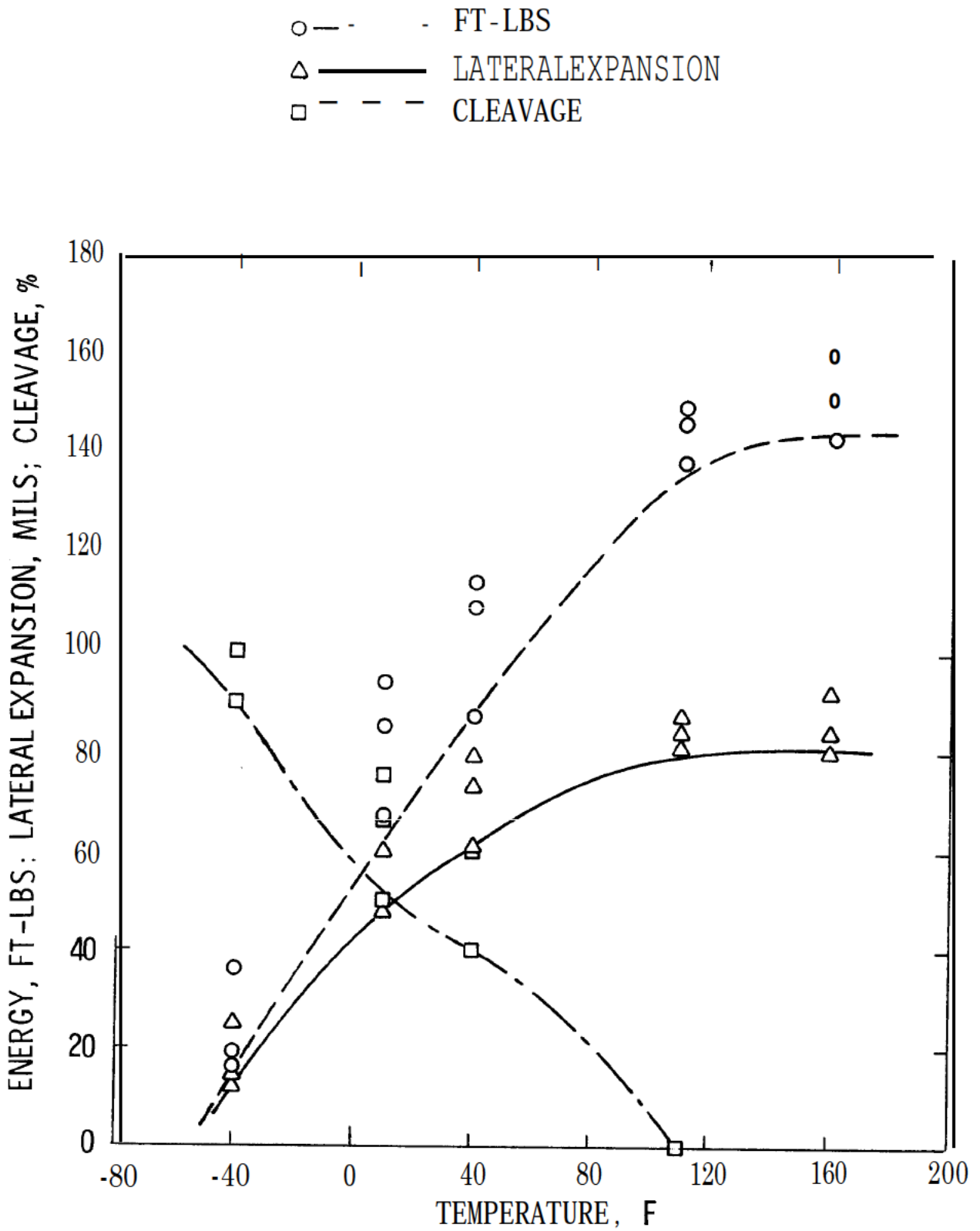


LOUISIANA  
POWER & LIGHT CO.  
Waterford Steam  
Electric Station

CHARPY TEST RESULTS  
INTERMEDIATE SHELL PLATE M-1003-2

Figure

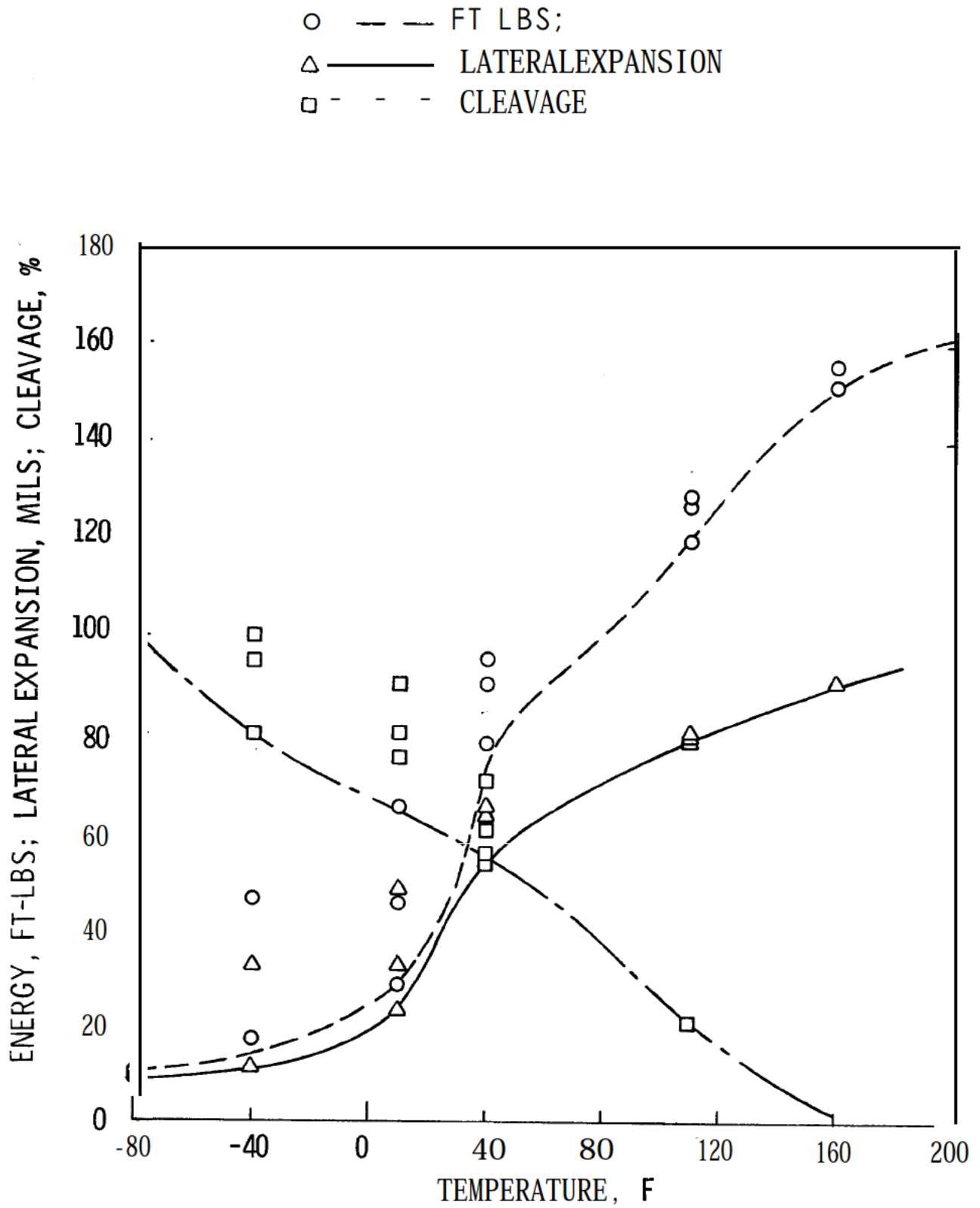
5.2-3



LOUISIANA  
POWER & LIGHT CO.  
Waterford Steam  
Electric Station

CHARPY TEST RESULTS  
INTERMEDIATE SHELL PLATE M-1003-3

Figure  
5.2-4



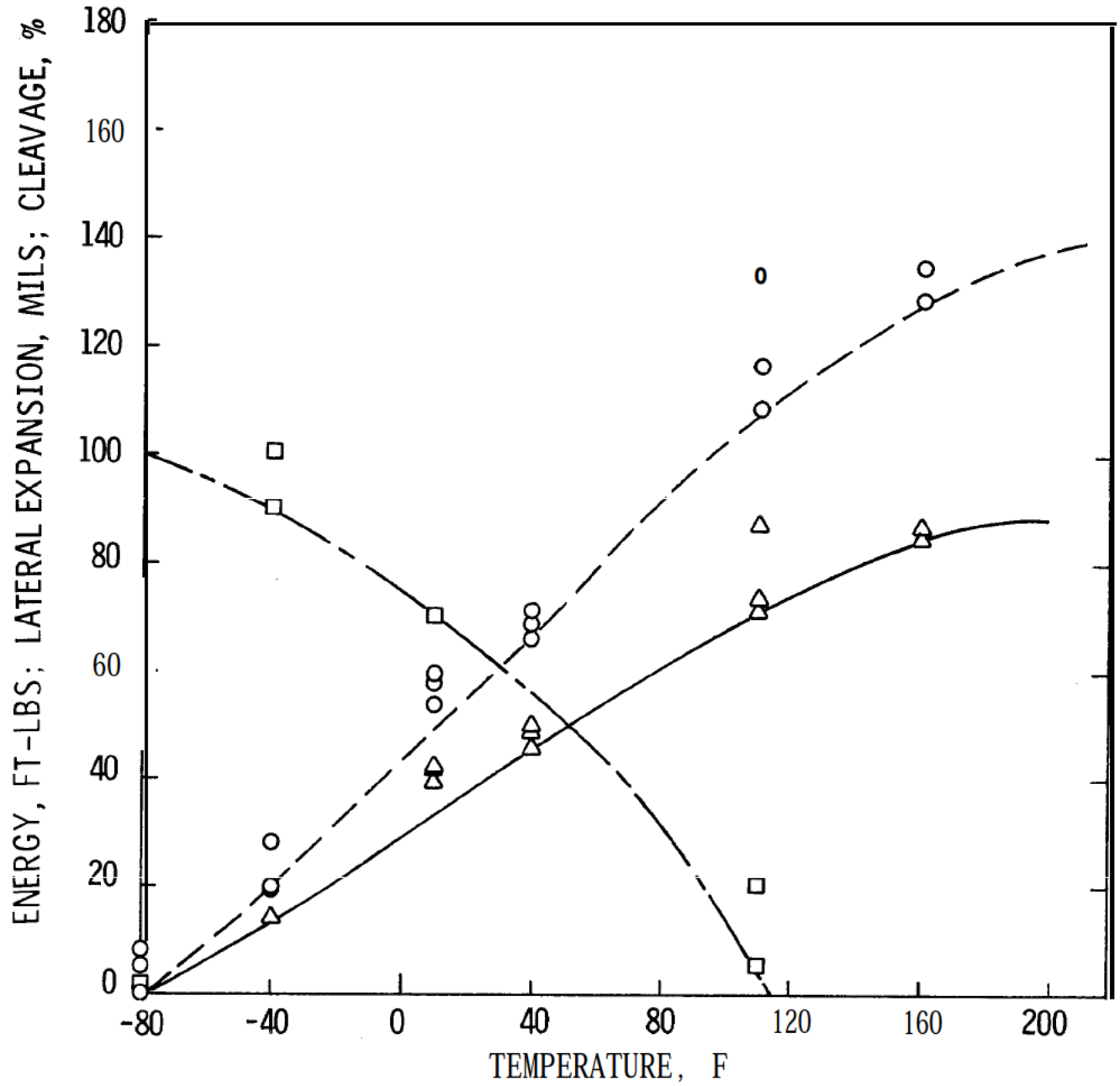
LOUISIANA  
POWER & LIGHT CO.  
Waterford Steam  
Electric Station

CHARPY TEST RESULTS  
UPPER SHELL PLATE M-1002-1

Figure

5.2-5

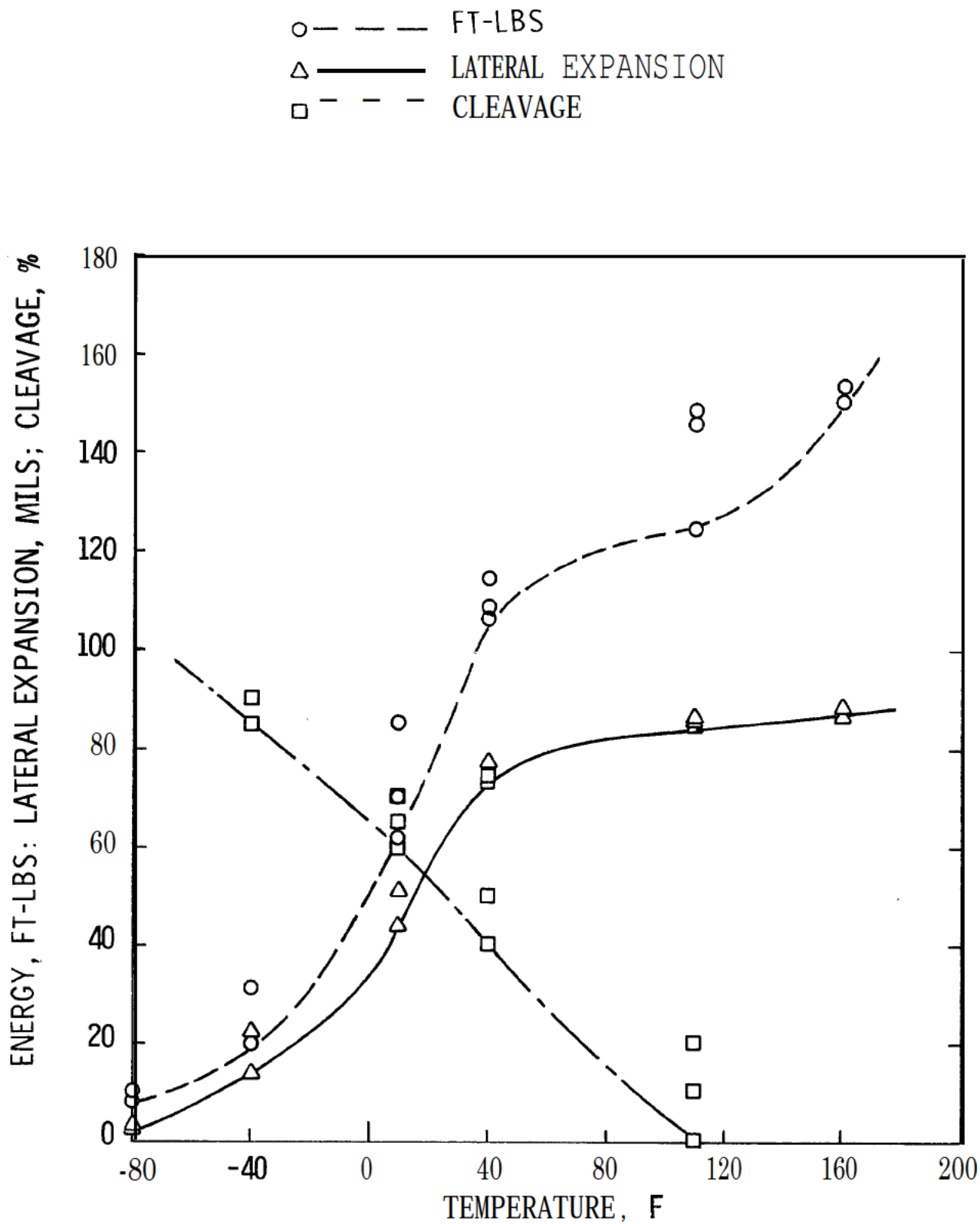
○ - - - FT-LBS  
 △ ——— LATERALEXPANSION  
 □ - - - CLEAVAGE



LOUISIANA  
 POWER & LIGHT CO.  
 Waterford Steam  
 Electric Station

CHARPY TEST RESULTS  
 UPPER SHELL PLATE M-1002-2

Figure  
 5.2-6

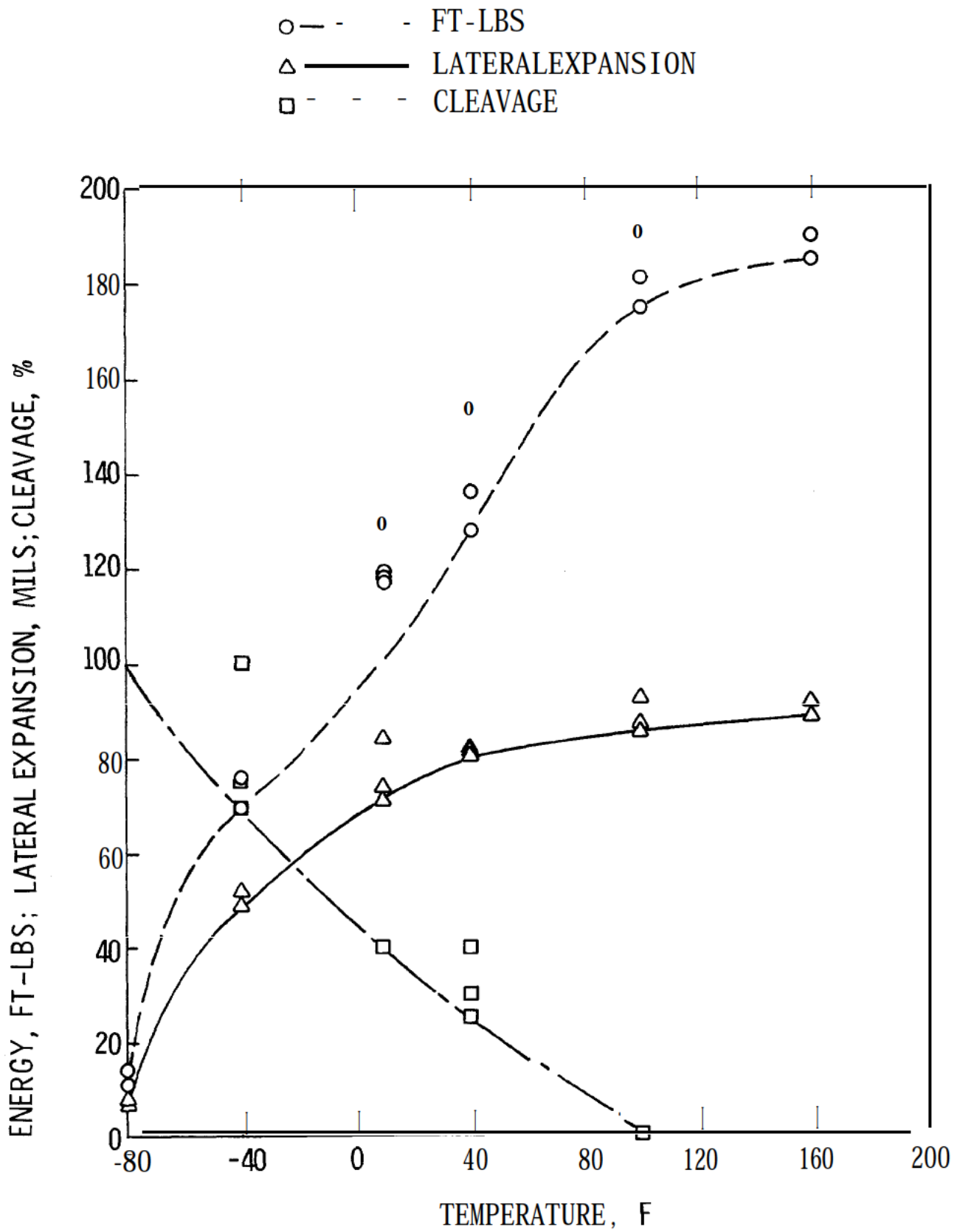


LOUISIANA  
POWER & LIGHT CO.  
Waterford Steam  
Electric Station

CHARPY TEST RESULTS  
UPPER SHELL PLATE M-1002-3

Figure

5.2-7

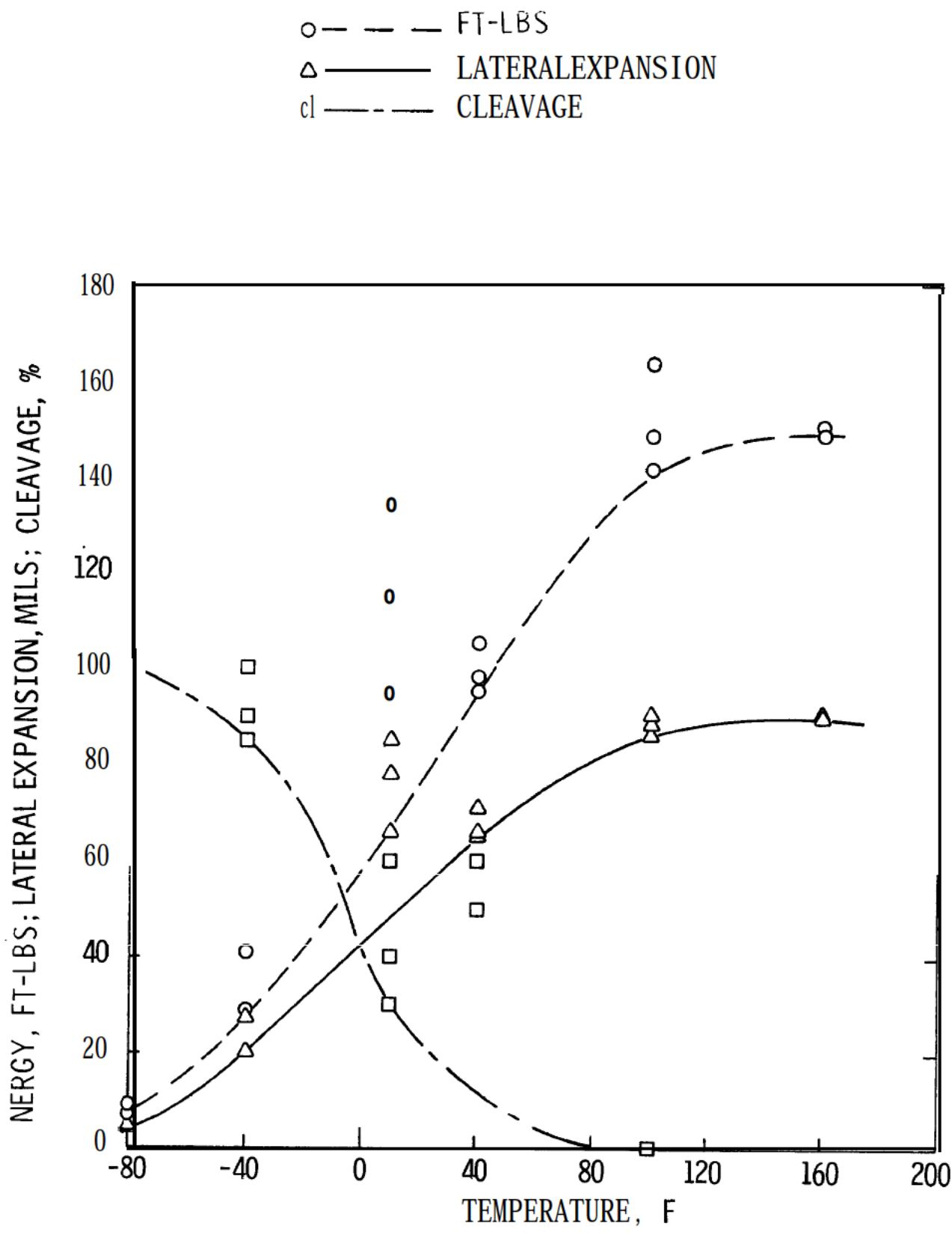


LOUISIANA  
POWER & LIGHT CO.  
Waterford Steam  
Electric Station

CHARPY TEST RESULTS  
BOTTOM HEAD TORUS M-1007-1

Figure

5.2-8



LOUISIANA  
POWER & LIGHT CO.  
Waterford Steam  
Electric Station

CHARPY TEST RESULTS  
BOTTOM HEAD DOME M-1008-1

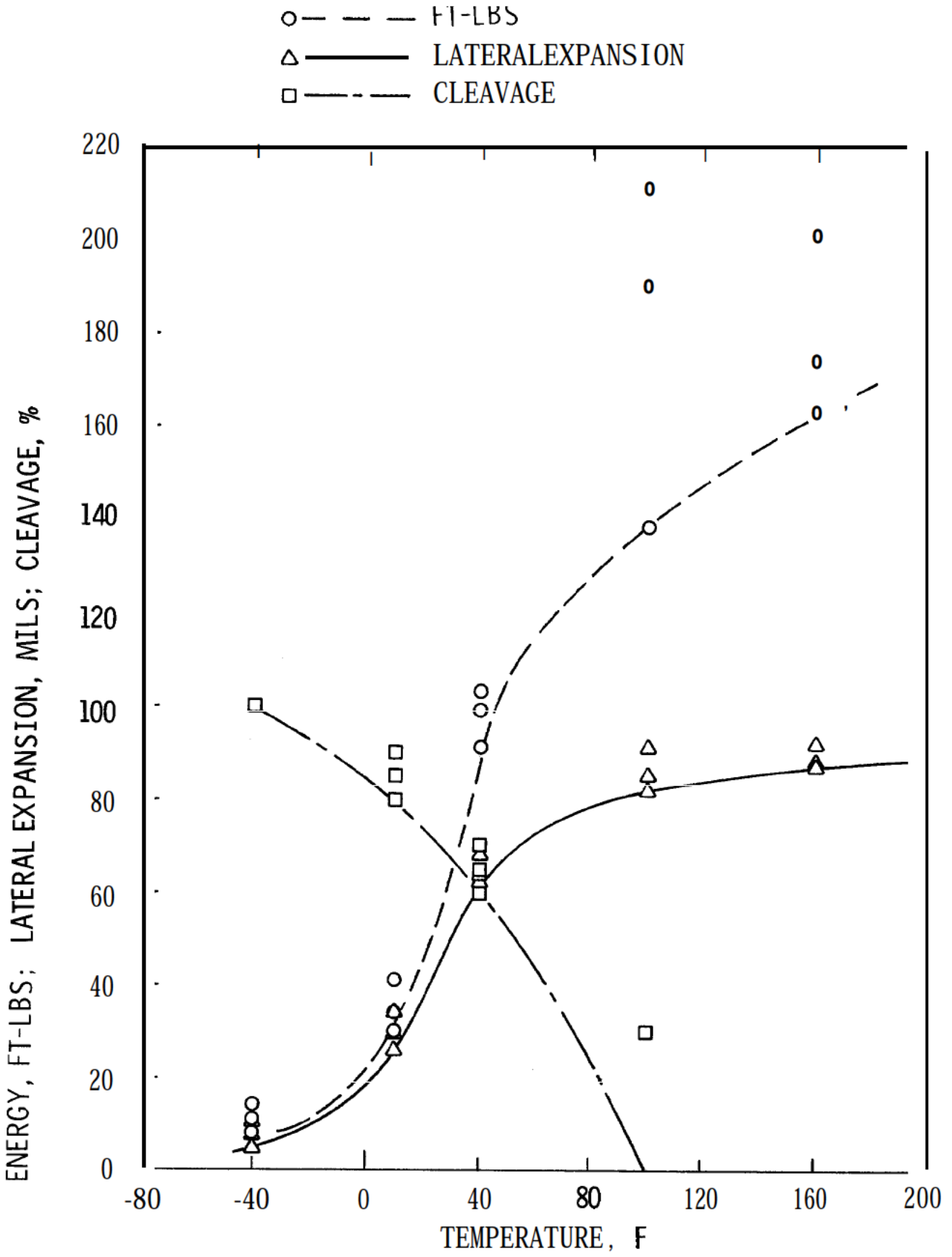
Figure  
5.2-9

→(EC-1020, R307)

FIGURE 5.2-10 HAS BEEN INTENTIONALLY DELETED.

←(EC-1020, R307)

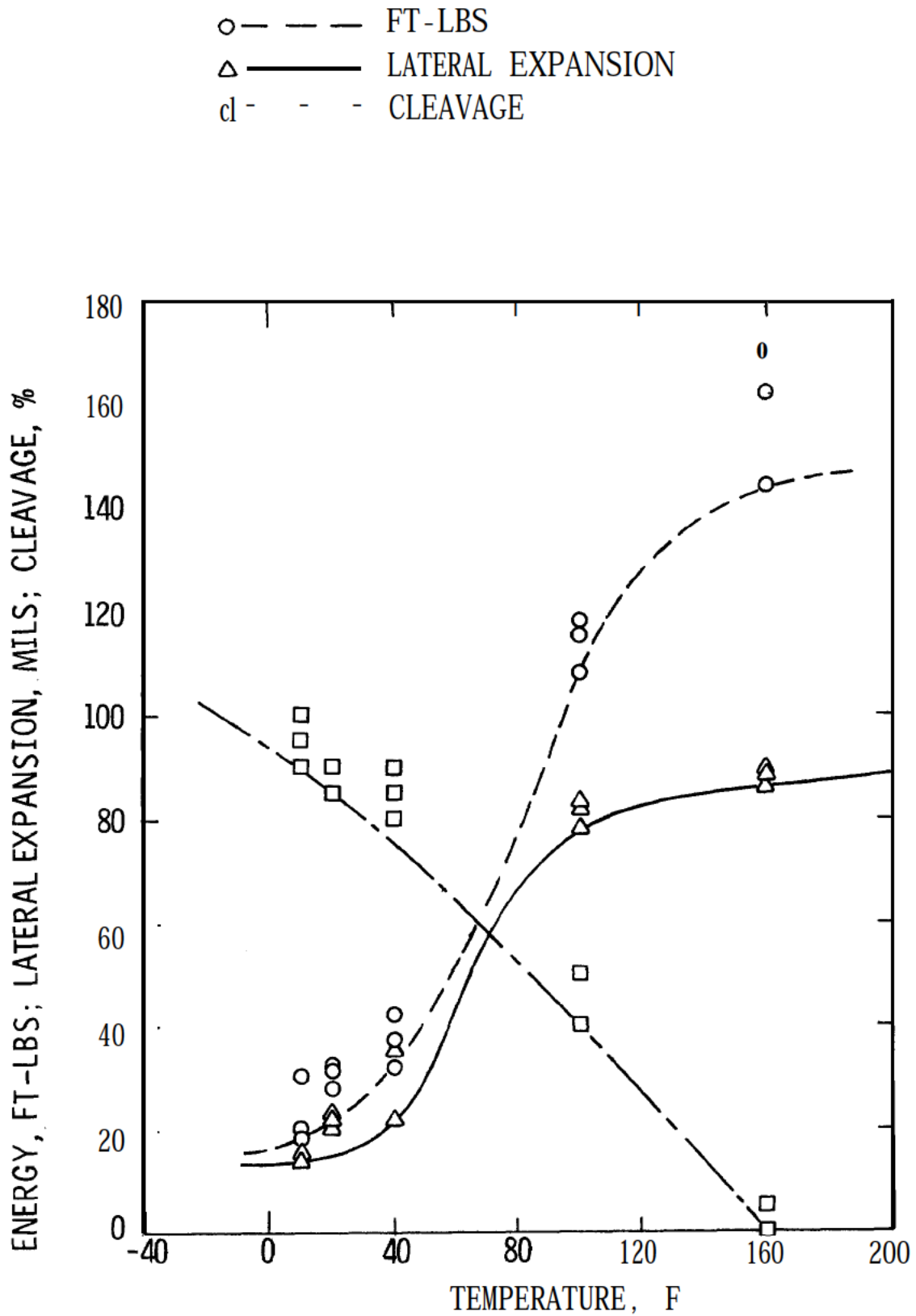




LOUISIANA  
POWER & LIGHT CO.  
Waterford Steam  
Electric Station

CHARPY TEST RESULTS  
LOWER SHELL PLATE M-1004-1

Figure  
5.2-1 1

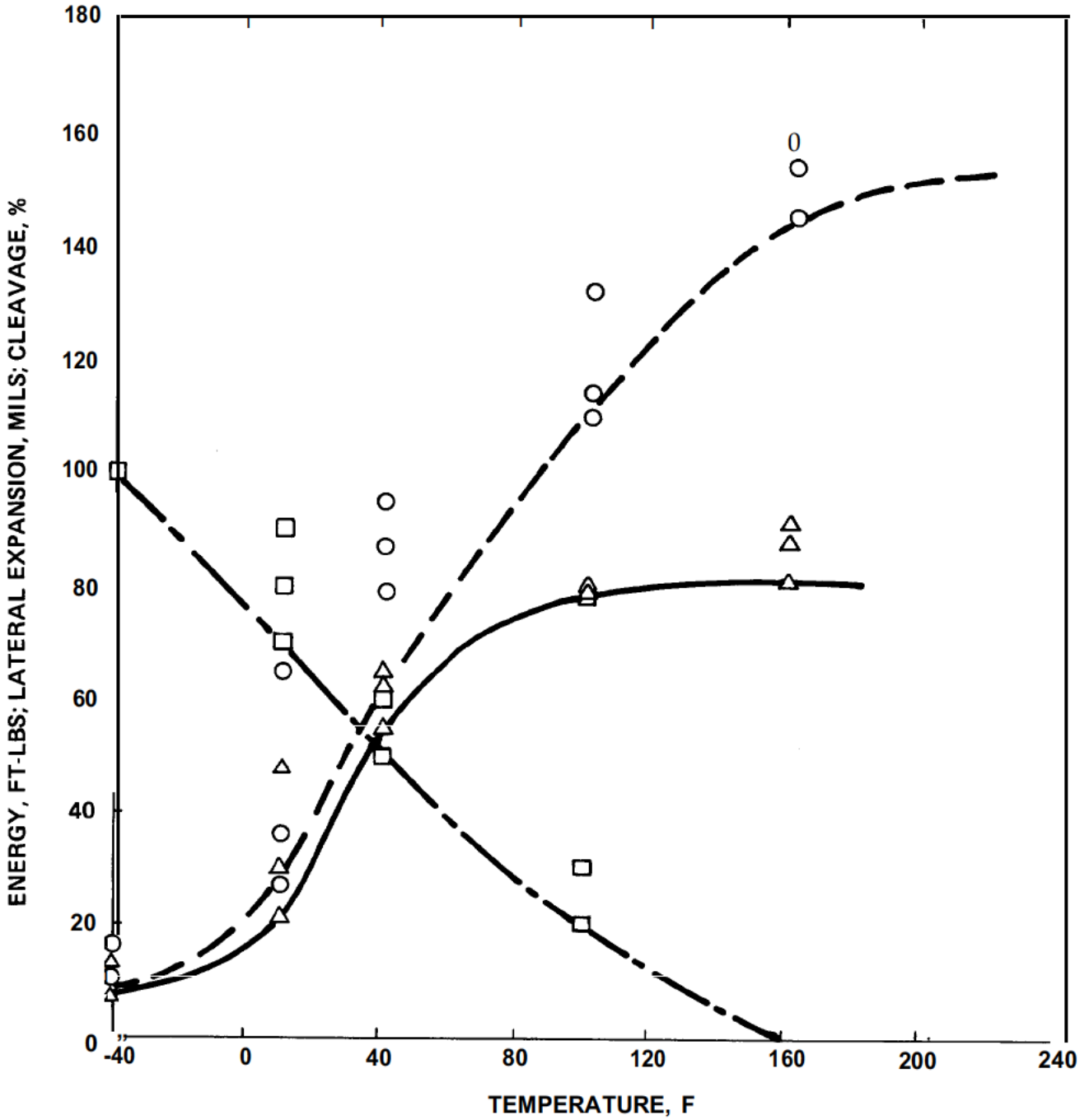


LOUISIANA  
POWER & LIGHT CO.  
Waterford Steam  
Electric Station

CHARPY TEST RESULTS  
LOWER SHELL PLATE M-1004-2

Figure  
5.2-1 2

○ — — — — FT-LBS  
 △ — — — — LATERAL EXPANSION  
 □ — — — — CLEAVAGE



LOUISIANA  
 POWER & LIGHT CO.  
 Waterford Steam  
 Electric Station

CHARPY TEST RESULTS  
 LOWER SHELL PLATE M-1004-3

Figure  
 5.2-13

→(EC-1020, R307)

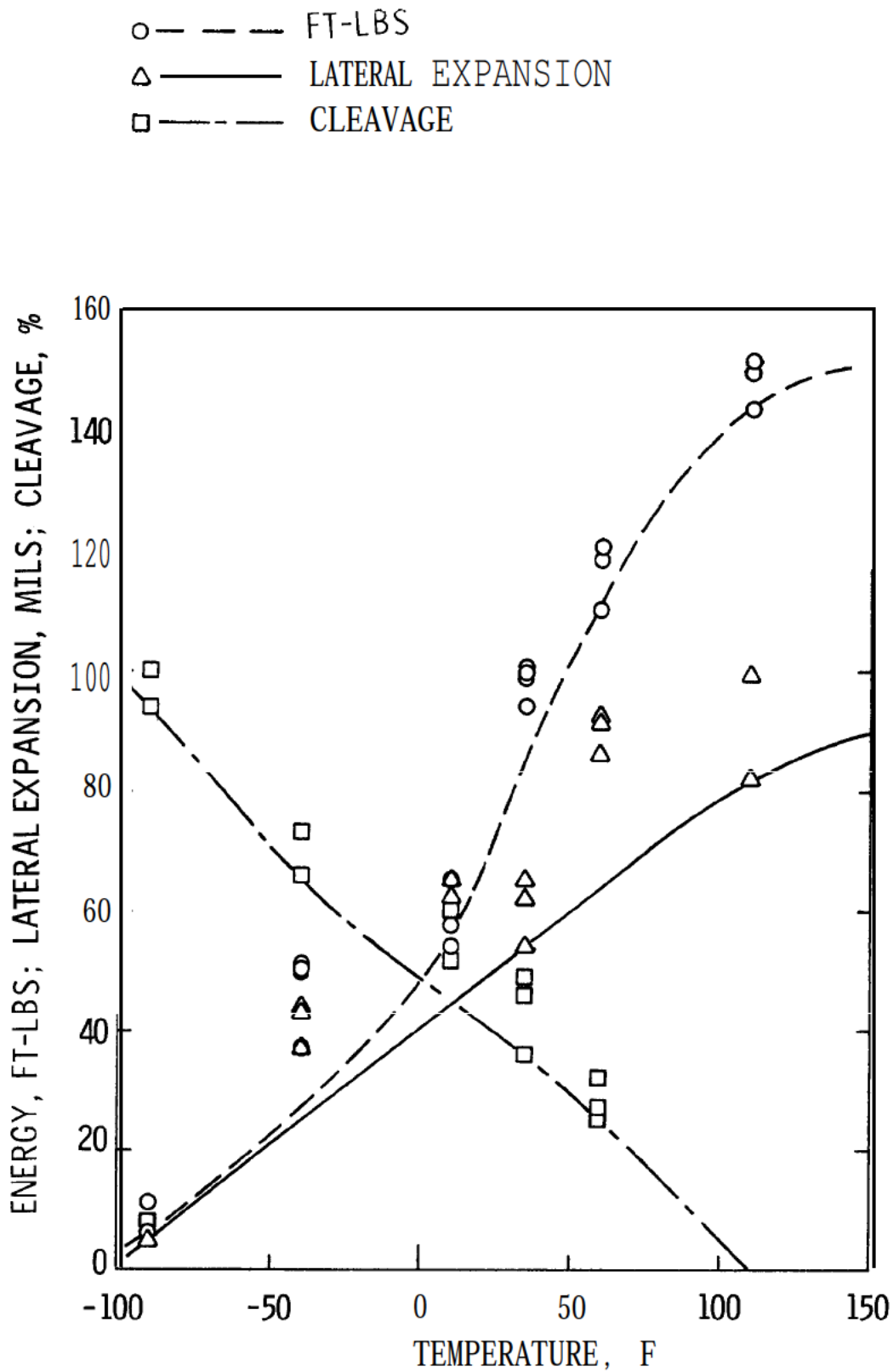
FIGURE 5.2-14 HAS BEEN INTENTIONALLY DELETED.

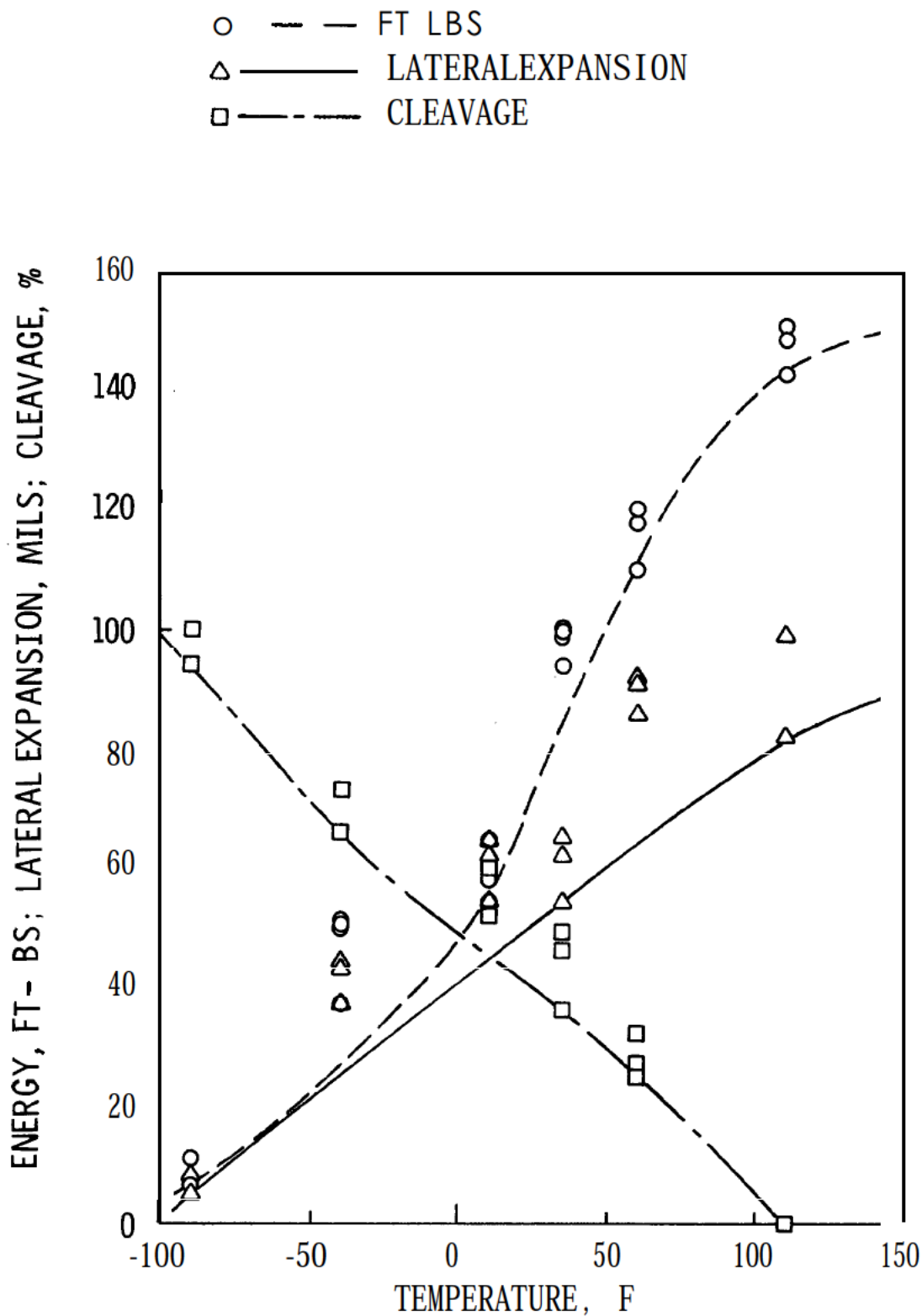
←(EC-1020, R307)

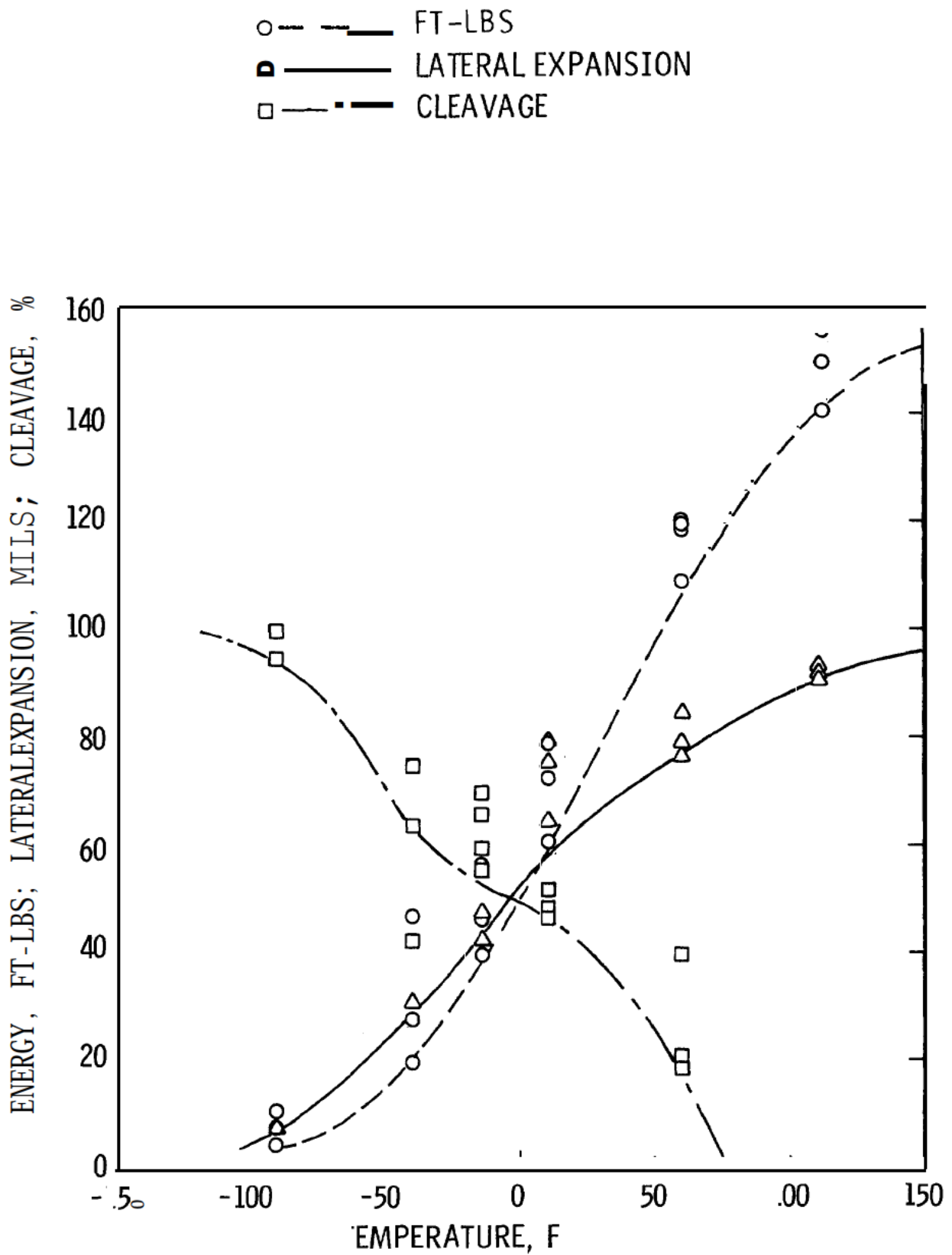
→(EC-1020, R307)

FIGURE 5.2-15 HAS BEEN INTENTIONALLY DELETED.

←(EC-1020, R307)







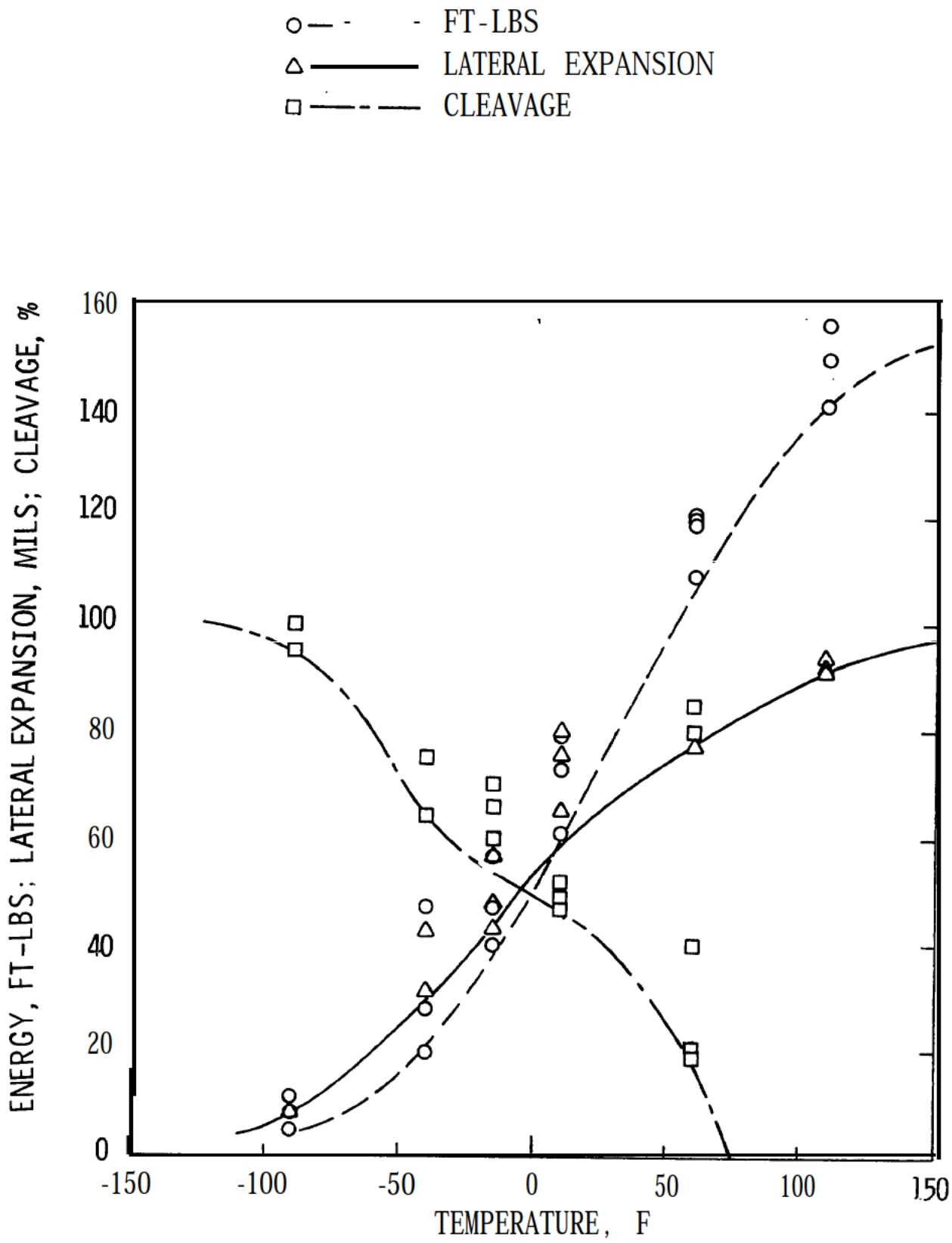
LOUISIANA  
POWER & LIGHT CO.  
Waterford Steam  
Electric Station

CHARPY TEST RESULTS  
INLET NOZZLE SAFE END M-1013-3

Figure

5.2-18

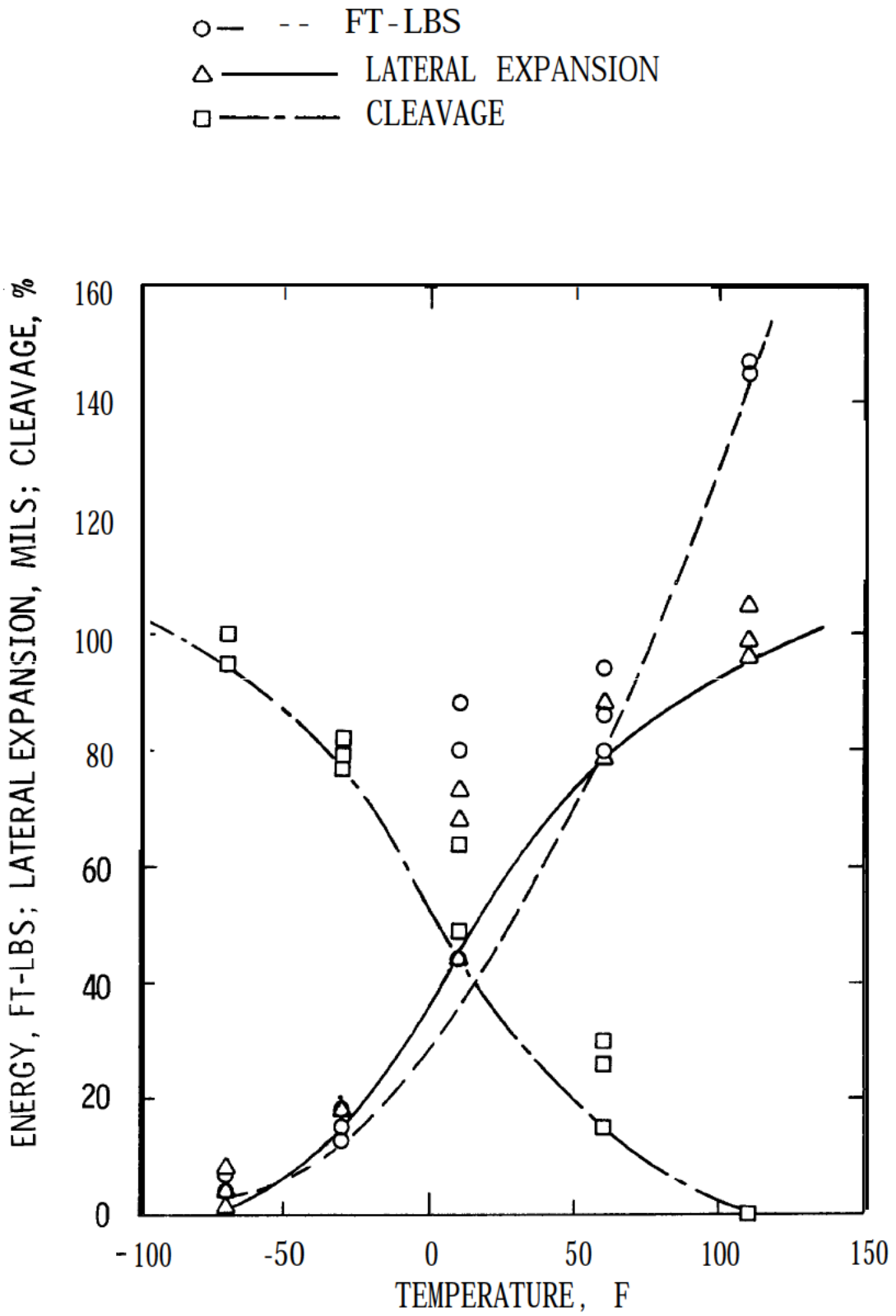




LOUISIANA  
POWER & LIGHT CO.  
Waterford Steam  
Electric Station

CHARPY TEST RESULTS  
INLET NOZZLE SAFE END M-1013-4

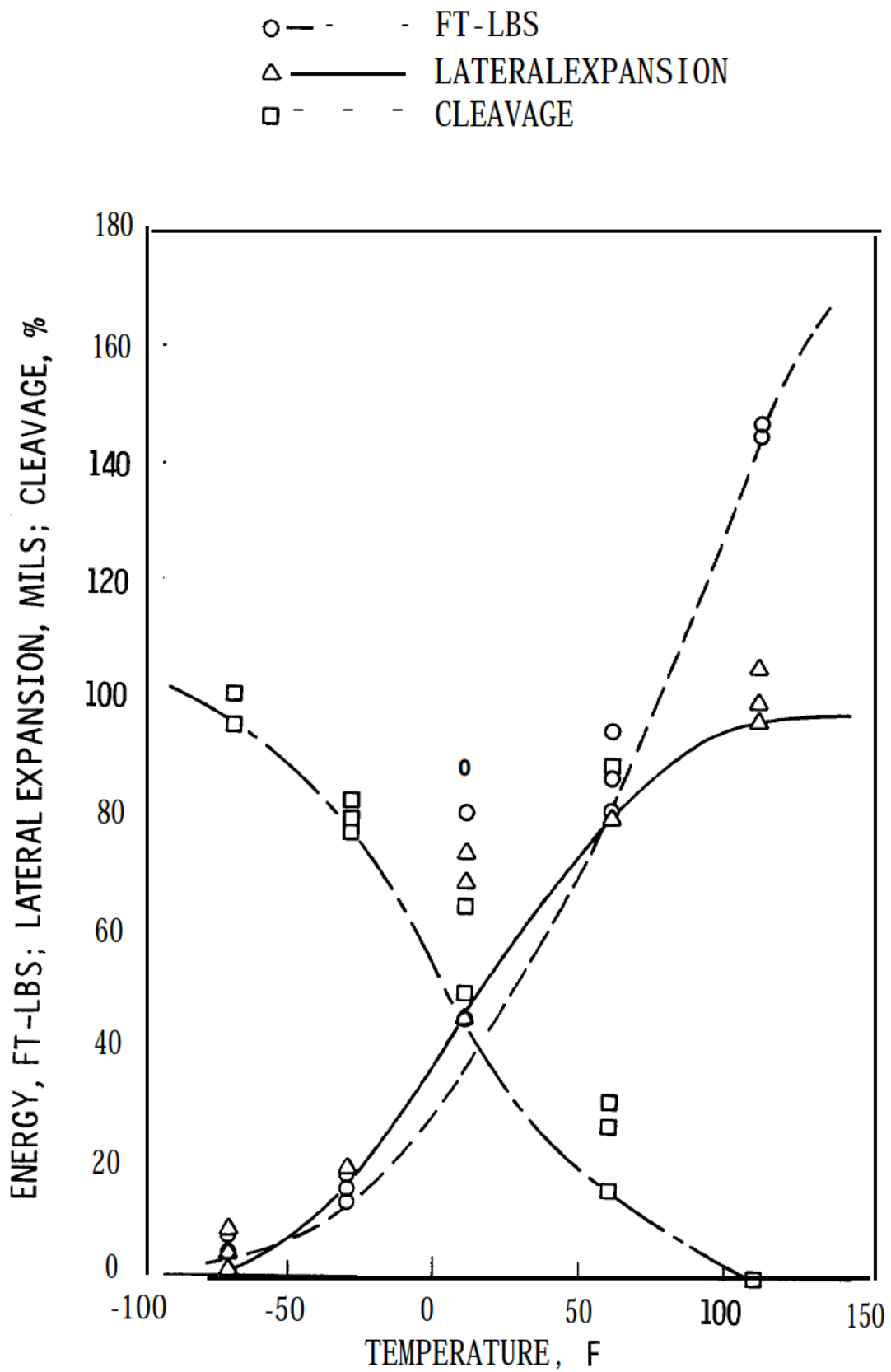
Figure  
5.2-19

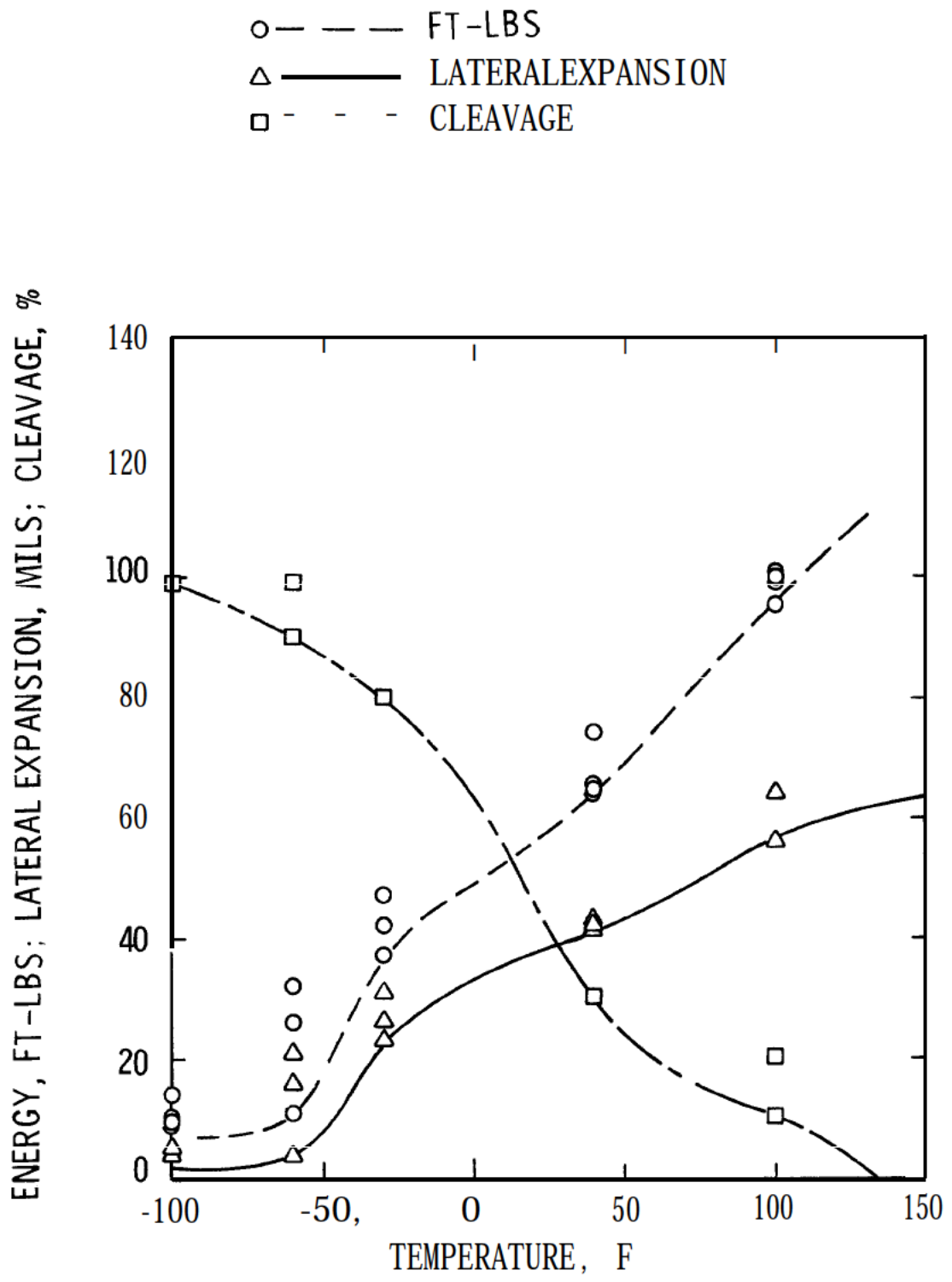


LOUISIANA  
POWER & LIGHT CO.  
Waterford Steam  
Electric Station

CHARPY TEST RESULTS  
OUTLET NOZZLE SAFE END M-1012-1

Figure  
5.2-20

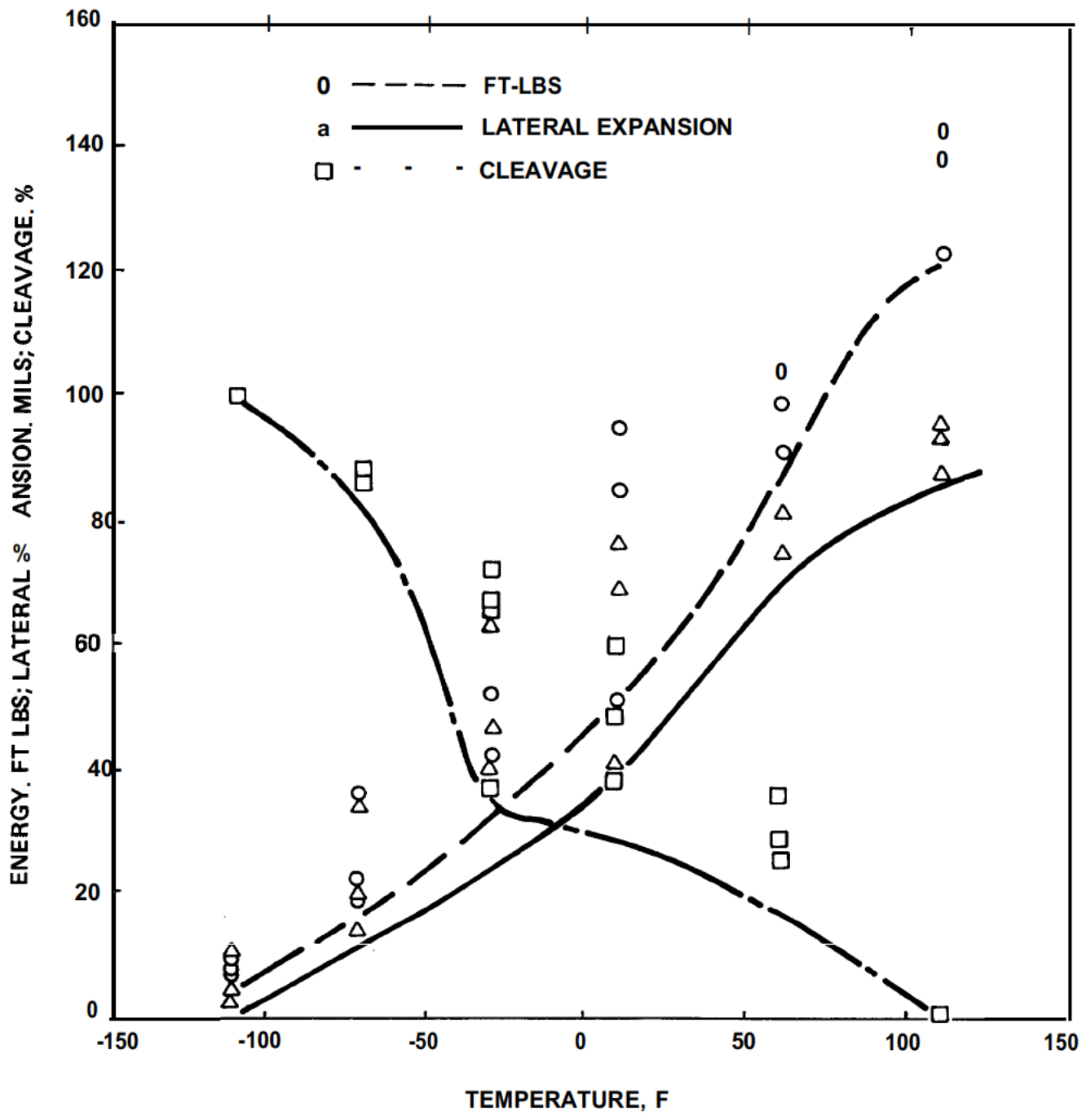




LOUISIANA  
POWER & LIGHT CO.  
Waterford Steam  
Electric Station

CHARPY TEST RESULTS  
OUTLET NOZZLE FORGING M-101 I-I

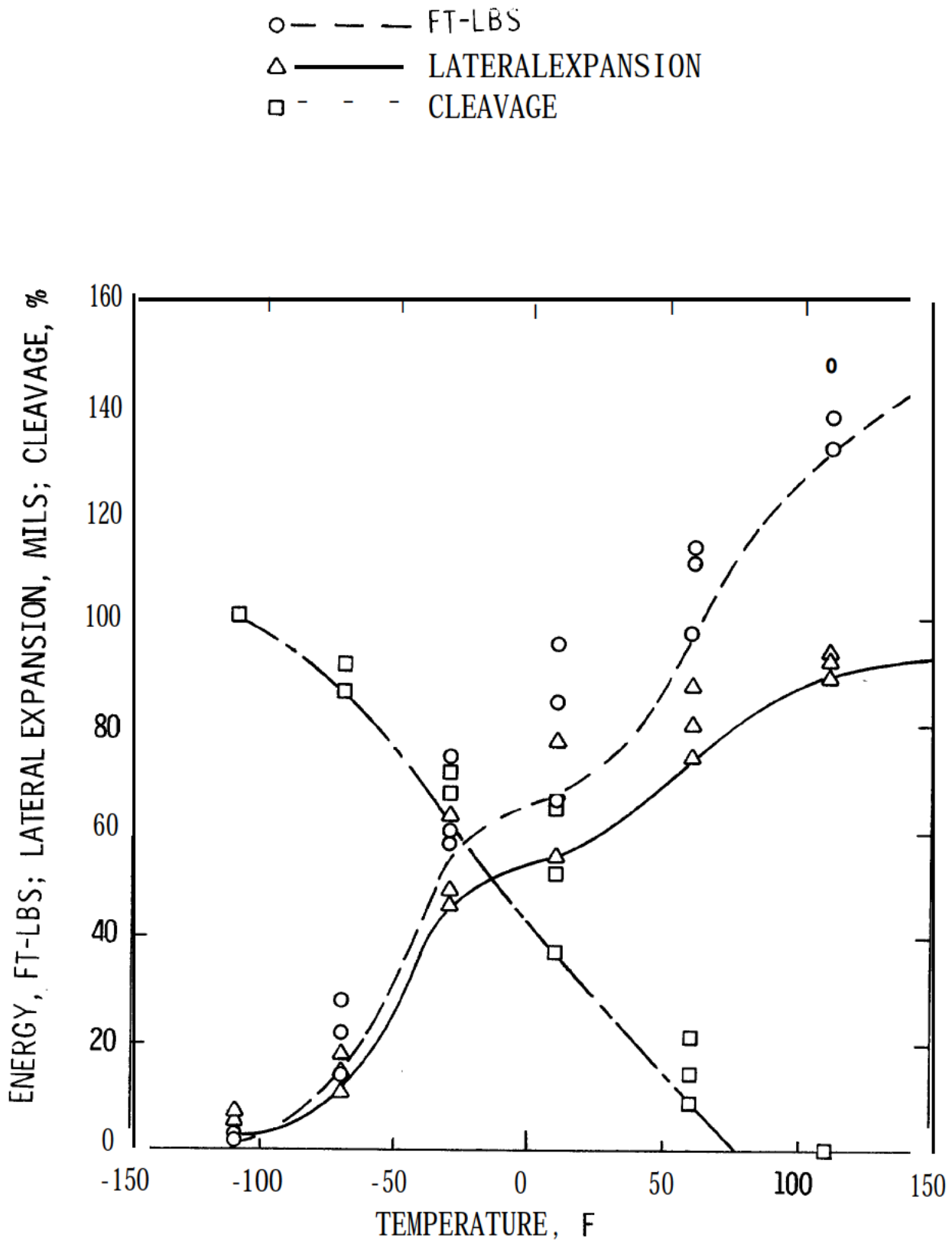
Figure  
5.2-22



LOUISIANA  
POWER & LIGHT CO.  
Waterford Steam  
Electric Station

CHARPY TEST RESULTS  
INLET NOZZLE FORGING M-1010-1

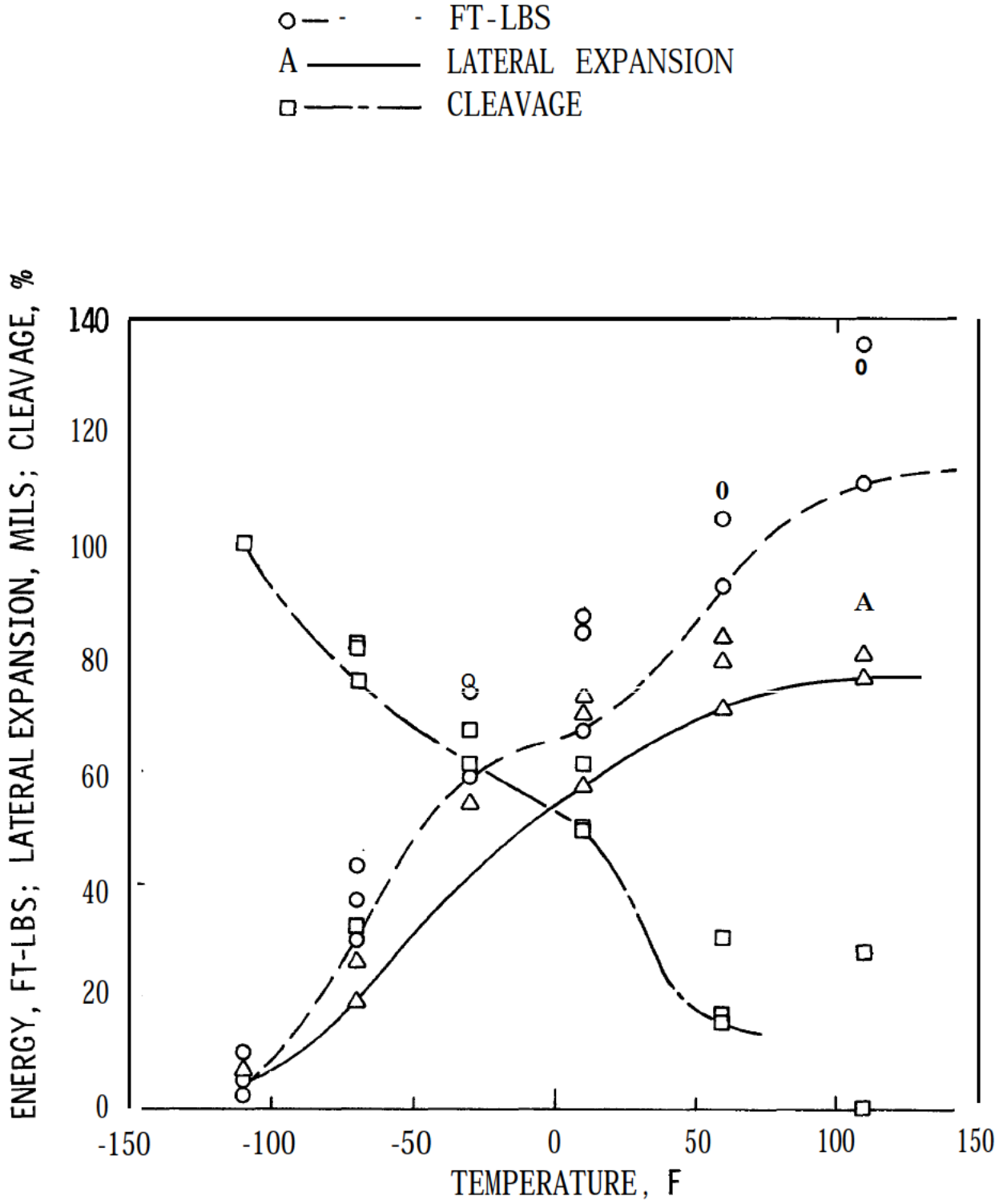
Figure  
5.2-23



LOUISIANA  
POWER & LIGHT CO.  
Waterford Steam  
Electric Station

CHARPY TEST RESULTS  
INLET NOZZLE, ORGING M-1010-2

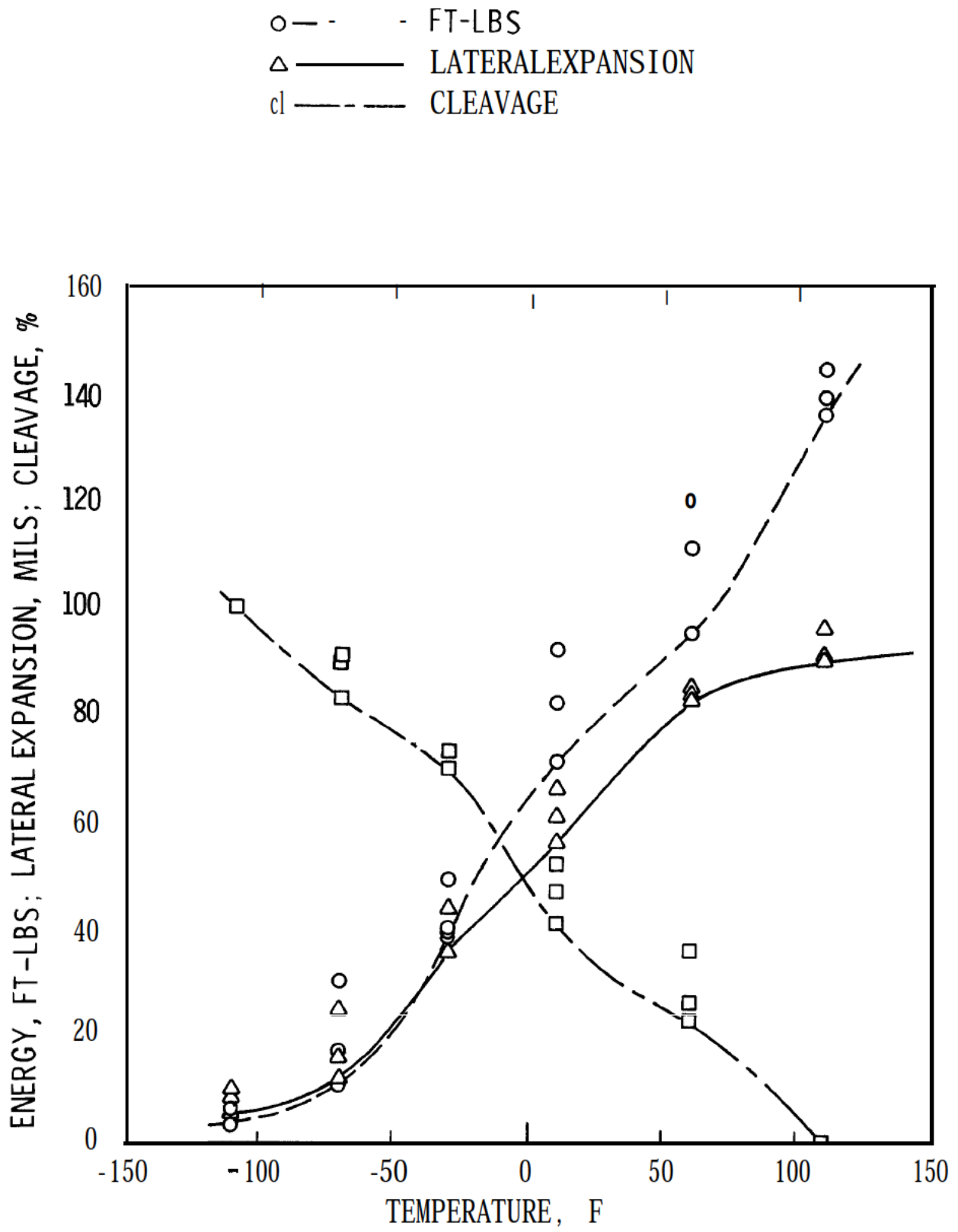
Figure  
5.2-24



LOUISIANA  
POWER & LIGHT CO.  
Waterford Steam  
Electric Station

CHARPY TEST RESULTS  
INLET NOZZLE FORGING M-1010-3

Figure  
5.2-25

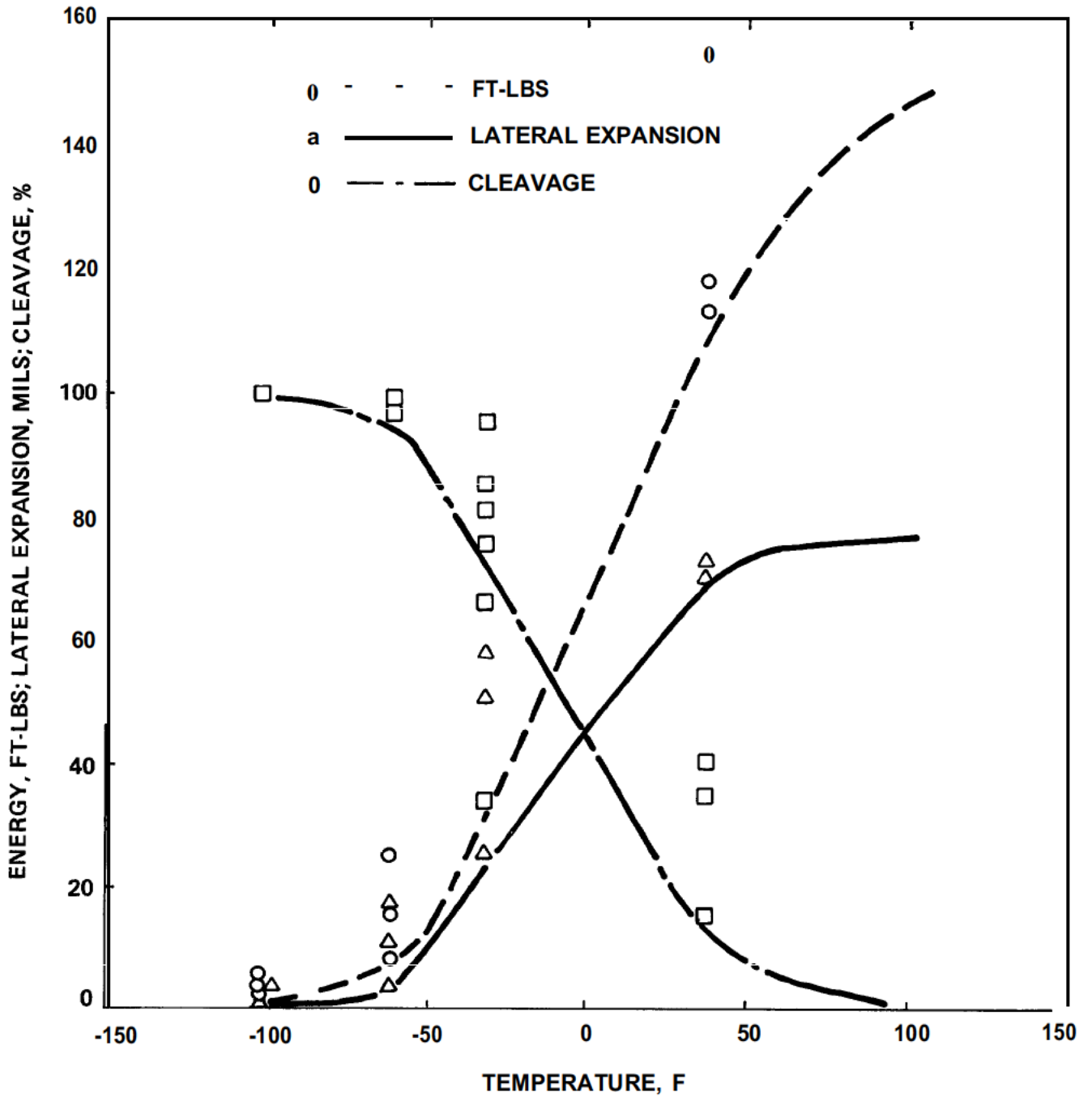


LOUISIANA  
POWER & LIGHT CO.  
Waterford Steam  
Electric Station

CHARPY TEST RESULTS  
INLET NOZZLE FORGING M-1010-4

Figure  
5.2-26

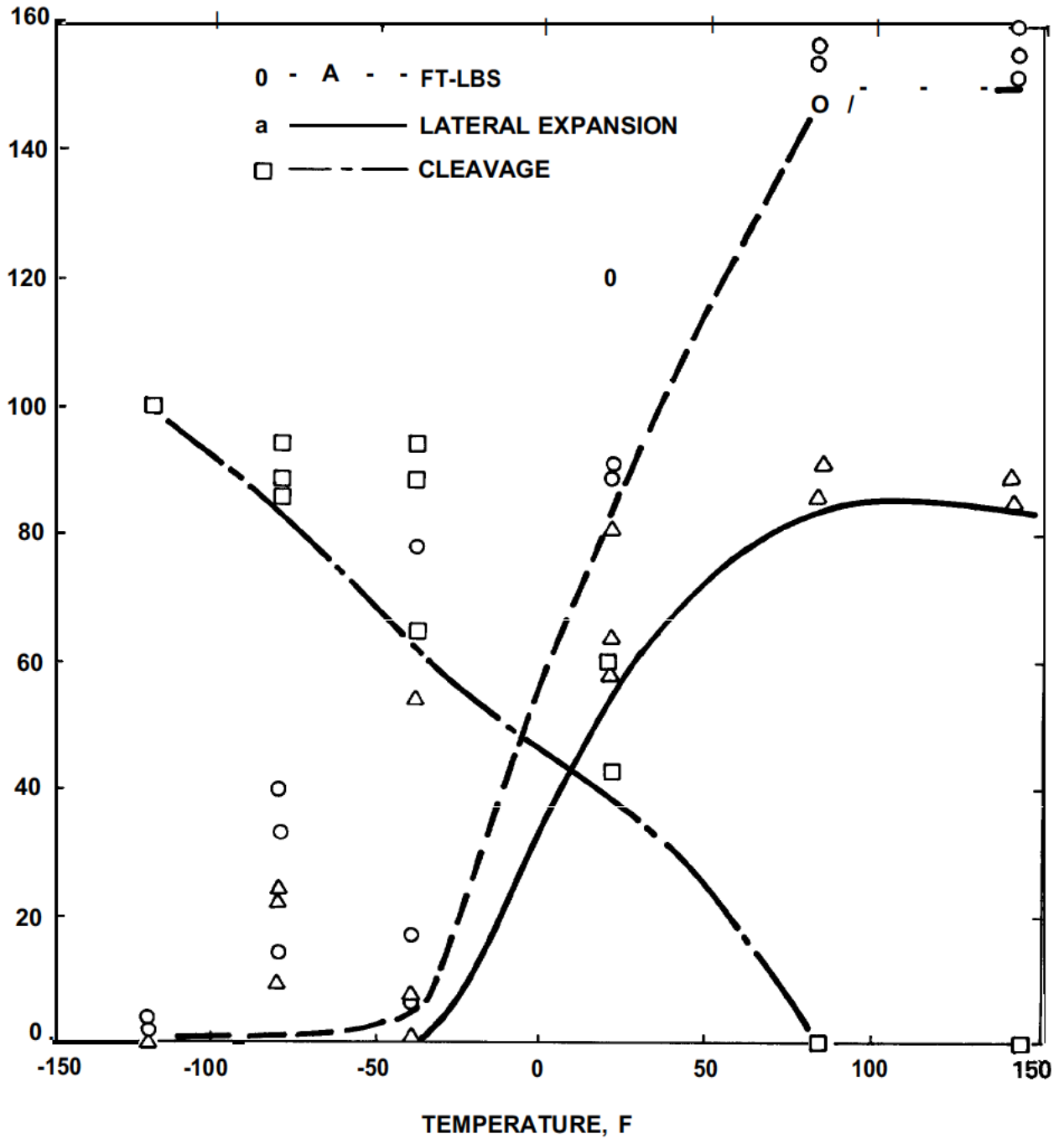




LOUISIANA  
POWER & LIGHT CO.  
Waterford Steam  
Electric Station

CHARPY TEST RESULTS  
OUTLET NOZZLE FORGING M-101 1-2

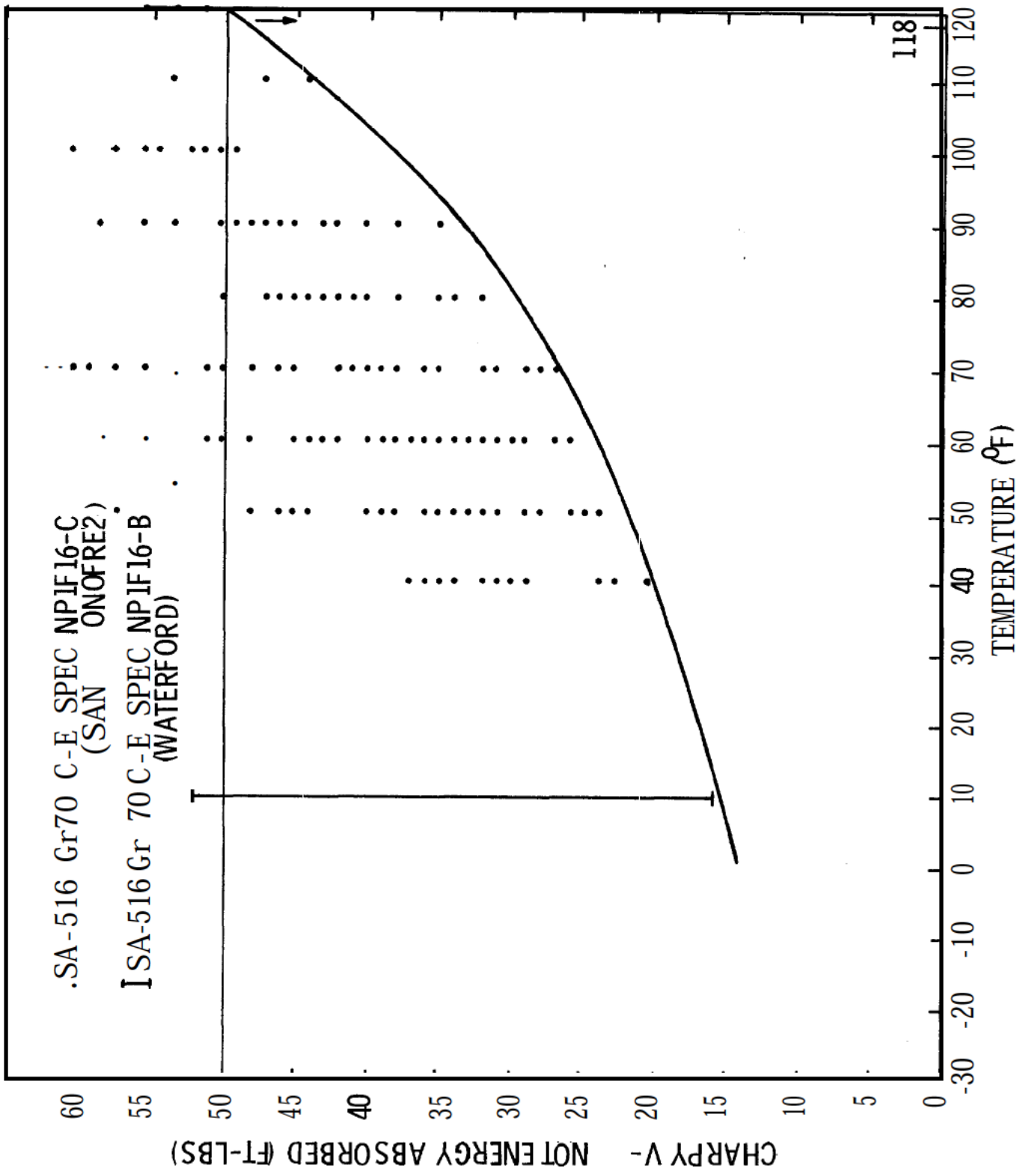
Figure  
5.2-27



LOUISIANA  
 POWER & LIGHT CO.  
 Waterford Steam  
 Electric Station

CHARPY TEST RESULTS  
 VESSEL FLANGE M-1001-1

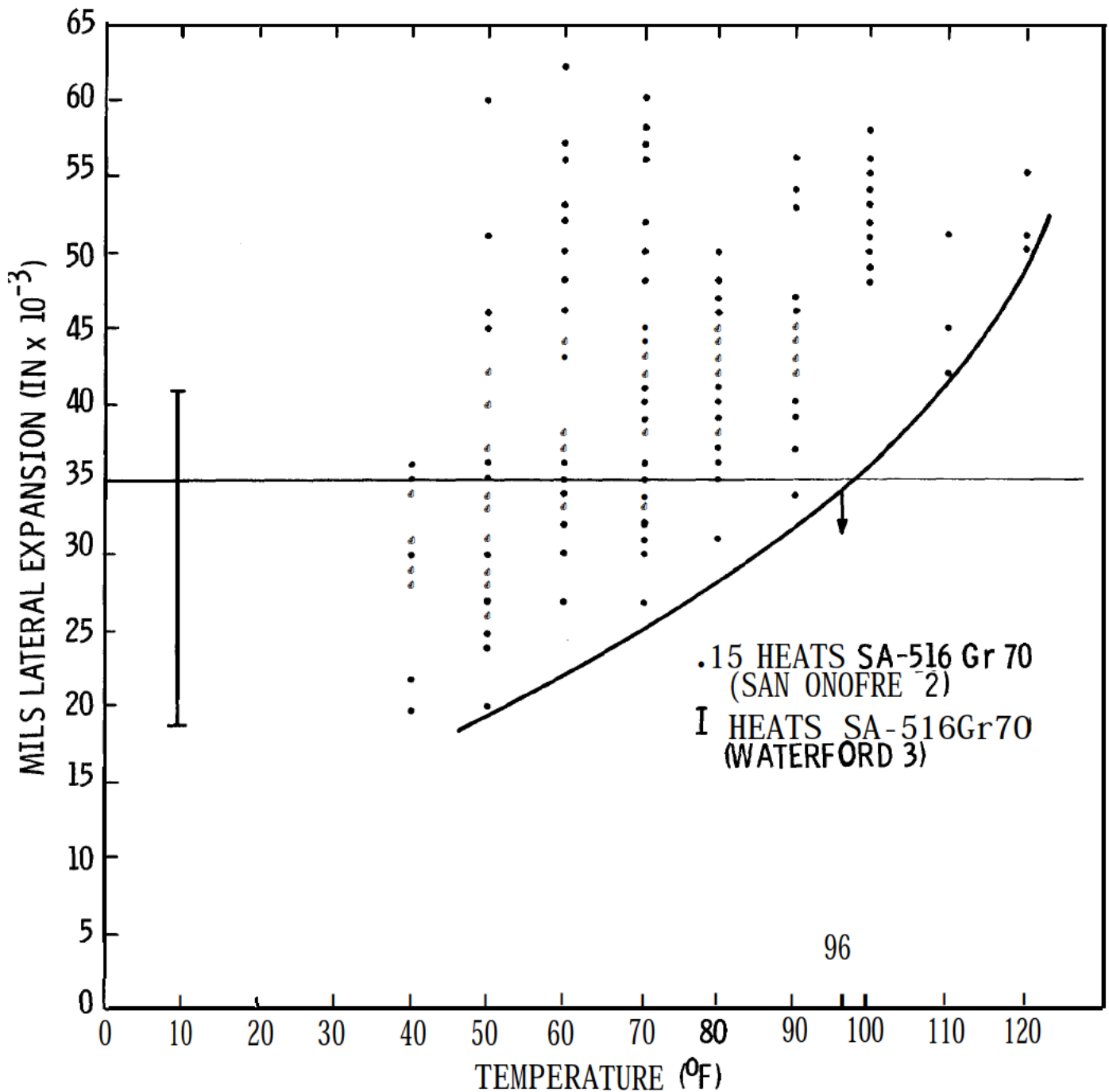
Figure  
 5.2-28



LOUISIANA  
 POWER & LIGHT CO.  
 Waterford Steam  
 Electric Station

CHARPY V-NOTCH TESTS  
 ENERGY ABSORBED VS TEMPERATURE

Figure  
 5.2-29



LOUISIANA  
POWER & LIGHT CO.  
Waterford Steam  
Electric Station

CHARPY V-NOTCH TESTS  
LATERAL EXPANSION VS TEMPERATURE

Figure

5.2-30

➔(DRN 00-1059, R11-A; 03-2059, R14)

APPENDIX 5.2A

WATERFORD-3 OVERPRESSURE PROTECTION REPORT

The overpressure protection report required by the ASME Boiler and Pressure Vessel Code, subsection NB-7300 of Section III, Division 1, is documented in Reference 5.2A-1.

➔(EC-8458, R307)

References:

5.2A-1 CN-SEE-II-09-28, "Waterford-3 Overpressure Protection Report for RSGs."

←(DRN 00-1059, R11-A; 03-2059, R14; EC-8458, R307)

WSES-FSAR-UNIT-3  
APPENDIX 5.2A  
TABLE OF CONTENTS

→(DRN 03-2059, R14)

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←(DRN 03-2059, R14)

WSES-FSAR-UNIT-3

APPENDIX 5.2A

LIST OF TABLES

→(DRN 03-2059, R14)

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WSES-FSAR-UNIT-3

APPENDIX 5.2A

LIST OF FIGURES

→(DRN 03-2059, R14)

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WSES-FSAR-UNIT-3

APPENDIX 5.2A

→(DRN 03-2059, R14)

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WSES-FSAR-UNIT-3

→(DRN 03-2059, R14)

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Revision 14 (12/05)



WSES-FSAR-UNIT-3

→(DRN 03-2059, R14)

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←(DRN 03-2059, R14)

Revision 14 (12/05)

WSES-FSAR-UNIT-3

→(DRN 03-2059, R14)

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WSES-FSAR-UNIT-3

→(DRN 03-2059, R14)

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←(DRN 03-2059, R14)

APPENDIX 5.2B

LOW TEMPERATURE OVERPRESSURE PROTECTION  
DURING HEATUP COOLDOWN AND COLD SHUTDOWN

LOW TEMPERATURE OVERPRESSURE PROTECTION  
DURING  
HEATUP, COOLDOWN, AND COLD SHUTDOWN

## 5.2B.1 SYSTEM DESCRIPTION

Overpressure protection of the Reactor Coolant System (RCS) during low temperature conditions is provided by relief valves SI-486 (2SI-R339A) and SI-487 (2SI-R340B) located in the Shutdown Cooling System (SDCS) suction line. The SDCS relief valves are shown on the piping and instrumentation diagram of Figure 6.3-1 (for Figure 6.3-1, Sheet 1, refer to Drawing G167, Sheet 1) and described in Subsection 9.3.6.2. This protection precludes any overpressurizing transient from exceeding the pressure-temperature (P-T) operating limits provided in the Technical Specifications.

→(DRN 03-2059, R14)

The protection provided by these relief valves is required during heat up and cooldown and during extended periods of cold shutdowns. Administrative controls and procedures are provided to ensure proper alignment of the system. To maintain RCS overpressure protection, the relief valves are aligned at all temperatures below the P-T curve limits corresponding to the pressurizer safety valve setpoint of 2500 psia. For temperatures above the P-T limits which correspond to the pressurizer safety valve setpoint, overpressure protection is provided by the pressurizer safety valves as described in Subsection 5.2.2 and Appendix 5.2A.

←(DRN 03-2059, R14)

## 5.2B.2 DESIGN CRITERIA

a) Credit for Operator Action

No credit is taken for operator action until 10 minutes after the operator is made aware that a transient is in progress.

b) Single Failure

The SDCS relief valve is designed to protect the reactor vessel given any event initiating a pressure transient as a result of an operator error or equipment malfunction. The redundant SDCS suction line trains meet the single failure criteria as described in Subsection 9.3.6.2. No single failure of an isolation valve will prevent the relief valves from performing their intended function.

c) Testability

Periodic testing of the SDCS suction isolation valves is defined in the Technical Specifications and Section XI of the ASME code.

d) Seismic Design and IEEE 279 Criteria

The SDCS suction line relief valves, isolation valves, associated interlocks and instrumentation are designed to safety class 2 seismic Category I requirements. The interlocks and instrumentation associated with the SDCS suction isolation valves satisfy the appropriate portions of IEEE 279-71 as discussed in Subsection 7.6.1.1.2.

## WSES-FSAR-UNIT-3

### e) Reliability

The use of the SDCS suction line relief valves for RCS overpressure protection does not reduce the reliability of the ECCS or SDCS.

### 5.2B.3 DESIGN AND ANALYSIS

To demonstrate that the SDCS overpressure protection meets the LTOP criteria listed in Subsection 5.2B.2, the following information is provided.

#### a) Limiting Transients

→(DRN 03-2059, R14)

←(DRN 03-2059, R14)

The most limiting transients initiated by a single operator error or equipment failure are:

1. An inadvertent safety injection actuation (mass input).
2. A reactor coolant pump start when a positive steam generator to reactor vessel  $\Delta T$  exists (energy input).

The transients were determined as most limiting by conservative analyses which maximize mass and energy additions to a water-solid RCS as a function of time.

→(DRN 03-2059, R14)

The results of the analyses demonstrate that the use of the SDCS suction line relief valves provide sufficient pressure relief capacity to mitigate the most limiting events.

#### b) Provision for Overpressure Protection

Above the P-T operating curve temperature corresponding to the LTOP enable temperature, RCS overpressure protection is provided by the pressurizer safety valves. Below this temperature, overpressure protection is provided by the SDCS relief valves.

→(EC-8458, R307)

An inadvertent pressure transient that increases RCS pressure is terminated by the SDCS relief valve with a conservative pressure margin below the Appendix G P-T limits and the SDCS design pressure. Current Appendix G P-T limits as described in Subsection 5.3.2 are bounded by the maximum allowable SDCS pressure. For the limiting transient (energy addition), the peak pressure translated to the most limiting location is 491 psia, which is less than 110% of the limiting component design pressure (493.2 psia)."

←(DRN 03-2059, R14; EC-8458, R307)

#### c) Equipment Parameters

The SDCS relief valve is a spring-loaded (bellows) liquid relief valve with sufficient capacity to mitigate the most limiting overpressurization event. Pertinent valve parameters and assumptions used in the analyses are as follows:

## WSES-FSAR-UNIT-3

<u>Parameters</u>	<u>Assumption</u>
Setpoint 430 psia	430 psia
Accumulation 10%	10%
Blowdown 10%	0%
→ (DRN 01-368; 03-2059, R14)	
← (DRN 01-368; 03-2059, R14)	
Capacity (@ 10% acc) 3345 gal/min	3089 gal/min (@ 10% acc)

### d) Administrative Controls

Administrative controls necessary to provide LTOP are limited to those controls that open the SDCS isolation valves. Before entering the low temperature region for which overpressure protection is necessary, RCS pressure is decreased to below the maximum pressure required for SDCS operation. Once the SDCS is aligned, no further specific administrative or procedural controls are needed to ensure proper overpressure protection. The SDCS will remain aligned whenever the RCS is at low temperatures and the reactor vessel head is secured. As shown on Figure 6.3-1 (for Figure 6.3-1, Sheet 1, refer to Drawing G167, Sheet 1) and in Tables 6.3-3 and 7.5-1, indication of SDCS isolation valve position is provided to enable the operator to determine that LTOP is operable.

→(DRN 03-2059, R14)

FIGURE 5.2B-1 HAS BEEN INTENTIONALLY DELETED

←(DRN 03-2059, R14)



→(DRN 03-2059, R14)

FIGURE 5.2B-2 HAS BEEN INTENTIONALLY DELETED

←(DRN 03-2059, R14)

→(DRN 03-2059, R14)

FIGURE 5.2B-3 HAS BEEN INTENTIONALLY DELETED

←(DRN 03-2059, R14)

### 5.3 REACTOR VESSEL

#### 5.3.1 REACTOR VESSEL MATERIALS

##### 5.3.1.1 Material Specifications

→(EC-1020, R307)

The principal ferritic materials used in the reactor vessel are listed in Table 5.2-3. These materials were specified to be in accordance with the ASME Boiler and Pressure Vessel Code, Section III, 1971 Edition including Summer 1971 Addenda, except for the Replacement Reactor Vessel Closure Head materials which were specified to be in accordance with the ASME Boiler and Pressure Vessel Code, Section III, 1998 Edition through 2000 Addenda.

←(EC-1020, R307)

##### 5.3.1.2 Special Processes Used for Manufacturing and Fabrication

The reactor vessel is a right circular cylinder with two hemispherical heads. No special manufacturing methods that could compromise the integrity of the vessel are used. The lower head is permanently welded to the lower end of the reactor vessel shell but the upper closure head can be removed to provide access to the reactor vessel internals. The head flange is drilled to match the vessel flange stud bolt locations. The stud bolts are fitted with spherical washers located between the closure nuts and the head flange. These washers maintain stud alignment during boltup when flexing of the head must be accommodated. The lower surface of the head flange is machined to provide a mating surface for the vessel closure seals.

The vessel flange is a forged ring with a machined ledge on the inside surface to support the core support barrel, which in turn supports the reactor internals and the core. The flange is drilled and tapped to receive the closure studs and is machined to provide a mating surface for the reactor vessel closure seals. An externally tapered transition section connects the flange to the cylindrical shell.

Sealing is accomplished by using two silver-plated, NiCrFe alloy, self-energized O-rings.

Nozzles are provided in the closure head for nuclear instrumentation and control element drive mechanisms (CEDM).

The inlet and outlet nozzles are located radially on a common plane just below the vessel flange. Ample thickness in this vessel course provides most of the reinforcement required for the nozzles. Additional reinforcement is provided for the individual nozzle attachments. A boss located around the outlet nozzles on the inside diameter of the vessel wall provides a mating surface for the core support barrel and guides the outlet coolant flow. This boss and the outlet sleeve on the core support barrel are machined to a common contour to minimize reactor coolant bypass leakage. Shell sections are joined to the nozzle region by a transition section.

Snubbers built into the lower portion of the reactor vessel shell limit the amplitude of flow-induced vibrations in the core support barrel.

## WSES-FSAR-UNIT-3

### 5.3.1.3

#### Special Methods for Nondestructive Examination

Prior to, during, and after fabrication of the reactor vessel, nondestructive tests based upon Section III of the ASME Boiler and Pressure Vessel Code were performed on all welds, forgings, and plates as indicated.

All full-penetration, pressure-containing welds were 100 percent radiographed to the standards of Section III of the ASME Boiler and Pressure Vessel Code. Weld preparation areas, back-chip areas, and final weld surfaces were magnetic-particle or dye-penetrant examined. Other pressure-containing welds, such as used for the attachments of nonferrous nickel-chromium-iron mechanism housings, vents, and instrument housings to the reactor vessel head, were inspected by liquid-penetrant tests of the root pass, the lesser of one-third of the thickness or each 1/2 in. of weld deposit, and the final surface. Additionally, the base metal weld preparation area was magnetic-particle examined prior to overlay with nickel-chromium-iron weld metal.

All forgings were inspected by ultrasonic testing, using longitudinal beam techniques. In addition, ring forgings were tested using shear wave techniques. Rejection under longitudinal beam inspection, with calibration so that the first back reflection is at least 75 percent of screen height, was based on indications causing complete loss of back reflection (when not associated with geometrical configuration).

All carbon-steel forgings and ferrite welds are also subjected to magnetic-particle examination after stress relief. Rejection is based on relevant indication of:

- a) Any cracks and linear indications
- b) Rounded indications with dimensions greater than 3/16 in.
- c) Four or more rounded indications in a line separated by less than 1/16 in. edge to edge
- d) Ten or more rounded indications in any 6.0 in.<sup>2</sup> in the most unfavorable locations

Plates were subjected to ultrasonic examination using straight beam techniques. Rejection was based on areas producing a continuous total loss of back reflection with a frequency and instrument adjustment that produce a minimum of 50 to a maximum of 75 percent of full scale reference back reflection from the opposite side of a sound area of the plate.

→(DRN 06-872, R15)

Any defect that showed a total loss of back reflection that could not be contained within a circle whose diameter is the greater of three inches or one-half the plate thickness was unacceptable. Two or more defects smaller than described above, which cause a complete loss of back reflection, shall be unacceptable unless separated by a minimum distance equal to the greatest diameter of the larger defect, unless the defects are contained within the area described above. All carbon and low-alloy steel products were magnetic-particle examined after accelerated cooling to the magnetic-particle acceptance standard cited above.

←(DRN 06-872, R15)

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→(DRN 06-872, R15)

Nondestructive testing of a vessel was performed throughout fabrication. The nondestructive examination requirements including calibration methods, instrumentation, sensitivity, and reproducibility of data, are in accordance with requirements of the ASME B&PV Code, Section III. (See Table 5.2-1). Strict quality control was maintained in critical areas such as calibration of test instruments.

←(DRN 06-872, R15)

All vessel bolting material received ultrasonic and magnetic-particle examination during the manufacturing process.

The bolting material receives a straight-beam, radial-scan, ultrasonic examination with a search unit not exceeding one square in. area. The standard for rejection was 50 percent loss of first back reflection or an indication in excess of 20 percent of the height of the back reflection. All hollow material receives a second ultrasonic examination using angle beam, radial scan with a search unit not exceeding one square in. in area. A reference specimen of the same composition and thickness containing a notch (located on the inside surface) one in. in length and a depth of three percent of nominal section thickness, or 3/8 in., whichever is less, was used for calibration.

Any indications exceeding the calibration notch amplitude are unacceptable. Use of these techniques ensures that no materials that have unacceptable flaws, observable cracks, or sharply defined linear defects were used.

→(DRN 06-911, R15)

The magnetic-particle inspection was performed both before and after threading of the studs. Axially aligned defects whose lengths were greater than one in. and nonaxial defects were unacceptable.

←(DRN 06-911, R15)

Upon completion of all postweld heat treatments, the reactor vessel was hydrostatically tested at 3125 psig after which all accessible ferritic weld surfaces, including those of welds used to repair material, were magnetic-particle inspected in accordance with Section III of the ASME Code.

### 5.3.1.4 Special Controls for Ferritic and Austenitic Stainless Steels

Special controls for ferritic and austenitic stainless steels are as follows:

→(DRN 00-1059, R11-A)

- a) Regulatory Guide 1.31, Control of Stainless Steel Welding is addressed in Subsection 5.2.3.4.
- b) Regulatory Guide 1.34, Control of Electroslag Weld Properties is addressed in Subsection 5.2.3.3.
- c) Regulatory Guide 1.43, Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components is addressed in Subsection 5.2.3.3.
- d) Regulatory Guide 1.44, Control of the Use of Sensitized Stainless Steel is addressed in Subsection 5.2.3.4.

←(DRN 00-1059, R11-A)

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→(DRN 00-1059, R11-A)

e) Regulatory Guide 1.50, Control of Preheat Temperature for Welding of Low-Alloy Steel is addressed in Subsection 5.2-3-3.

f) Regulatory Guide 1.71, Welder Qualification for Areas of Limited Accessibility is addressed in Subsection 5.2.3.3.

←(DRN 00-1059, R11-A)

g) Regulatory Guide 1.99, Effects of Residual Element on Predicted Radiation Damage to Reactor Vessel Materials

→(DRN 03-2059, R14)

Westinghouse previously took exception to the methods and procedure for predicting radiation damage to pressure vessel steels contained in Regulatory Guide 1.99. Westinghouse's formal position on Regulatory Guide 1.99 was forwarded to the U.S. NRC in September, 1975 <sup>(1)</sup>. The methods contained in Regulatory Guide 1.99 for predicting  $RT_{NDT}$  shift and Charpy upper shelf energy decreases with irradiation are not appropriate for determining the irradiation behavior of A533-B, Class 1, materials. The methods utilized are based on non-A533-B materials data and incorporate incorrect assumptions concerning the irradiation behavior of vessel materials.

The curve shown in Figure 5.3-1 is utilized for predicting the  $RT_{NDT}$  shift of reactor vessel material with low copper content. The curve is based on 550 °F irradiation data for A533-B materials. The data base was collected from published works on the subject of irradiation damage in reactor vessel materials and from test data generated by a joint research program with Westinghouse (then Combustion Engineering), NRC and the Naval Research Laboratory (NRL). Table 5.3-1 lists this data. The indicated literature references for the data are listed in Table 5.3-2. The  $RT_{NDT}$  shift prediction curves are shown in relation to the data in Figure 5.3-1. Weld and plate irradiation behavior is considered separately, because research has shown that some weld metal tends to be more sensitive to irradiation damage. The curve is conservatively drawn, envelopes the data, and follows trends described by the data.

Regulatory Guide 1.99 is now used without exception. \

←(DRN 03-2059, R14)

### 5.3.1.5 Fracture Toughness

→(EC-1020, R307, LBDCR 17-020, R310)

The reactor vessel materials were ordered to the ASME 1971 Code Section III, Summer 1971 Addenda, specification, except for the Replacement Reactor Vessel Closure Head materials which were ordered to the ASME Boiler and Pressure Vessel Code, Section III, 1998 Edition through 2000 Addenda. The materials meet the Charpy impact requirements of Subsection NB-2300 (three tests at a temperature to verify 30 ft.-lbs. of absorbed energy) of the Summer 1971 Addenda to the Code. Transverse (weak direction) Charpy test data was used to develop  $RT_{NDT}$ s per Section III subsection NB-2331 of the 1973 Summer Addenda to the Code for the vessel shell plates. The highest  $RT_{NDT}$  value for the Waterford 3 reactor vessel plate materials is 0°F (lower shell plate M-1004-2)<sup>[8]</sup>. This  $RT_{NDT}$  was determined using longitudinal dropweight test results and transverse Charpy results for conservatism, since the transverse dropweight results would have yielded a lower (less conservative)  $RT_{NDT}$ . Dropweight testing of the reactor vessel weld materials as part of weld material certification was not required by the Summer 1971 Addenda to the Code, and thus it was not consistently performed. A combination of available dropweight data from WF3 welds and sister welds at other facilities, conservatively bounding  $RT_{NDT}$  developed from the WF3 SMAW welds, and C-E generic  $RT_{NDT}$  for Linde type 0091 flux submerged arc welds was used to determine the  $RT_{NDT}$  of all welds<sup>[8]</sup>. Drop weight testing of vessel weld and heat affected zone material from WPQs was not required by the 1971 Summer addenda of the Code, and the materials were not available.

←(EC-1020, R307; LBDCR 17-020, R310)

→(LBDCR 17-020, R310)

Transverse (weak direction) Charpy impact data on plate M-1004-2, weld and heat-affected-zone (HAZ) material is reported in Subsection 5.3.1.6-1, as results of the baseline surveillance testing. This testing, which establishes an  $RT_{NDT}$  in a manner consistent with Appendix G 10CFR50, yields an  $RT_{NDT}$  for plate

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M-1004-2 of  $-20^{\circ}\text{F}$  ( $0^{\circ}\text{F}$  when more conservative longitudinal dropweight data is used<sup>[8]</sup>). The  $RT_{\text{NDT}}$  for the plate, weld and HAZ materials are shown in Table 5.3-3, and the Charpy data is plotted in Figures 5.3-2, -3 and -4. The properties in Table 5.3-3 were updated from their original values<sup>(8,9)</sup> based on use of hyperbolic tangent fitting of the lower bound Charpy curve for each surveillance material in accordance with ASME B&PV Code Section III NB-2331 and ASTM E185-82.

From both the initial vessel material fracture toughness testing and surveillance material baseline testing, the lowest reported Charpy upper shelf energy for the reactor vessel beltline materials is 106 ft-lbs (intermediate shell longitudinal weld 101-124A) for welds and 108 ft-lbs (intermediate shell plate M-1003-1) for plates<sup>[8]</sup>. The lowest reported USE for extended beltline materials is 87 ft-lbs (upper to intermediate girth weld 106-121) for welds and 95 ft-lbs (Upper Shell Plate M-1002-2) for plates<sup>(8)</sup>. These are well in excess of the 75 ft-lb requirement of 10 CFR 50, Appendix G<sup>[4]</sup>.

In the case of the surveillance program materials, for which two sets of fracture toughness data exist, the CE initial fracture toughness test results were used for calculating future projected USE,  $RT_{\text{PTS}}$ , and ART because the surveillance program baseline Charpy tests were not conducted in sets of 3 per temperature.

←(LBDCR 17-020, R310)

### 5.3.1.6 Material Surveillance

The irradiation surveillance program for Waterford 3 will be conducted to assess the neutron-induced changes in the  $RT_{\text{NDT}}$  (reference temperature) and the mechanical properties of the reactor vessel materials. Changes in the impact and mechanical properties of the material will be evaluated by the comparison of pre- and post-irradiation test specimens. The capsules containing the surveillance test specimens used for monitoring the neutron induced property changes of the reactor vessel materials will be irradiated under conditions which represent, as closely as practically possible, the irradiation conditions of the reactor vessel.

ASTM E-185-82, Standard Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels, and 10CFR50, Appendix H, Reactor Vessel Material Surveillance Program Requirements, present criteria for monitoring changes in the fracture toughness properties of reactor vessel beltline materials through surveillance programs. This reactor vessel surveillance program for Waterford 3 adheres to all of the requirements of ASTM E-185-82 and satisfies 10CFR50, Appendix H.

#### 5.3.1.6.1 Test Materials Selection

→(DRN 03-2059, R14)

Regions of both the intermediate and lower shells of the reactor vessel are nearest to the reactor core and, therefore, sustain the greatest neutron exposure. The material from which surveillance test specimens were manufactured were cut from that plate in the core region which would become the limiting plate with respect to reactor operation during its lifetime. This material (lower shell plate M-1004-2) was selected on the basis of highest initial  $RT_{\text{NDT}}$ , chemical composition and fluence. The test materials were processed so that they are representative of the materials in the completed reactor vessel. A record of chemical analyses, fabrication history and mechanical properties of the shell plate from which the surveillance test materials were prepared is maintained. The results of mill test chemistries for the six plates of the beltline region of the vessel are presented in Table 5.2-5.

→(DRN 00-1059, R11-A)

→(DRN 03-2059, R14)

Three metallurgically different materials representative of the reactor vessel were used for test specimens. These include base metal, weld metal and heat affected zone (HAZ) materials. In addition to the materials from sections of reactor vessel shell plate, material from a standard heat of ASTM A533-B Class I manganese-molybdenum-nickel steel made available by the USAEC sponsored Heavy Section Steel Technology (HSST) program is also included. This

←(DRN 00-1059, R11-A)

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reference material has been fully processed and characterized, and is used for Charpy impact specimen correlation monitors to permit comparisons among the irradiation data from operating power reactors and irradiation data from experimental reactors. Compilation of data generated from post-irradiation tests of the correlation monitors will be carried out by the HSST program.

### 5.3.1.6.1.1 Base Metal

→(LBDCR 17-002, R310)

Base metal test material was manufactured from sections of lower shell plate M-1004-2 which was found to have the combination of  $RT_{NDT}$ , chemical composition (Cu and P), and neutron fluence during service, which would first appear to limit the vessel operating lifetime. The unirradiated  $RT_{NDT}$  of each plate in the intermediate and lower shells was determined from drop weight and Charpy data as required in NB-2331 of the ASME Boiler Code, 1971 Edition Summer 1973 Addenda, Section III and is shown in Table 5.2-6. All base metal test material is cut from one shell.

←(LBDCR 17-002, R310)

The section of shell plate used for the base metal test material is adjacent to the test material used for ASME Code Section III tests and is at a distance of at least one plate thickness from any water-quenched edge. This material was heat-treated to a metallurgical condition which is representative of the final metallurgical condition of the base metal in the completed reactor vessel.

### 5.3.1.6.1.2 Welded Plates

Weld metal and HAZ material were produced by welding together sections from the selected base metal plate and another intermediate plate of the reactor vessel. The HAZ test material was manufactured from a section of the same shell plate used for base metal test material.

The sections of shell plate used for weld metal and HAZ test material are adjacent to the test material used for ASME Code Section III tests and are at a distance of at least one plate thickness from any water-quenched edge. The procedure used for making the intermediate-to-lower shell girth weld in the reactor vessel was followed in the manufacture of the weld metal and HAZ test materials. The welded plates were heat-treated to metallurgical conditions that are representative of the final metallurgical conditions of similar materials in the completed reactor vessel.

The test specimens used in establishing the unirradiated  $RT_{NDT}$  temperature of the base metal were obtained from  $1/4 T$  (where  $T$  is plate thickness) locations of sections of the plate used in the core region. The heat-affected-zone samples were taken from the inner region of the deposited weld metal. The impact properties of the specimen locations are representative of the material through the entire thickness. Use of the  $RT_{NDT}$  values obtained from samples taken from the inner regions of the test materials represent a conservative approach for establishing the initial minimum operating temperature and the base for the predicted minimum operating temperature after irradiation, because the advantages of the more favorable  $RT_{NDT}$  properties of the surface regions are not taken into consideration.



### 5.3.1.6.2 Test Specimens

#### 5.3.1.6.2.1 Type and Quantity

→(DRN 00-1059, R11-A; LBDCR 16-042, R310)

The magnitude of the neutron-induced property changes of the reactor vessel materials is determined by comparing the results of tests using irradiated impact and tensile specimens to the results of similar tests using unirradiated specimens. The changes in  $RT_{NDT}$  of the vessel materials are determined by adding to the reference temperature ( $RT_{NDT}$ ) the amount of the temperature shift in the Charpy test curves between the unirradiated material and the irradiated material, measured at the 30 ft.-lb. energy level. The new values of  $RT_{NDT}$  are known as adjusted reference temperature.

←(DRN 00-1059, R11-A; LBDCR 16-042, R310)

Drop weight, Charpy impact, and tensile test specimens were provided for unirradiated tests. Drop weight tests were conducted in accordance with ASTM E-208. Charpy impact tests were conducted in accordance with ASTM E-3. Tensile tests were conducted in accordance with ASTM E-8 and E-21. Correlation of drop weight and Charpy impact tests to establish  $RT_{NDT}$  were made in accordance with NB-2300 of the ASME Code, Section III. Charpy impact and tensile test specimens are provided for post-irradiation tests.

The total quantity of specimens furnished for carrying out the overall requirements of this program is presented in Table 5.3-4. A sufficient amount of base metal, weld metal, and HAZ test material to provide two additional sets of test specimens has been obtained with full documentation and identification for future evaluation should the need arise. Each of the test materials has been chemically analyzed for approximately 21 elements, including all those listed in ASTM E-185-82.

#### 5.3.1.6.2.2 Unirradiated Specimens

The type and quantity of test specimens provided for establishing the properties of the unirradiated reactor vessel materials are presented in Table 5.3-5. The data from tests of these specimens provide the basis for determining the neutron-induced property changes of the reactor vessel materials.

##### a) Drop Weight Test Specimens

Twelve drop weight test specimens of base metal (longitudinal and transverse), weld metal, and HAZ material are provided for establishing the NDTT of the unirradiated surveillance materials. These data form the basis for  $RT_{NDT}$  determination.  $RT_{NDT}$  is the reference temperature from which subsequent neutron-induced changes are determined.

##### b) Charpy Impact Test Specimens

Thirty test specimens each of base metal (longitudinal and transverse), weld metal, and HAZ material are provided. This quantity exceeds the minimum number of test specimens recommended by ASTM E-185 for developing a Charpy impact energy transition

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curve and is intended to provide a sufficient number of data points for establishing accurate Charpy impact energy transition temperatures for these materials. This data, together with the drop weight NDTT, is used to establish an  $RT_{NDT}$  for each material.

### c) Tensile Test Specimens

Eighteen tensile test specimens each of base metal (longitudinal and transverse), weld metal and HAZ materials are provided. This quantity also exceeds the minimum number of test specimens recommended by ASTM E-185 and is intended to permit a sufficient number of tests for accurately establishing the tensile properties for these materials at a minimum of three test temperatures (e.g., ambient, operating and design).

### 5.3.1.6.2.3 Irradiated Specimens

Both tensile and impact test specimens are used for determining changes in the static and dynamic properties of the materials due to neutron irradiation. A total of 288 Charpy impact and 54 tensile test specimens is provided. The type and quantity of test specimens provided for establishing the properties of the irradiated materials over the lifetime of the vessel are presented in Table 5.3-6. The attachment of the capsule assemblies to the inside wall of the reactor vessel is described in CENPD-155P. <sup>(2)</sup>

### 5.3.1.6.3 Specimen Irradiation

#### 5.3-1.6.3.1 Encapsulation of Specimens

The test specimens are placed within corrosion-resistant capsule assemblies:

- a) To prevent corrosion of the carbon steel test specimens by the primary coolant during irradiation.
- b) To physically locate the test specimens in selected locations within the reactor
- c) To provide a means by which the irradiation conditions (fluence,<sup>(a)</sup> flux spectrum, temperature) can be determined
- d) To facilitate the removal of a desired quantity of test specimens from the reactor when a specified fluence has been attained.

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(a) Time integrated neutron flux.

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A typical capsule assembly, illustrated in Figure 5.3-5, consists of a series of seven specimen compartments, connected by wedge couplings, and a lock assembly. Each compartment enclosure of the capsule assembly is internally supported by the surveillance specimens and is externally pressure tested to 3125 psia during final fabrication. The wedge couplings also serve as end caps for the specimen compartments and position the compartments within the capsule holders which are attached to the reactor vessel. The lock assemblies fix the locations of the capsules within the holders by exerting axial forces on the wedge coupling assemblies which cause these assemblies to exert horizontal forces against the sides of the holders preventing relative motion. The lock assemblies also serve as a point of attachment for the tooling used to remove the capsules from the reactor.

Each capsule assembly is made up of four Charpy impact test specimen (Charpy impact) compartments and three tensile test specimen - flux/temperature monitor (tensile-monitor) compartments. Each capsule compartment is assigned a unique identification so that a complete record of test specimen location within each compartment can be maintained.

### a) Charpy Impact Compartments

Each Charpy impact compartment (Figure 5.3-6) contains 12 impact test specimens. This quantity of specimens provides an adequate number of data points for establishing a Charpy impact energy transition curve for a given irradiated material. Comparison of the unirradiated and irradiated Charpy impact energy transition curves permits determination of the  $RT_{NDT}$  changes due to irradiation for the various materials.

The specimens are arranged vertically in four 1 x 3 arrays and are oriented with the notch toward the core. The temperature differential between the specimen and the reactor coolant is minimized by using spacers between the specimens and the compartment and by sealing the entire assembly in an atmosphere of helium.

### b) Tensile - Monitor Compartments

Each tensile-monitor compartment (Figure 5.3-7) contains three tensile test specimens, a set of flux spectrum monitors and a set of temperature monitors, for estimating the maximum temperature to which the specimens have been exposed. The entire tensile-monitor compartment is sealed within an atmosphere of helium.

The tensile specimens are placed in a housing machined to fit the compartment. Split spacers are placed around the gage length of the specimen to minimize the temperature differential between the specimen gage length and the coolant.

### 5.3.1.6.3.2 Flux and Temperature Measurement

→(LBDCR 17-002, R310)

The changes in the  $RT_{NDT}$  of the reactor vessel materials are derived from specimens irradiated to various fluence levels and in different neutron energy spectra. In order to permit accurate predictions of the ART of the vessel materials, complete information on the neutron flux, neutron energy spectra, and the irradiation temperature of the surveillance specimens must be available.

←(LBDCR 17-002, R310)

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### a) Flux Measurements

Fast neutron flux measurements are obtained by insertion of threshold detectors into each of the six irradiation capsules. Such detectors are particularly suited for the proposed application, because their effective threshold energies lie in the low Mev range. Selection of threshold detectors is based on the recommendations of ASTM E-261, "Method of Measuring Neutron Flux by Radioactive Techniques".

These neutron threshold detectors and the thermal neutron detectors, listed in Table 5.3-7, can be used to monitor the thermal and fast neutron spectra incident on the test specimen. These detectors possess reasonable long half-lives and activation cross sections covering the desired neutron energy range.

One set of flux spectrum monitors is included in each tensile monitor compartment. Each detector is placed inside a sheath which identifies the material and facilitates handling. Cadmium covers are used for those materials (e.g., uranium, nickel, copper and cobalt) which have competing neutron capture activities.

The flux monitors are placed in holes drilled in stainless steel housings as shown in Figure 5.3-7 at three axial locations in each capsule assembly (Figure 5.3-5) to provide an axial profile of the level of fluence which the specimens attain.

In addition to these detectors, the program also includes correlation monitors (Charpy impact test specimens made from a reference heat of ASTM A533-B, Class 1, manganese-molybdenum-nickel steel) which are irradiated along with the specimens made from reactor vessel materials. The changes in impact properties of the reference material provide a cross-check on the dosimetry in any given surveillance program. These changes also provide data for correlating the results from this surveillance program with the results from experimental irradiations and other reactor surveillance programs using specimens of the same reference material.

### b) Temperature Estimates

Because the changes in mechanical and impact properties of irradiated specimens are highly dependent on the irradiation temperature, it is necessary to have knowledge of the temperature of specimens as well as the pressure vessel. During irradiation, instrumented capsules are not practical for a surveillance program extending over the design lifetime of a power reactor. The maximum temperature of the irradiated specimens can be estimated with reasonable accuracy by including within the capsule assembly small pieces of low melting point alloys or pure metals. The compositions of candidate materials with melting points in the operating range of power reactors are listed in Table 5.3-8. The monitors are selected to bracket the operating temperature range of the reactor vessel.

The temperature monitors consist of a helix of low melting alloy wire inside a sealed quartz tube. A stainless steel weight is provided to destroy the integrity of the wire when the melting point of the alloy is reached. The compositions and therefore the melting temperatures of the temperature monitors are differentiated by the physical lengths of the quartz tubes which contains the alloy wires.

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A set of temperature monitors is included in each tensile-monitor compartment. The temperature monitors are placed in holes drilled in stainless steel housings as shown in Figure 5.3-7 and are also placed at three axial locations in each capsule assembly (Figure 5.3-5) to provide an axial profile of the maximum temperature to which the specimens were exposed.

### 5.3.1.6.3.3 Irradiation Locations

The encapsulated test specimens are irradiated at approximately identical radial positions about the midplane of the core. The test specimens are enclosed within six capsule assemblies the axial positions of which are bisected by the midplane of the core. A summary of the specimens contained in each of these capsule assemblies is presented in Table 5.3-9.

The test specimens contained in the capsule assemblies are used for monitoring the neutron-induced property changes of the reactor vessel materials. These capsules, therefore, are positioned near the inside wall of the reactor vessel so that the irradiation conditions (fluence, flux spectrum, temperature) of the test specimens resemble as closely as possible the irradiation conditions of the reactor vessel. The neutron fluence of the test specimens is expected to be approximately 50 percent greater than that seen by the adjacent vessel wall.

The  $RT_{NDT}$  changes resulting from the irradiation of these specimens closely approximate the  $RT_{NDT}$  changes in the materials of the reactor vessel.

The capsule assemblies are placed in capsule holders positioned circumferentially about the core at locations which include the regions of maximum flux. Figure 5.3-8 shows the location of the capsule assemblies.

All capsule assemblies are inserted into their respective capsule holders during the final reactor assembly operation.

### 5.3.1.6.3.4 Capsule Assembly Removal

The capsule assemblies remain within their capsule holders until the test specimens contained therein have attained desired levels of exposure (EFPY). At that time, selected capsule assemblies are removed. The distribution of target exposures for removal of capsule assemblies is presented in Table 5.3-10.

→

The target exposure levels for the surveillance capsules are based on the time intervals indicated in the withdrawal schedule in ASTM E-185-82, referenced in 10CFR50, Appendix H.

←

Withdrawal schedules may be modified to coincide with those refueling outages or plant shutdowns which most closely approach the withdrawal schedule.

During unit start-up and shutdown, the rates of temperature and pressure changes are limited. The design number of cycles for heatup and cooldown is based upon a rate of 100° F/hr and for cyclic operation.

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The maximum allowable reactor coolant system pressure at any temperature is based upon the stress limitations for brittle fracture considerations. These limitations are derived by using the rules contained in Section III of the ASME Code including Appendix G, Protection Against Nonductile Failure and the rules contained in 10CFR50, Appendix G, Fracture Toughness Requirements. Compliance with the criteria in 10CFR50, Appendix H is discussed in Subsection 5.3.1.6.

### 5.3.1.7 Reactor Vessel Fasteners

The stud material for the reactor vessel closure head is fabricated from SA 540 Grade B24 Class III material. The nuts and washers for the reactor vessel fasteners are made from SA 540 Grade B24 or B23 material. These materials were ordered prior to the issuance of 10CFR50, Appendix G and Regulatory Guide 1.65, "Materials and Inspections for Reactor Vessel Closure Studs". Material tests for Waterford 3 stud material demonstrates adequate toughness in accordance with ASME Code Section III, 1971 Edition through Summer 1971 Addenda, and an acceptable level of ultimate tensile strength which is consistent with the recommendations of Reg Guide 1.65 (see Table 5.3-11). Test results demonstrate that the stud material meets the 25 mil lateral expansion, 45 ft-lb criteria of 10CFR50, Appendix G at 10°F and that they do not exceed 170 KSI ultimate tensile strength (UTS).

Testing adequate to establish compliance with the ASME Code Section III, 1971 Edition through Summer 1971 Addenda, was done for the nut and washer material. 10CFR50, Appendix G, Section IV, "Fracture Toughness Requirements", Paragraph 4 requires that material for bolting and other fasteners with nominal diameters exceeding one in. shall meet the minimum requirements of 25 mils lateral expansion and 45 ft-lbs in terms of Charpy V-notch tests conducted at the preload temperature or at the lowest service temperature, whichever is lower.

In order to determine whether the nuts and washers met the 10CFR50 Appendix G requirements, all Charpy test data for SA540 Gr B-24 steel (Southern California Edison Electric Co.'s San Onofre Generating Station Units 2 and 3 and Waterford Unit 3 reactor vessel studs, washers and nuts) was accumulated. Waterford Unit 3 data is presented in Table 5.3-12 and all the data is graphically presented in Figure 5.3-9 (Charpy Energy Absorbed) and Figures 5.3-10 (Mils Lateral Expansion). It can be seen that the available data plotted in Figures 5.3-9 indicates that a lower bound curve through minimum points yields a Charpy absorbed energy value of approximately 46 ft-lbs at +60°F. Further, a similar lower bound curve in Figure 5.3-10 yields a value of 27 mils lateral expansion at +60°F. Since both these curves are minimum point curves for data points from six separate heats, the temperature necessary to meet 10CFR50, Appendix G requirements ( $C_v = 45$  ft-lbs, lateral expansion = 25 mils) is 60°F.

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### 5.3.2 PRESSURE TEMPERATURE LIMITS

#### 5.3.2.1 Limit Curves

→(LBDCR 17-002, R310)

The reactor vessel beltline material consists of six plates. The nil ductility transition temperatures ( $T_{NDTT}$ ) of each plate was established by drop weight test. Charpy tests were then performed to determine at what temperature the plates exhibited 50 ft-lb absorbed energy and 35 mils lateral expansion. From this testing a reference temperature for transverse direction ( $RT_{NDT}$ ) of 0°F was established.

←(LBDCR 17-002, R310)

For the remaining material in the RCS, a limiting  $RT_{NDT}$  of 90°F was established based upon SA 105 Class 2 material used to fabricate the lower driver mount flanges of the reactor coolant pumps.

→(DRN 00-1059, R11-A)

As a result of fast neutron irradiation in the region of the core,  $RT_{NDT}$  will increase with operation. The techniques used to analytically and experimentally predict the integrated fast neutron ( $E \geq 1$  MeV) fluxes of the reactor vessel are described in Subsections 5.3.1.4 and 5.3.1.6. Extent of compliance with Regulatory Guide 1.99 is discussed in Subsection 5.3.1.4.

→(DRN 03-2059, R14, LBDCR 17-020, R310)

Since the neutron spectra and flux measured at the samples and reactor vessel inside radius should be nearly identical, the measured reference transition temperature shift for a sample can be applied to the adjacent section of the reactor vessel for later stages in plant life equivalent to the difference in calculated flux magnitude. The maximum exposure of the reactor vessel will be obtained from the measured sample exposure by application of the calculated azimuthal neutron flux variation. The peak end-of-license (32 EFPY) neutron fluence ( $E > 1.0$  MeV) at the core midplane for the Waterford Unit 3 reactor vessel is  $2.48 \times 10^{19}$  n/cm<sup>2</sup>[6]. That is the fluence corresponding to the clad-base metal interface. Projections of neutron fluence beyond Cycle 11 were based on a 1.5% uprate (3441 MWt) at the start of Cycle 12 and a 8% uprate (3716 MWt) at the start of Cycle 14. The highest predicted Adjusted Reference Temperature (ART) at 32 EFPY is 27.5°F<sup>(7)</sup>. This corresponds to an integrated fast neutron fluence ( $E > 1.0$  MeV) at the ¼ thickness of  $1.48 \times 10^{19}$  n/cm<sup>2</sup> and was determined using the methodology of Regulatory Guide 1.99, Revision 2. The actual shift in  $RT_{NDT}$  is established periodically during plant operation by testing of reactor vessel material samples which are irradiated cumulatively by securing them near the inside wall of the reactor vessel as described in Subsection 5.3.1.6 and shown in Figure 5.3-8. To compensate for any increase in the  $RT_{NDT}$  caused by irradiation, limits on the pressure-temperature relationship are periodically changed to stay within the stress limits during heatup and cooldown. During the first 10 years of reactor operation, a conservatively high fluence of  $9.2 \times 10^{18}$  n/cm<sup>2</sup> was assumed which corresponds to 3580 Mwt and 80 percent load factor. The corresponding  $\Delta RT_{NDT}$  was 75°F based on the curve shown in Figure 5.3-1. Thus, for this interval, the upper limit to the  $RT_{NDT}$  was (initial + shift) or 22°F + 75°F = 97°F. This is greater than (and conservatively bounds) the 40 years Adjusted Reference Temperature at 88°F calculated from the methodology of Regulatory Guide 1.99, Revision 2. The limit lines identified in Technical Specification 3/4.4.8 are based on the following<sup>[7]</sup>:

←(DRN 00-1059, R11-A; 03-2059, R14, LBDCR 17-020, R310)

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→(LBDCR 17-002, R310)

a) Heatup and Cooldown Curves (from Section XI of the ASME Code Appendix G-2215)

$$K_{IC} = 2K_{IM} + K_{IT}$$

$K_{IC}$  = Allowable stress intensity factor at temperatures related to  $RT_{NDT}$

$K_{IM}$  = Stress intensity factor attributed to primary membrane stress

$K_{It}$  = Stress intensity factor attributed to thermal gradients

The above equation is applied to the reactor vessel beltline region.

For plant heatup, the thermal stress varies from compressive at the inner wall to tensile at the outer wall. Since the thermal tensile stress adds to the pressure-induced stress, transient heatup bounds steady-state heatup (no thermal stress) for an outside surface defect. For internal defects, thermally-induced compressive stresses tend to alleviate the tensile stresses induced by internal pressure. However,  $K_{IC}$  is governed by the temperature at the crack tip, which is lower than the RCS temperature used to control heatup and cooldown rates. Therefore,  $K_{IC}$  will be lower at the crack tip than at the vessel inner wall. These competing effects do not always have the same net effect throughout RCS heatup; therefore, both steady-state and transient heatup is considered.

For plant cooldown, the thermal stress is tensile on the interior vessel wall and compressive on the exterior wall. A 1/4T inside surface crack will have a higher  $K_{IC}$  during cooldown than at steady state for a given RCS temperature, but thermal stresses will be higher during cooldown. Similarly to the heatup condition, RCS temperature is used to control the cooldown rates, but the crack tip temperature is different from RCS temperature. Therefore, both steady state and transient cooldown conditions must be analyzed. Since the stress is highest on the interior vessel wall, which also receives the most irradiation and degradation in material properties, the exterior 1/4T crack is not considered for cooldown.

For steady-state, the maximum allowable pressure is determined by:

$$P = \frac{K_{IC}}{2M_m} \left( \frac{t}{R_i} \right)$$

where  $M_m$  is the membrane tension factor given by Appendix G-2214.1 of Section XI,  $t$  is the vessel wall thickness, and  $R_i$  is the vessel inner radius. For finite heatup or cooldown rates, the maximum allowable pressure is determined by:

$$P = \frac{K_{IC} - K_{IT}}{2M_m} \left( \frac{t}{R_i} \right)$$

The thermal stress intensity factor,  $K_{IT}$ , can be calculated using correlations for assumed temperature distributions in Appendix G of Section XI of the Code or by solving the conduction equation for the pressure vessel wall geometry to determine the temperature distribution as a function of space and time, finding the stress due to differential thermal expansion, and using the following equations:

$$\sigma(x) = C_0 + C_1 \left( \frac{x}{a} \right) + C_2 \left( \frac{x}{a} \right)^2 + C_3 \left( \frac{x}{a} \right)^3$$

Where  $C_0, C_1, C_2, C_3$  are determined from the stress distribution at a specified time,  $x$  is a dummy variable representing the distance from the interior or exterior surface of the vessel wall, and  $a$  is the depth of the crack.



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For a 1/4T inside surface defect:

$$K_{IT} = (1.0359C_0 + 0.6322C_1 + 0.4753C_2 + 0.3855C_3) * \sqrt{\pi a}$$

For a 1/4T outside surface defect:

$$K_{IT} = (1.043C_0 + 0.630C_1 + 0.481C_2 + 0.401C_3) * \sqrt{\pi a}$$

The heatup and cooldown curves are analyzed by varying the temperature at specified rates, determining the thermal stress intensity factor, calculating the allowable stress intensity factor based on ART, and determining the allowable operating pressure at each temperature.

b) System In-service Testing

The in-service testing curve is developed in the same manner as in a) above with the exception that a safety factor of 1.5 is allowed by ASME III in lieu of 2.

c) Vessel Flange Requirements

10CFR50 Appendix G Table 1 summarizes the minimum RCS temperatures allowed for hydrostatic testing and for normal operations with the core critical and not critical based on whether the vessel pressure is greater or less than 20% of the pre-service hydrostatic test pressure.

d) Lowest Service Temperature

The ASME Code Section III, NB-2332 (b) requires a lowest allowable service temperature of  $RT_{NDT} + 100^\circ\text{F}$  for piping, pumps, and valves. Under this lowest service temperature, the pressurizer pressure must be <20% of the pre-service hydrostatic test pressure.

e) Maximum Pressure for Shutdown Cooling

This pressure is established by considering the design pressure of the Shutdown Cooling System, shutoff head of the low pressure safety injection (LPSI) pumps, elevation head from the pressurizer to the LPSI pumps, and the design temperature of the Shutdown Cooling System

The pressure-temperature limitation curves are a combination of the most conservative portions of (a)-(e) above and are predicted for 40-year life. Instrumentation errors and hydrostatic head corrections are considered when implementing the curves. During plant life operation, the surveillance capsules (refer to Subsection 5.3.3.7) are removed from their locations in the reactor vessel for testing<sup>[6,9]</sup>. The data obtained will be compared to that used to develop the predicted limitation curves presented in the Technical Specifications. Curves are redrawn as necessary to reflect new test data or when the current curves expire.

←(LBDCR 17-002, R310)

5.3.2.2 Operating Procedures

→(DRN 06-911, R15, LBDCR 17-002, R310)

Pressure-temperature limitations and additional information are described in the Technical Specifications. The pressure-temperature limit curves provided in Section 3/4.4 have been prepared in accordance with Appendix G, ASME Code Section XI. Maintenance of reactor coolant system (RCS) pressure and temperature within these prescribed limits ensure that the integrity of the reactor coolant pressure boundary (RCPB) is maintained.

←(DRN 06-911, R15, LBDCR 17-002, R310)

5.3.2.3 Fracture Toughness for Pressurized Thermal Shock Events

→(DRN 00-1059, R11-A; 03-2059, R14)

An evaluation was performed of the Waterford Unit 3 reactor vessel beltline materials relative to the Pressurized Thermal Shock (PTS) screening criteria of 10CFR50.61<sup>[3]</sup> and the upper shelf screening criteria of 10 CFR Part 50, Appendix G<sup>[4]</sup>. The PTS values are calculated in accordance with 10CFR50.61<sup>[3]</sup>. The predicted upper shelf energy values are evaluated using the methods of Regulatory Guide 1.99, Revision 2<sup>[5]</sup>, for each beltline material. The calculation of the PTS and upper shelf energy values represents power uprate conditions, including a 1.5% uprate (3441 MWt) at the start of Cycle 12 and a 8% uprate (3716 MWt) at the start of Cycle 14.

The determination of the chemistry factor values per Position 1.1 and 2.1 of Reference 5 is detailed in Reference 6 and summarized in Table 5.3-14. (In this table, the “Chemistry Factor Basis” refers to values from Table 1 and 2 of 10CFR50.61<sup>[3]</sup> for weld and plate, respectively, and to “surveillance data” for the values derived in Reference 6.) The neutron fluence is the peak value for the corresponding plate and weld for 32 EFPY.  $RT_{PTS}$  was determined for each material in the beltline region is given by the following expression:

$$RT_{PTS} = \text{Initial } RT_{NDT} + \Delta RT_{PTS} + \text{Margin}$$

Initial  $RT_{NDT}$  is the reference temperature for the unirradiated material as defined in paragraph NB-2331 of Section III of the ASME Boiler and Pressure Vessel Code. Measured values of initial  $RT_{NDT}$  are available for each of the materials.  $\Delta RT_{PTS}$  is the mean value of the adjustment in reference temperature caused by irradiation and is calculated as follows:

$$\Delta RT_{PTS} = CF * f^{(0.28 - 0.10 \log f)}$$

where  $f$  is the vessel fluence at the clad-base metal interface given in units of  $10^{19}$  n/cm<sup>2</sup>. Margin is determined based on the uncertainty in initial  $RT_{NDT}$  and the uncertainty in the  $\Delta RT_{PTS}$  prediction. Margin is calculated as:

$$M = 2 \sqrt{\sigma_i^2 + \sigma_{\Delta}^2}$$

→(LBDCR 17-020, R310)

The initial  $RT_{NDT}$  values are based on measured values and, therefore,  $\sigma_i$  is equal to 0°F. The uncertainty in the  $\Delta RT_{PTS}$  prediction,  $\sigma_{\Delta}$ , is 28°F for welds and 17°F for plates. However, the value of  $2\sigma_{\Delta}$  does not have to exceed  $\Delta RT_{PTS}$ . (Note: The value of  $\sigma_{\Delta}$  for the two materials for which credible surveillance data are available does not have to exceed 14°F for welds and 8.5°F for base metal). The values of  $\sigma_i$ ,  $\sigma_{\Delta}$ , and the total margin are given for each material in Table 5.3-14. Total margin in these cases is  $2\sigma_{\Delta}$  or  $\Delta RT_{PTS}$ , whichever is smaller. Values of  $RT_{PTS}$  are given for each beltline material at 32 EFPY in Table 5.3-14. The highest value is 30.9°F for lower shell plate M-1004-2 at 32 EFPY. All the projected values for the Waterford Unit 3 reactor vessel beltline materials are well below the Pressurized Thermal Shock (PTS) screening criteria of 270°F for axial welds and plates, and 300°F for circumferential welds.

←(DRN 00-1059, R11-A; 03-2059, R14, LBDCR 17-020, R310)

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→(DRN 03-2059 R14, LBDCR 17-020, R310)

The predicted upper shelf energy values for each of the Waterford 3 Unit beltline material were evaluated in accordance with 10 CFR Part 50, Appendix G<sup>[4]</sup> using Position 1.2 of Regulatory Guide 1.99, Revision 2<sup>[5]</sup>. The predictions were based on the predicted fluence<sup>[6]</sup> at the vessel ¼T location at 32 EFPY. (Note: The peak beltline fluence at ¼T was used for all the beltline materials rather than taking credit for the relatively small variation in the axial fluence between the intermediate and lower shells.) The projected upper shelf energy values far exceed the 50 ft-lb screening criterion of 10CFR50, Appendix G<sup>[4]</sup> at 32 EFPY.

The plate and weld data from the surveillance capsule analyses<sup>[6]</sup> were also evaluated using Position 2.2 of Regulatory Guide 1.99, Revision 2<sup>[5]</sup>. The projections of upper shelf energy decrease were based on the upper bound of the measurements extended parallel to the lines in Figure 2 of Reference 5. The projected shelf decrease for plate M-1004-2 based on the measurements is 14% at 32 EFPY. The projected shelf decrease for weld 101-171 based on the measurements is 9.5% at 32 EFPY. The projected upper shelf energies are given below at 32 EFPY. The corresponding values based on Position 1.2 are also shown below. This comparison demonstrates the low radiation sensitivity of the Waterford Unit 3 beltline materials, and it adds confidence to the expectation that those materials will far exceed the 50 ft-lb screening criterion of 10CFR50, Appendix G<sup>[4]</sup> at 32 EFPY.

### Comparison of Upper Shelf Energy Decrease Predictions

Material	32 EFPY USE (ft-lbs) using Position 1.2	32 EFPY USE (ft-lbs) using Position 2.2
Plate M-1004-2	111	121
Weld 101-171	123	141

←(LBDCR 17-020, R310)

←(DRN 03-2059, R14)

### 5.3.3 REACTOR VESSEL INTEGRITY

→(EC-1020, R307)

C-E designed and fabricated the reactor vessel for Waterford 3 C-E has been involved in reactor vessel design and fabrication since the late 1950's, and this proven expertise is reflected in the Waterford 3 reactor vessel and the satisfactory performance of a large number of reactor vessels in operating plants. Westinghouse designed the Replacement Reactor Vessel Closure Head (RRVCH). The RRVCH was fabricated by Doosan.

←(EC-1020, R307)

Vessel integrity is ensured by the use of proven fabrication techniques and well characterized steels which exhibit uniform properties and consistent behavior. The characterization of these materials was established through industrial and governmental studies which examined the prefabrication material properties through to irradiated service operation. Inservice inspection and material surveillance programs are also conducted during the service life of the vessel, which further ensures that vessel integrity is maintained.

#### 5.3.3.1 Design

Applicable design codes are found in Table 5.2-1. A schematic of the reactor vessel is shown on Figure 5.3-11. Additional information can be found in Subsection 5.3.1.2.

### 5.3.3.2 Materials of Construction

The reactor vessel shell is fabricated from SA-533, Grade B, Class 1, material. This material has a minimum tensile strength of 80 ksi and a minimum yield strength of 50 ksi. This shell material responds well to quench and tempered heat treatment, which in combination with fine-grain melting practice produces high quality plate with excellent fracture toughness properties. The nozzles, also having excellent toughness properties, are fabricated from SA508, Class 2 forgings. The welding materials used include Mil Spec B-4 wire for submerged arc processes and E-8018C-3 material for manual arc processes. The stainless steel cladding utilized is nominal 19Cr-9Ni.

### 5.3.3.3 Fabrication Methods

→(EC-1020, R307)

The reactor vessel is constructed of formed plates welded into cylinders and hemispherical heads. The closure head, upper shell, and nozzles are forgings. This typifies construction of the reactor vessel in the preceding introductory material. No special fabrication methods were used in the reactor vessel fabrication. The basic design and fabrication of the reactor vessel are as follows.

The reactor vessel is a vertically mounted cylindrical vessel with a hemispherical lower head welded to the vessel and a removable hemispherical upper closure head. The pressure vessel is approximately 520 in. high (overheads) by 172 in. inside diameter and is all welded manganese molybdenum nickel steel plate and forging construction. Except for the Replacement Reactor Vessel Closure head which is clad with 3/16 in. minimum. Type 308L (and 309L used for base layer) stainless steel, the internal surfaces that are in contact with the reactor coolant are clad with 1/8 in. minimum Type 304 austenitic stainless steel and have a finish of 250 micro in. or better. The closure head flange and reactor vessel shell flange provide the structural rigidity necessary for bolting the head to the shell.

←(EC-1020, R307)

The reactor vessel fabrication is begun with an upper vessel assembly which consists of the upper shell, intermediate shell, nozzles, and reactor vessel shell flange. Both the upper and intermediate shells consist of three 120 degree segments formed from plate material and welded together to form cylindrical shells. Once the shells are welded, the upper shell is welded to the reactor vessel shell flange. The intermediate shell is then welded to form the upper vessel assembly. Four inlet nozzles and two outlet nozzles are then welded to complete the upper vessel assembly.

The lower vessel assembly consists of the lower shell and the bottom head. The lower shell is formed from three plates into 120 degree segments and welded together to form a cylindrical shell. The bottom head is constructed of six peel segments and a dome section, all formed from plate material. These are welded together to form a hemispherical head. The lower shell and bottom head are then welded together to complete the lower vessel assembly.

The closure head is fabricated separately. It is bolted to the reactor vessel only for hydrostatic testing.

→(EC-1020, R307)

The closure head assembly consists of the closure head forging, control element drive mechanism (CEDM) housings, and instrument nozzles. Penetrations are then machined in the closure head for 87 control rod mechanisms, 10 instrumentation nozzles, and one vent pipe. Attachment of these complete the closure head assembly. The closure head is attached to the reactor vessel by 54 seven in. diameter studs which are threaded into the vessel flange and extend through the closure head flange.

←(EC-1020, R307)

Previous experience using the above procedures in fabricating other reactor vessels is summarized in Subsection 5.3.3.

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### 5.3.3.4 Inspection Requirements

→(EC-1020, R307)

Inspection requirements of ASME Code, Section III, 1971 Edition including Summer 1971 Addenda and, for the replacement closure head, Section III, 1998 Edition through 2000 Addenda, are discussed in Subsection 5.3.1.3.

←(EC-1020, R307)

### 5.3.3.5 Shipment and Installation

→(EC-1020, R307)

The reactor vessel is shipped by barge to the site mounted on the shipping skid used for installation. The vessel is protected by closing all openings (including the top of the vessel) with metal shipping covers and pressurizing with inert gas. The replacement closure head is shipped on a separate skid. During shipment, the environment within the replacement closure head is maintained clean and dry, and is protected from external humidity and atmosphere by its shipping skid, shrink wrap and the use of desiccants. Vessel surfaces and covers are sprayed with a strippable coating for protection against corrosion during shipping and installation. Prior to the welding of inter-connecting piping and installation of insulation, the temporary protective coating is removed by peeling.

←(EC-1020, R307)

### 5.3.3.6 Operating Conditions

Operating parameters are provided in Subsection 4.4.3. Design transient information is supplied in Subsection 3.9.1.1.

### 5.3.3.7 In-service Surveillance

#### 5.3.3.7.1 Irradiated Materials Surveillance

This program is described in Subsection 5.3.1.6.

#### 5.3.3.7.2 In-service Inspection

This program is described in Subsection 5.2.4.

## SECTION 5.3: REFERENCES

1. Letter from A. E. Scherer (C-E) to Secretary of the Commission (NRC), LD-75-655, September 26, 1975.
2. "C-E Procedure for Design, Fabrication, Installation and Inspection of Surveillance Specimen Holder Assemblies," Combustion Engineering Topical Report, CENPD-155P, September 1974.
3. 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events," Federal Register, Volume 60, No. 243, dated December 19, 1995.
4. Code of Federal Regulations, 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements," U.S. Nuclear Regulatory Commission, Washington, D.C., Federal Register, Volume 60, No.243, dated December 19, 1995.
5. Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," U.S. Nuclear Regulatory Commission, May 1988.
6. WCAP-16002, Revision 0, "Analysis of Capsule 263 from the Entergy Operations Waterford Unit 3 Reactor Vessel Radiation Surveillance Program," March 2003.
7. WCAP-16088, Revision 1, "Waterford Unit 3 Reactor Vessel Heatup and Cooldown Limit Curves for Normal Operation," September 2003.

→(LBDCR 17-002, R310)

8. WF3-EP-16-00001, "WCAP-18002-NP, Waterford Unit 3 TLAA on Reactor Vessel Integrity," July, 2015.
9. WCAP-17969-NP, "Analysis of Capsule 83° from the Entergy Operations, Inc Waterford Unit 3 Reactor Vessel Radiation Surveillance Program," March 2015.

←(DRN 03-2059, R14, LBDCR 17-002, R310)

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TABLE 5.3-1 (Sheet 1 of 5)

DATA POINTS USED TO ESTABLISH C-E RT<sub>NDT</sub> SHIFT vs. FLUENCE AND PERCENT C  
UPPER SHELF DROP vs. (FLUENCE) 1/2 DESIGN CURVES FOR A533-B REACTOR VESSEL MATERIALS

Data Point	Lit Ref <sup>(h)</sup>	Type Mtl	Plate or Weld	Thickness (in.)	Cu (Wt%)	p (Wt%)	S (Wt%)	$\phi t$ (n/cm <sup>2</sup> x 10 <sup>19</sup> ) <sup>(a)</sup>	$(\phi t)^{1/2}$ (n/cm <sup>2</sup> ) <sup>1/2</sup> x 10 <sup>9</sup>	$\Delta RT_{NDT}$ <sup>(b)</sup> (°F)	C <sub>v</sub> Initial Upper Shelf (ft-lb)	C <sub>v</sub> Post-Irradiation Upper Shelf (ft-lb)	C <sub>v</sub> Upper Shelf Drop (ft-lb)	Percent C <sub>v</sub> Upper Shelf Drop (%)
32	1	A533-B1	Plate	4	0.14	0.009	0.022	2.3	4.79	120	100	77	23	23
33	1	A533-B1	Plate	8	0.14	0.010	0.023	2.3	4.79	95	116	98	18	15.5
34	1	A533-B1	Plate	8-1/8	0.19	0.010	0.017	1.7	4.12	190	101	86	15	14.8
35	1	A533-B1	Plate	6-3/8	0.09	0.008	0.015	0.2	1.41	0	137	137	0	0
36	1	A533-B1	Plate	6-3/8	0.09	0.008	0.015	2.0	4.47	80	137	130	7	5.1
37	1	A533-B1	Plate	6-3/8	0.09	0.008	0.015	2.0	4.47	90	137	75	-	-
38	1	A533-B2	Plate	6-3/8	0.09	0.008	0.015	0.5	2.23	35	120	-	-	-
39	1	A533-B2	Plate	6-3/8	0.09	0.008	0.015	2.0	4.47	75	120	70	50	-
40	1	A533-B1	Plate	7-1/2	0.12	0.008	0.015	1.7	4.12	70	136	100	36	-
41	1	A533-B1	Plate	7-1/2	0.11	0.008	0.019	1.7	4.12	85	126	121	5	3.9
42	1	A533-B1	Plate	5-3/4	0.12	0.008	0.018	1.8	4.24	50	148	115	33	22.3
43	2	A533-B1	Plate	8	0.09	0.008	0.014	0.5	2.23	0	135	135	0	0
44	2	A533-B1	Plate	8	0.09	0.008	0.014	2.4	4.89	85	135	125	10	7.4
45	3	A533-B1	Plate	6-1/4	0.09	0.003	0.014	2.5	5.0	60	123	128	-	-
48	1	A533-B1	Weld-S/A <sup>(d)</sup>	7-1/2	0.22	0.015	0.011	1.7	4.1	200	109	63	46	42.2
49	1	A533-B1	Weld-E/S <sup>(e)</sup>	5-3/4	0.19	0.008	0.014	1.6	4.0	165	82	79	3	3.6

- a. Fluence: Neutron energies > 1 Mev, irradiation temperature - 550 °F
- b. Based on NDTT measured at the C<sub>v</sub> 30 ft-lb level
- c. The weld from which these specimens were taken (S/A) was back chipped and rewelded probably with a manual arc. The specimens were taken from different areas of the weld so the chemistries of the specimen could tend to vary greatly.
- d. S/A = Submerged arc
- e. E/S = Electroslag
- f. Percent C<sub>v</sub>, upper shelf drop as reported in reference
- g. Irradiation temperature - 530 °F
- h. Refer to Table 5.3-2

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TABLE 5.3-1 (Sheet 2 of 5)

DATA POINTS USED TO ESTABLISH C-E RT<sub>NDT</sub>-SHIFT vs. FLUENCE AND PERCENT C<sub>v</sub> UPPER SHELF DROP vs. (FLUENCE) 1/2 DESIGN-CURVES FOR A533-B REACTOR VESSEL MATERIALS

Data Point	Lit Ref <sup>(h)</sup>	Type Mtl	Plate or Weld	Thickness (in.)	Cu (Wt%)	p (Wt%)	S (Wt%)	$\phi t$ (n/cm <sup>2</sup> x 10 <sup>19</sup> ) <sup>(a)</sup>	$(\phi t)^{1/2}$ (n/cm <sup>2</sup> ) <sup>1/2</sup> x 10 <sup>9</sup>	$\Delta RT_{NDT}$ <sup>(b)</sup> (°F)	C <sub>v</sub> Initial Upper Shelf (ft-lb)	C <sub>v</sub> Post-Irradiation Upper Shelf (ft-lb)	C <sub>v</sub> Upper Shelf Drop (ft-lb)	Percent C <sub>v</sub> Upper Shelf Drop (%)
50 <sup>(c)</sup>	2	A533-B1	Weld-S/A <sup>(d)</sup>	8	0.09	0.010	0.014	0.5	2.23	0	145	145	0	0
51 <sup>(c)</sup>	2	A533-B1	Weld-S/A <sup>(d)</sup>	8	0.09	0.010	0.014	2.4	4.89	90	145	125	20	13.8
52 <sup>(c)</sup>	2	A533-B1	Weld-S/A <sup>(d)</sup>	8	0.14	0.010	0.014	0.5	2.2	105	105	70	35	33.3
53 <sup>(c)</sup>	2	A533-B1	Weld-S/A <sup>(d)</sup>	8	0.14	0.010	0.014	2.4	4.8	210	105	65	40	38
54	3	A533-B1	Weld-E/S <sup>(e)</sup>	6-1/4	0.09	0.002	0.012	2.5	5.0	100	88	78	10	11.3
58	4	A533-B1	Plate (surf.)	12	0.18	0.008	0.008	1.0	-	101	-	-	-	-
59	4	A533-B1	Plate (1/2 T)	12	0.18	0.008	0.008	1.0	-	126	-	-	-	-
60	4	A533-B1	Plate (3/8 T)	12	0.18	0.008	0.008	1.0	-	70	-	-	-	-
61	5	A533-B1	Plate	12	0.14	0.008	0.016	4.52-5.59	7.0	80	120	100	20	16.6
62	5	A533-B1	Plate	12	0.14	0.008	0.016	3.64-4.24	6.32	135	>120	~108	-	-
63	5	A533-B1	Plate	12	0.14	0.008	0.016	1.18-1.33	3.6	85	>120	~110	-	-
64	5	A533-B1	Weld-(S/A) <sup>(d)</sup>	11- 3/4	0.22	0.019	0.13	2.73-4.25	5.91	256	115	~60	55	47.8

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TABLE 5.3-1 (Sheet 3 of 5)

DATA POINTS USED TO ESTABLISH C-E RT<sub>NDT</sub> SHIFT vs. FLUENCE AND PERCENT C<sub>v</sub>

UPPER SHELF DROP vs. (FLUENCE) 1/2 DESIGN CURVES FOR A533-B REACTOR VESSEL-MATERIALS

Data Point	Lit Ref <sup>(h)</sup>	Type Mtl	Plate or Weld	Thickness (in.)	Cu (Wt%)	p (Wt%)	S (Wt%)	$\phi t$ (n/cm <sup>2</sup> x 10 <sup>19</sup> ) <sup>(a)</sup>	$(\phi t)^{1/2}$ (n/cm <sup>2</sup> ) <sup>1/2</sup> x 10 <sup>9</sup>	$\Delta RT_{NDT}$ (°F) <sup>(b)</sup>	C <sub>v</sub> Initial Upper Shelf (ft-lb)	C <sub>v</sub> Post-Irradiation Upper Shelf (ft-lb)	C <sub>v</sub> Upper Shelf Drop (ft-lb)	Percent C <sub>v</sub> Upper Shelf Drop (%)
65	6	A533-B1	Plate	6	0.03	0.008	0.008	2.8	-	65	-	-	-	-
66	6	A533-B1	Plate	6	0.03	0.008	0.008	2.8	-	40	-	-	-	-
67	7	A533-B1	Plate	12	0.18	0.008	0.008	0.47	-	70	-	-	-	-
68	7	A533-B1	Plate	12	0.18	0.008	0.008	0.94	3.06	95	104	100	4	3.8
69	7	A533-B1	Plate	12	0.18	0.008	0.008	1.05	-	130	-	-	-	-
70	8	A533-B1	Weld-S/A <sup>(d)</sup>	12	0.23	0.011	0.008	2.5	5.0	270	125	70	55	44
71	8	A533-B1	Plate	12	0.18	0.008	0.008	2.8	5.29	200	104	73	31	29.8
72	9	A533-B1	Plate	6	0.13	0.008	0.008	2.8	5.29	125	135	100	35	25.9
73	9	A533-B1	Plate	6	0.13	0.008	0.007	2.8	5.29	140	110	90	20	13.18
74	9	A533-B1	Plate	6	0.03	0.008	0.008	3.1	5.56	70	145	138	7	4.8
75	10	A533-B1	Plate (3/8 T)	12	0.14	0.008	0.016	0.5	-	50	-	-	-	-
77	8	A533-B1	Plate	12	0.14	0.008	0.016	2.7	5.19	170	122	~102	20	14 <sup>(f)</sup>
78	8	A533-B1	Plate	12	0.14	0.008	0.016	2.6	5.09	165	99	~85	14	14 <sup>(f)</sup>
79	11	A533-B1	Plate	8-10	0.17	0.009	0.015	2.1	4.58	145	115	93	22	19.1
80	11	A533-B1	Plate	8-10	0.24	0.008	0.011	3.7	6.08	165	110	84	26	23.6
81	11	A533-B1	Weld-(S/A) <sup>(d)</sup>	8-10	0.36	0.015	0.012	3.4	5.83	315	107	56	51	47.6



WSES-FSAR-UNIT-3

TABLE 5.3-1 (Sheet 4 of 5)

DATA POINTS USED TO ESTABLISH C-E RT<sub>NDT</sub> SHIFT vs. FLUENCE AND PERCENT C<sub>v</sub>

UPPER-SHELF DROP vs. (FLUENCE) 1/2 DESIGN-CURVES FOR A533-B REACTOR VESSEL MATERIALS

Data Point	Lit Ref <sup>(h)</sup>	Type Mtl	Plate or Weld	Thickness (in.)	Cu (Wt%)	p (Wt%)	S (Wt%)	$\phi t$ (n/cm <sup>2</sup> x 10 <sup>19</sup> ) <sup>(a)</sup>	$(\phi t)^{1/2}$ (n/cm <sup>2</sup> ) <sup>1/2</sup> x 10 <sup>9</sup>	$\Delta RT_{NDT}$ <sup>(b)</sup> (°F)	C <sub>v</sub> Initial Upper Shelf (ft-lb)	C <sub>v</sub> Post-Irradiation Upper Shelf (ft-lb)	C <sub>v</sub> Upper Shelf Drop (ft-lb)	Percent C <sub>v</sub> Upper Shelf Drop (%)
82	11	A533-B1	Weld S/A <sup>(d)</sup>	8-10	0.20	0.016	0.013	3.4	5.83	95	129	98	31	24.03
83	11	A533-B1	Plate	8-10	0.17	0.009	0.015	6.7	8.18	210	115	94	21	18.2
84	11	A533-B1	Plate	8-10	0.24	0.008	0.011	6.1	7.8	185	110	86	24	21.8
85	11	A533-B1	Weld-S/A <sup>(d)</sup>	8-10	0.36	0.015	0.012	6.1	7.8	350	107	56	51	47.6
86	11	A533-B1	Weld-S/A <sup>(d)</sup>	8-10	0.20	0.016	0.013	6.1	7.8	125	129	98	31	24.03
87	11	A533-B1	Plate	8-10	0.09	0.009	0.017	4.4	6.63	35	104	107	-	-
88	11	A533-B1	Plate	8-10	0.09	0.009	0.017	5.7	7.54	55	104	111	-	-
89	11	A533-B1	Plate	8-10	0.09	0.011	0.018	4.0	6.32	45	119	130	-	-
90	11	A533-B1	Plate	8-10	0.09	0.011	0.018	5.4	7.34	85	119	130	-	-
91	11	A533-B1	Weld-S/A <sup>(d)</sup>	8-10	0.07	0.010	0.010	4.9	7.0	35	157	155	2	1.27
92	11	A533-B1	Weld-S/A <sup>(d)</sup>	8-10	0.07	0.010	0.010	5.0	7.07	50	157	155	2	1.27
93	11	A533-B1	Weld-S/A <sup>(d)</sup>	8-10	0.05	0.004	0.004	4.9	7.0	≤20	144	147	-	-
94	12	A533-B1 (same material as Pt. 74)	Plate	6	0.03	0.008	0.008	15.8 <sup>(g)</sup>	12.56	260	145	99	46	31.7

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TABLE 5.3-1 (Sheet 5 of 5)  
 DATA POINTS USED TO ESTABLISH C-E RT<sub>NDT</sub> SHIFT vs. FLUENCE AND PERCENT C<sub>v</sub>  
 UPPER SHELF DROP vs. (FLUENCE) 1/2 DESIGN CURVES FOR A533-B REACTOR VESSEL MATERIALS

Data Point	Lit Ref <sup>(h)</sup>	Type Mtl	Plate or Weld	Thickness (in.)	Cu (Wt%)	p (Wt%)	S (Wt%)	$\phi t$ (n/cm <sup>2</sup> x 10 <sup>19</sup> ) <sup>(a)</sup>	$(\phi t)^{1/2}$ (n/cm <sup>2</sup> ) <sup>1/2</sup> x 10 <sup>9</sup>	$\Delta RT_{NDT}^{(b)}$ (°F)	C <sub>v</sub> Initial Upper Shelf (ft-lb)	C <sub>v</sub> Post-Irradiation Upper Shelf (ft-lb)	C <sub>v</sub> Upper Shelf Drop (ft-lb)	Percent C <sub>v</sub> Upper Shelf Drop (%)
95	12	A533-B1 (same material as Pt. 74)	Plate	6	0.03	0.008	0.008	24.1(g)	15.52	335	145	70	75	51.7
96	12	A533-B1 (same material as Pt. 74)	Plate	6	0.03	0.008	0.008	27.8	16.67	295	145	99	46	31.7
97	13	A533-B1 (same material as Pt. 73)	Plate	6	0.13	0.008	0.007	0.095	-	0	-	-	-	-
98	13	A533-B1 (same material as Pt. 85)	Weld	8-10	0.36	0.015	0.018	0.095	-	55	-	-	-	-
A	14	A302-B	Weld	10-1/2	0.22	-	-	0.2	-	95	-	-	-	-
B	15	A302-B	Weld	6	0.27	0.014	0.012	0.7	-	140	-	-	-	-
C	16	A508-2	Weld	6-1/2	0.23	0.012	0.016	0.49	-	140	-	-	-	-
D	1	A533-2	Weld	4	0.27	0.016	0.015	1.4	-	205	-	-	-	-

TABLE 5.3-2 (Sheet 1 of 2)

LITERATURE REFERENCES FOR TABLE 5.3-1

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TABLE 5.3-2 (Sheet 2 of 2)

LITERATURE REFERENCES FOR TABLE 5.3-1

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13. Naval Research Laboratory, unpublished data.
14. Ireland D. R., and Scotti, V. G., "Final Report on Examination and Evaluation of Capsule A for the Connecticut Yankee Reactor Pressure Vessel Surveillance Program," Battelle (Columbus) Memorial Institute, Docket No. 50-213, 1970.
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WSES-FSAR-UNIT-3

TABLE 5.3-3

Revision 310 (12/17)

SUMMARY OF INITIAL SURVEILLANCE MATERIALS TESTING<sup>(8,9)</sup>

→(LBDCR 17-020, R310)

Material and Code	C <sub>V</sub> Upper Shelf (ft-lb)	30 ft-lb Fit (a) (°F)	50 ft-lb Fit (a) (°F)	35 Mills Lat. Exp. Fit (a) (°F)	NDTT (°F)	RT <sub>NDT</sub> (°F)	RT Yield Strength (ksi) Static Dynamic	
Base Metal Plate M-1004-2 (WR)	141	-24.5	2.8	-6.7	-20	-20	69	97
Base Metal Plate M-1004-2 (RW)	170	-13.5	11.7	5.3	0	0 <sup>(b)</sup>	70	103
Weld Metal M-1004-1/M-1004-3	156	-84.4	-65	-68.2	-80	-80	85	113
HAZ Metal M-1004-2	170	-117	-90.1	-89.7	-50	-50	70	113
SRM HSST Plate O1MY-RW	130	28	70	48	0	--	--	--

(a) Determined from lower bound curve.

(b) Per ASME Sec. III NB-2331, RT<sub>NDT</sub> is developed using transverse direction Charpy specimens for plate. However, for conservatism, the RT<sub>NDT</sub> of M-1004-2 includes longitudinal drop-weight test data, so the RT<sub>NDT</sub> is 0°F<sup>(8)</sup> when used in reactor vessel material property predictions.

←(LBDCR 17-020, R310)

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TABLE 5.3-4

Revision 15 (03/07)

TOTAL QUANTITY OF SPECIMENS

Type of Specimen	Orientation	Base Metal	Weld Metal	HAZ	SRM <sup>(a)</sup>	Totals
Drop Weight →(DRN 06-897, R15) ←(DRN 06-897, R15)	Longitudinal	12	—	—	—	12
	Transverse	12	12	12	—	36
Charpy Impact →(DRN 06-897, R15) ←(DRN 06-897, R15)	Longitudinal	78	—	—	39	117
	Transverse	102	102	102	—	306
Tensile →(DRN 06-897, R15) ←(DRN 06-897, R15)	Longitudinal	18	—	—	—	18
	Transverse	36	36	36	—	108
→(DRN 06-897, R15) Totals ←(DRN 06-897, R15)		258	150	150	39	597

(a) Standard Reference Material

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TABLE 5.3-5

Revision 15 (03/07)

TYPE AND QUANTITY OF-SPECIMENS FOR UNIRRADIATED TESTS

Type of Specimen	Orientation	Quantity of Specimens				Totals
		Base Metal	Weld Metal	HAZ	SRM <sup>(a)</sup>	
Drop Weight →(DRN 06-897, R15) ←(DRN 06-897, R15)	Longitudinal	12	---	---	---	12
	Transverse	12	12	12	---	36
Charpy Impact →(DRN 06-897, R15) ←(DRN 06-897, R15)	Longitudinal	30	---	---	15	45
	Transverse	30	30	30	---	90
Tensile →(DRN 06-897, R15) ←(DRN 06-897, R15)	Longitudinal	18	---	---	---	18
	Transverse	18	18	18	---	54
→(DRN 06-897, R15) Totals ←(DRN 06-897, R15)		120	60	60	15	255

(a) Standard Reference Material characterized by HSST Program, specimens provided only for correlation with characterization tests.

TYPE AND QUANTITY OF SPECIMENS FOR  
IRRADIATION EXPOSURE AND IRRADIATED TESTS

Type of Specimen	Orientation	Quantity of Specimens				Totals
		Base Metal	Weld Metal	HAZ	SRM <sup>(a)</sup>	
→ <sup>(DRN 06-897, R15)</sup> Charpy Impact	Longitudinal	48	--	24	--	72
← <sup>(DRN 06-897, R15)</sup>	Transverse	72	72	72	--	216
Tensile	Longitudinal	--	--	--	--	
→ <sup>(DRN 06-897, R15)</sup> ← <sup>(DRN 06-897, R15)</sup>	Transverse	18	18	18	--	54
→ <sup>(DRN 06-897, R15)</sup> Totals		138	90	90	24	342
← <sup>(DRN 06-897, R15)</sup>						

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a. Standard Reference Material



## WSES-FSAR-UNIT-3

TABLE 5.3-7

CANDIDATE MATERIALS FOR NEUTRON THRESHOLD DETECTORS

Material	Reaction	Threshold Energy (MeV)	Half-Life
Uranium	$U^{238} (n, f) Sr^{90}$	0.7	28 years
Sulfur	$S^{32} (n, p) P^{32}$	2.9	14.3 days
Iron	$Fe^{54} (n, p) Mn^{54}$	4.0	314 days
Nickel	$Ni^{58} (n, p) Co^{58}$	5.0	71 days
Copper	$Cu^{63} (n, \alpha) Co^{60}$	7.0	5.3 years
Titanium	$Ti^{46} (n, p) Sc^{46}$	8.0	84 days
Cobalt	$Co^{59} (n, \gamma) Co^{60}$	Thermal	5.3 years

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TABLE 5.3-8

COMPOSITION AND MELTING POINTS OF  
CANDIDATE MATERIALS FOR TEMPERATURE MONITORS

Composition (wt%)	Melting Temperature (°F)
80 Au, 20 Sn	536
90.0 Pb, 5.0 Sn, 5.0 Ag	558
97.5 Pb, 2.5 Ag	580
97.5 Pb, 0.75 Sn, 1.75 Ag	590

WSES FSAR UNIT 3

TABLE 5.3 9

Revision 13 A (09/04)

TYPE AND QUANTITY OF SPECIMENS CONTAINED IN EACH  
IRRADIATION CAPSULE ASSEMBLY

→(DRN 04-1049, R13-A)

Capsule Location	Base Metal			Weld Metal		HAZ		Reference	Total Specimens	
	L <sup>(b)</sup>	T <sup>(c)</sup>	Tensile	Impact	Tensile	Impact	Tensile	Impact <sup>(a)</sup>	Impact	Tensile
Vessel 97°	12	12	3	12	3	12	3		48	9
Vessel 104°		12	3	12	3	12	3	12	48	9
Vessel 284°	12	12	3	12	3	12	3		48	9
Vessel 263°		12	3	12	3	12	3	12	48	9
Vessel 277°	12	12	3	12	3	12	3		48	9
Vessel 83°	12	12	3	12	3	12	3		48	9
Totals	48	72	18	72	18	72	18	24	288	54

←(DRN 04-1049, R13-A)

- a. Reference material correlation monitors
- b. L = Longitudinal
- c. T = Transverse

WSES-FSAR-UNIT-3

TABLE 5.3-10

Revision 14 (12/05)

CAPSULE ASSEMBLY REMOVAL SCHEDULE

→(DRN 03-2059, R14)

<u>Capsule No./ID</u>	<u>Azimuthal Location (deg)</u>	<u>Lead Factor</u>	<u>Removal Time (EFPY)*</u>	<u>Target Fluence (n/cm<sup>2</sup>)</u>
1/W-83	83	1.18	26	$2.47 \times 10^{19}$
2/W-97	97	1.18	4.44**	$6.47 \times 10^{18**}$
3/W-104	104	0.83	Standby	--
4/W-263	263	1.18	13.83**	$1.45** \times 10^{19**}$
5/W-277	277	1.18	Standby	--
6/W-284	284	0.83	Standby	--

←(DRN 03-2059, R14)

\*EFPY - Effective Full Power Years, withdrawal time may be modified to coincide with those refueling outages or plant shutdowns most closely approaching the withdrawal schedule.

\*\* - Values represent actual data on removed capsule

→(DRN 04-1049, R13-A)

NOTE: As required by 10CFR Appendix H, Section III.B.3, submit a proposed withdrawal schedule with technical

justification as specified in 10CFR50.4 for NRC approval prior to implementation.

←(DRN 04-1049, R13-A)

WSES-FSAR-UNIT-3

TABLE 5.3-11

WATERFORD UNIT 3 REACTOR VESSEL CLOSURE STUDS DATA

Piece Number	Drawing Number	Code Number	Heat No.	Material	Ultimate Tensile Strength		Test Temp (°F)	Fracture Toughness Charpy Energy (ft-lbs.)	Mils Lateral Expansion
					Test Temp (°F)	Strength (KSI)			
98	E74170-161-03	M-1028-1	80751	SA-540 Grade B24	70	168	10	49-48-47	28-27-25
98-1	E74170-161-03	M-1028-1	80751	SA-540 Grade B24	70	162.5	10	50-51-51	25-28-29
69	E74170-161-03	M-1028-1	80751	SA-540 Grade B24	70	163	10	50-48-49	26-25-29
69-1	E74170-161-03	M-1028-1	80751	SA-540 Grade B24	70	157	10	57-56-57	38-35-37
70	E74170-161-03	M-1028-1	80751	SA-540 Grade B24	70	156	10	56-54-53	30-34-34
70-1	E74170-161-03	M-1028-1	80751	SA-540 Grade B24	70	164	10	50-50-48	34-29-31
72	E74170-161-03	M-1028-1	80751	SA-540 Grade B24	70	158	10	56-56-55	38-34-32
72-1	E74170-161-03	M-1028-1	80751	SA-540 Grade B24	70	159	10	50-50-51	26-25-28
74	E74170-161-03	M-1028-1	80751	SA-540 Grade B24	70	157	10	50-52-50	31-32-27
74-1	E74170-161-03	M-1028-1	80751	SA-540 Grade B24	70	158	10	50-51-50	25-25-30
76	E74170-161-03	M-1028-1	80751	SA-540 Grade B24	70	154	10	54-54-53	27-31-33
76-1	E74170-161-03	M-1028-1	80751	SA-540 Grade B24	70	161	10	50-50-51	25-30-29

W3SES-FSAR-UNIT-3

TABLE 5.3-12

WATERFORD UNIT 3 REACTOR VESSEL NUTS AND WASHERS DATA

<u>Piece Number</u>	<u>Drawing Number</u>	<u>Code Number</u>	<u>Heat No.</u>	<u>Material</u>	<u>Tensile Strength</u>		<u>Fracture Toughness</u>		
					<u>Test Temp. (F)</u>	<u>Strength (KSI)</u>	<u>Test Temp. (F)</u>	<u>Charpy Energy (Ft-lbs)</u>	<u>Mils Lateral Expansion -</u>
41	E74170-161-03	M-1029-1	18551	SA-540 Grade B23	70	163.5	10	38-40-38	19-21-18
41-1	E74170-161-03	M-1029-1	18551	SA-540 Grade B23	70	164.5	10	42-40-38	20-22-18
48	E74170-161-03	M-1029-1	18551	SA-540 Grade B23	70	170.0	10	37-39-38	18-19-21
48-1	E74170-161-03	M-1029-1	18551	SA-540 Grade B23	70	165.0	10	43-45-42	25-27-24

**WSES-FSAR-UNIT-3**

TABLE 5.3-13 Revision 310 (12/17)

WATERFORD UNIT 3  
REACTOR VESSEL MATERIALS

→(LBDCR 17-002, R310)

Product Form	Material Identification	Drop Weight NDTT ( F)	Initial RT <sub>NDT</sub> ( F)	Chemical Content %		Phosphorus
				Nickel	Copper	
Plate	M-1003-1	-30	-25.1	0.71	0.02	0.004
Plate	M-1003-2	-40	-20	0.67	0.02	0.006
Plate	M-1003-3	-30	-20	0.70	0.02	0.007
Plate	M-1004-1	-40	-37.6	0.62	0.03	0.006
Plate	M-1004-2	0	0	0.58	0.03	0.005
Plate	M-1004-3	-20	-20	0.62	0.03	0.007
Weld	101-124 A,B,& C <sup>a</sup>	-60	-60	0.96	0.02	0.010
Weld	101-142 A,B,& C <sup>b</sup>	-80	-80	< 0.20	0.03	0.007
Weld	101-171 c	-70 to-80	-70	0.16	0.05	0.008
Plate	M-1002-1	-20	-15.4	0.64	0.13	0.006
Plate	M-1002-2	-20	-1.4	0.61	0.13	0.006
Plate	M-1002-3	-20	-20	0.65	0.10	0.006
Weld	101-122 A,B, & C <sup>(d)</sup>					
	Heat #606L40	---(f)	-56 <sup>(f)</sup>	0.27	0.23	0.013
	Heat FOCA	---(g)	-30 <sup>(g)</sup>	0.98	0.02	0.008
	Heat JBHA	-40	-40	0.97	0.03	0.008
	Heat HODA	-60 <sup>(h)</sup>	-60 <sup>(h)</sup>	0.96	0.02	0.009
Weld	106-121 <sup>(e)</sup>					
	Heat #P4767	---(f)	-56 <sup>(f)</sup>	1.00	0.16	0.016
	Heat IAGA	-30	-30	0.98	0.03	0.009
	Heat KOHA	---(g)	-30 <sup>(g)</sup>	0.93	0.03	0.007

a. Intermediate shell course longitudinal seam weld

b. Lower shell course longitudinal seam weld

c. Intermediate - lower shell girth weld

d. Upper shell longitudinal seam weld

e. Upper-intermediate girth weld

f. No drop weight data available. RT<sub>NDT</sub> is standard value for Linde type 0091 flux in 10 CFR 50.61<sup>(3)</sup>

g. No drop weight data available. RT<sub>NDT</sub> is assigned the maximum value of SMAW data, which is -30°F.

h. NDTT and RT<sub>NDT</sub> determined from sister weld data, WMC source document SIS-0139216447

←(LBDCR 17-002, R310)

W3SES-FSAR-UNIT-3

TABLE 5.3-14

Revision 310 (12/17)

CALCULATION OF THE WATERFORD UNIT 3 RT<sub>PTS</sub> VALUES FOR 32 EFPY

Material	Chemistry Factor Basis	CF (°F)	Fluence (x10 <sup>19</sup> n/cm <sup>2</sup> )	RT <sub>NDT</sub> <sup>(a)</sup> (°F)	ΔRT <sub>PTS</sub> <sup>(b)</sup> (°F)	σ <sub>i</sub> (°F)	σ <sub>Δ</sub> (°F)	Margin (°F)	RT <sub>PTS</sub> <sup>(c)</sup>
Intermediate Shell Plate M-1003-1	Table 2	20	2.48	-25.1	24.9	0	12.4	24.9	24.7
Intermediate Shell Plate M-1003-2	Table 2	20	2.48	-20	24.9	0	12.4	24.9	29.8
Intermediate Shell Plate M-1003-3	Table 2	20	2.48	-20	24.9	0	12.4	24.9	29.8
Lower Shell Plate M-1004-1	Table 2	20	2.48	-37.6	24.9	0	12.4	24.9	12.2
Lower Shell Plate M-1004-2	Surveillance Data	12.4	2.48	0	15.4	0	7.7	15.4	30.9
Lower Shell Plate M-1004-3	Table 2	20	2.48	-20	24.9	0	12.4	24.9	29.8
Intermediate Shell Longitudinal Weld Seams 101-124 A,B,C	Table 1	27	2.48	-60	33.6	0	16.8	33.6	7.2
Lower Shell Longitudinal Weld Seams 101-142 A,B,C	Table 1	35	2.48	-80	43.5	0	21.8	43.5	7.1
Intermediate to Lower Shell Girth Weld Seam 101-171	Surveillance Data	16.2	2.48	-70	20.2	0	10.1	20.2	-29.7

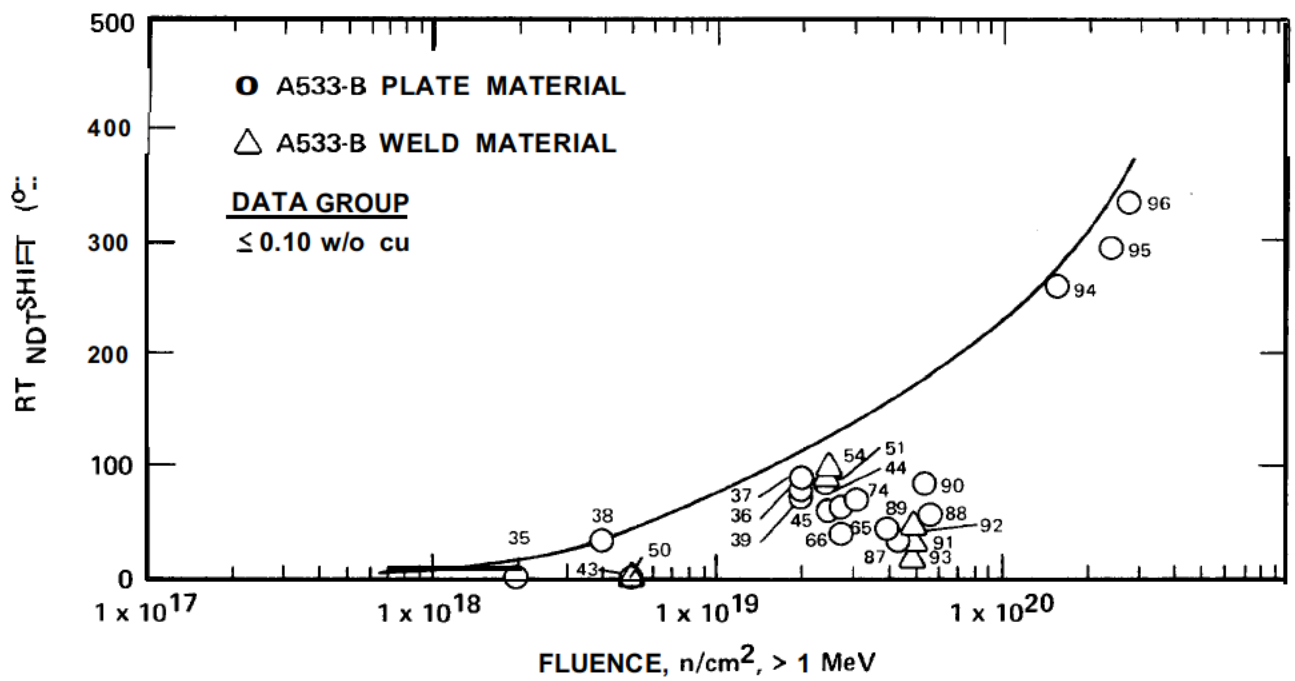
Notes:

(a) Initial reference temperature (RT<sub>NDT</sub>) values are measured. Thus, σ<sub>i</sub> equal to 0°F.

(b) ΔRT<sub>PTS</sub> = CF \* FF

(c) RT<sub>PTS</sub> = Initial RT<sub>NDT</sub> + ΔRT<sub>PTS</sub> + Margin (°F)

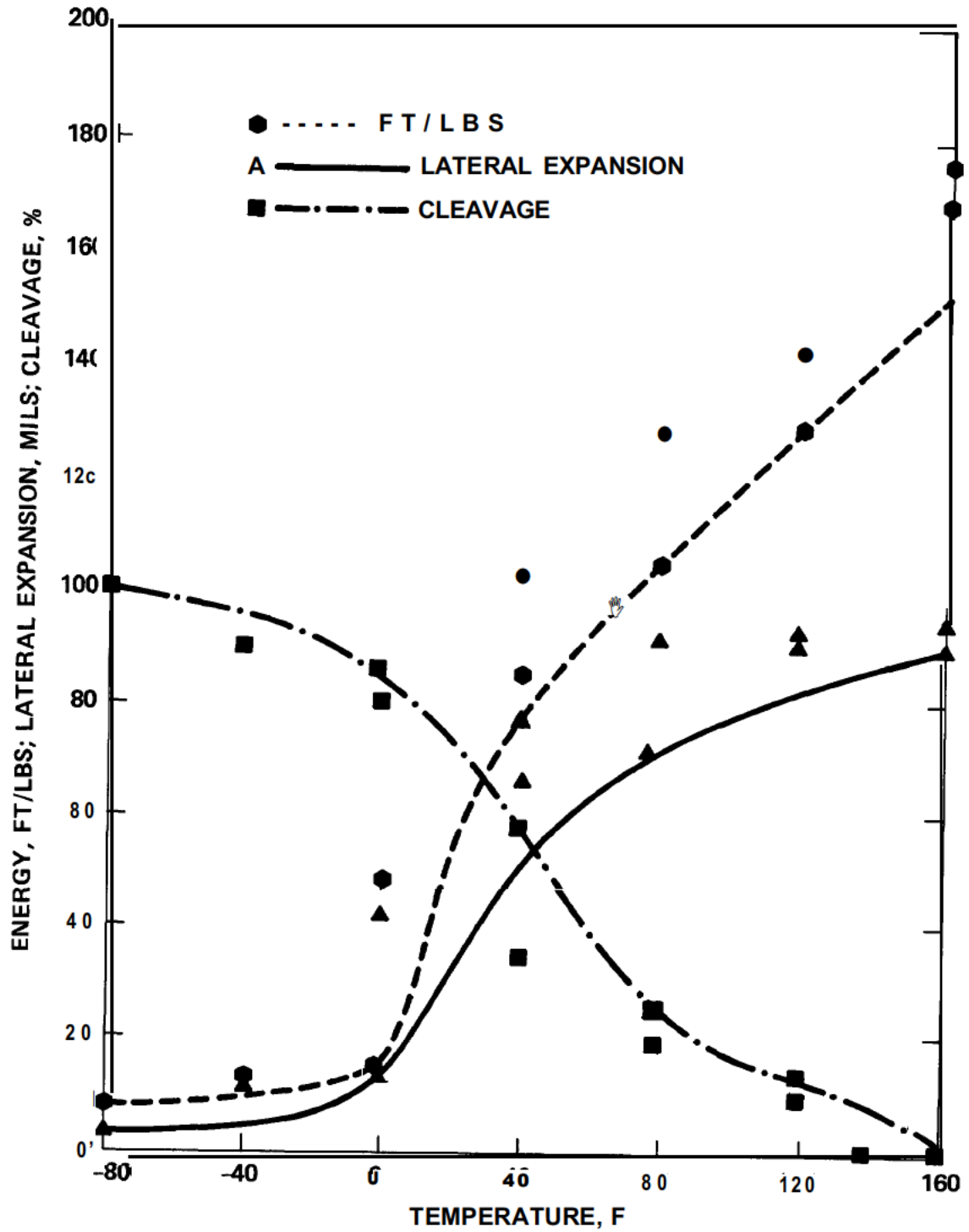




LOUISIANA  
 POWER & LIGHT CO.  
 Waterford Steam  
 Electric Station

C-ERTNDT SHIFT CURVE A533-B PLATE AND WELD  
 MATERIAL DATA GROUP (≤ 0.10 WT% CU)

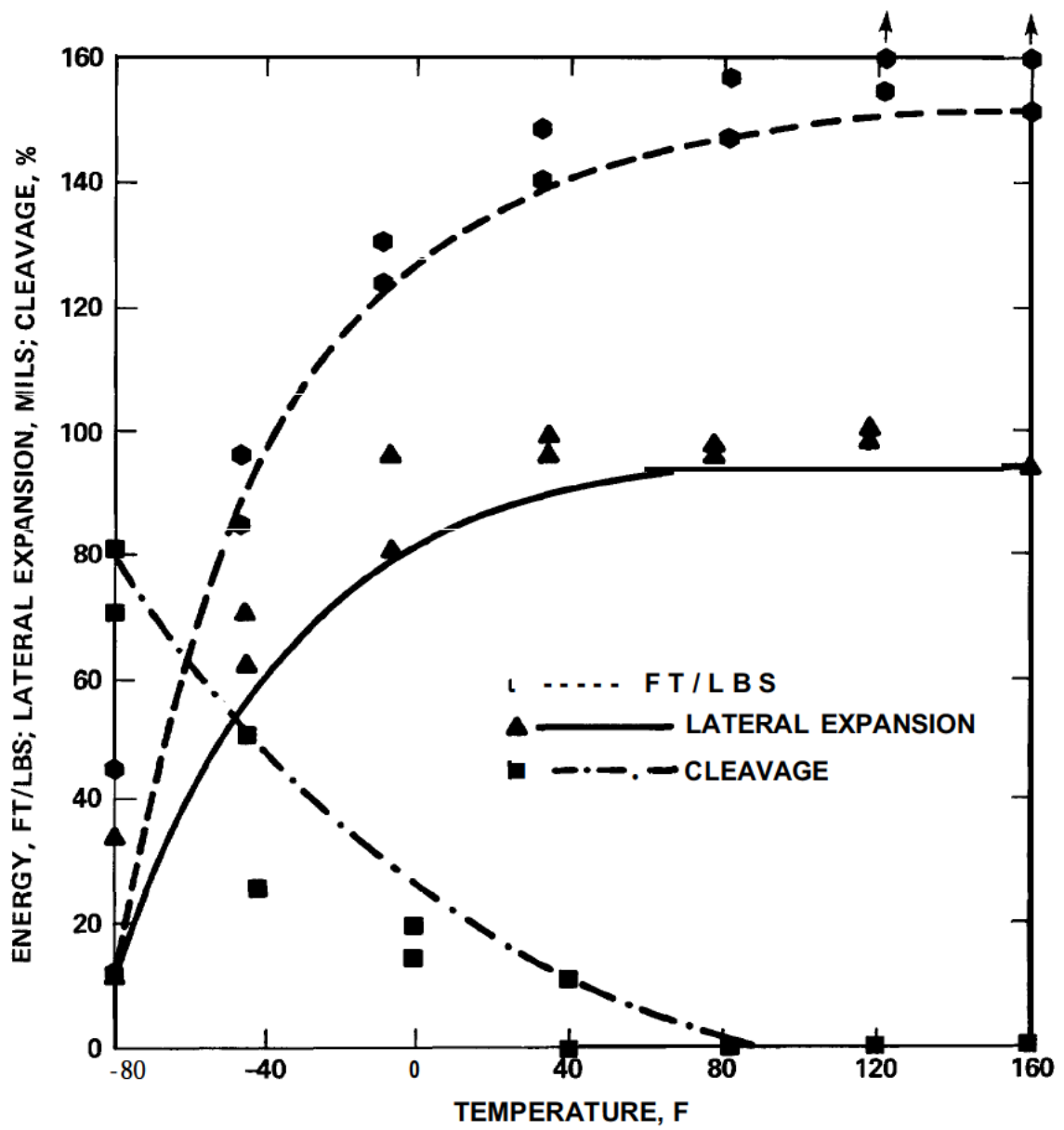
Figure  
 5.3-1



LOUISIANA  
POWER & LIGHT CO.  
Waterford Steam  
Electric Station

CHARPY TEST RESULTS  
BASE METAL PLATE  
M-1004-2 (RW)

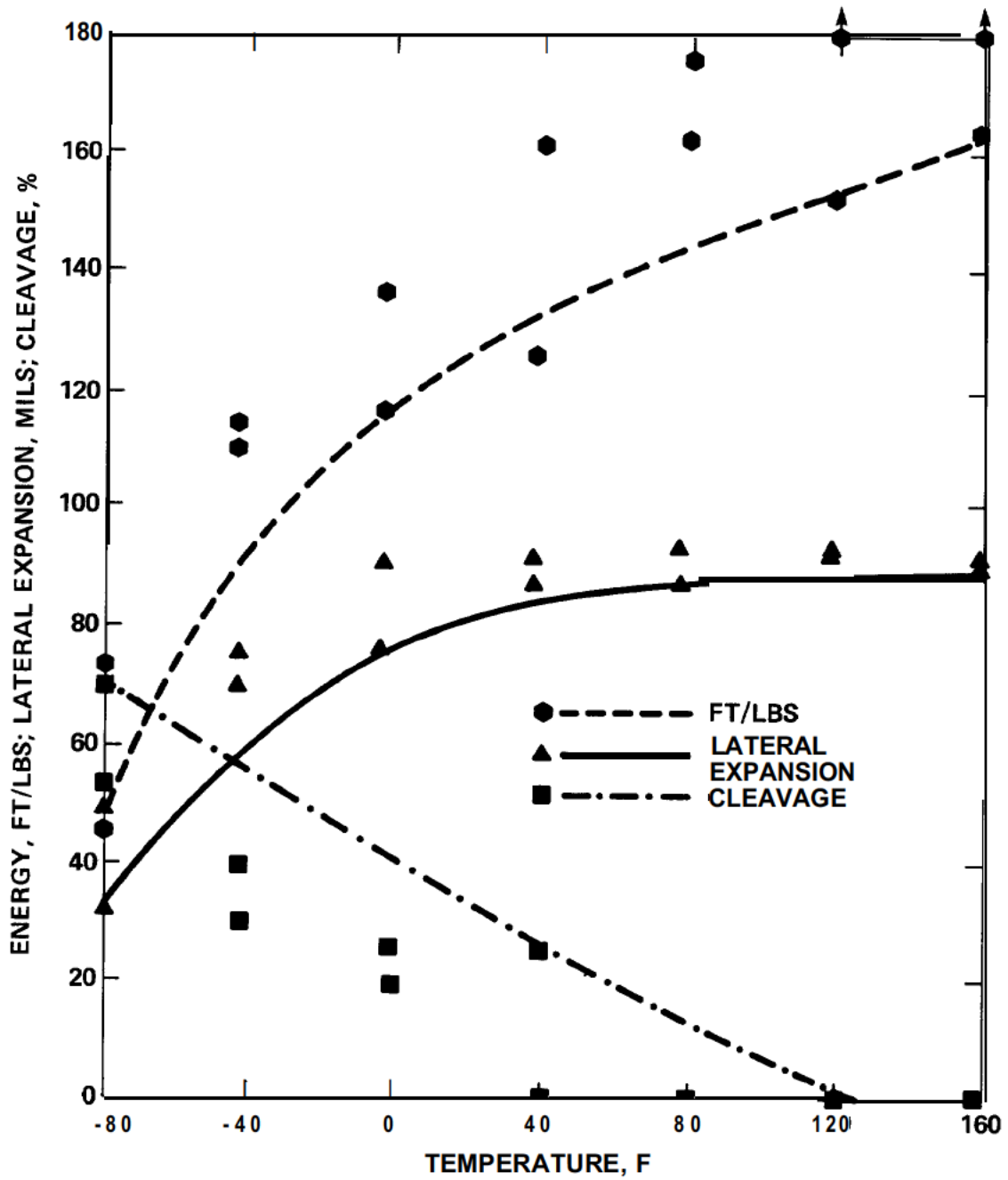
Figure  
5.3-2



LOUISIANA  
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Waterford Steam  
Electric Station

CHARPY TEST RESULTS  
WELD METAL PLATE  
M-1004-1/M-1004-3

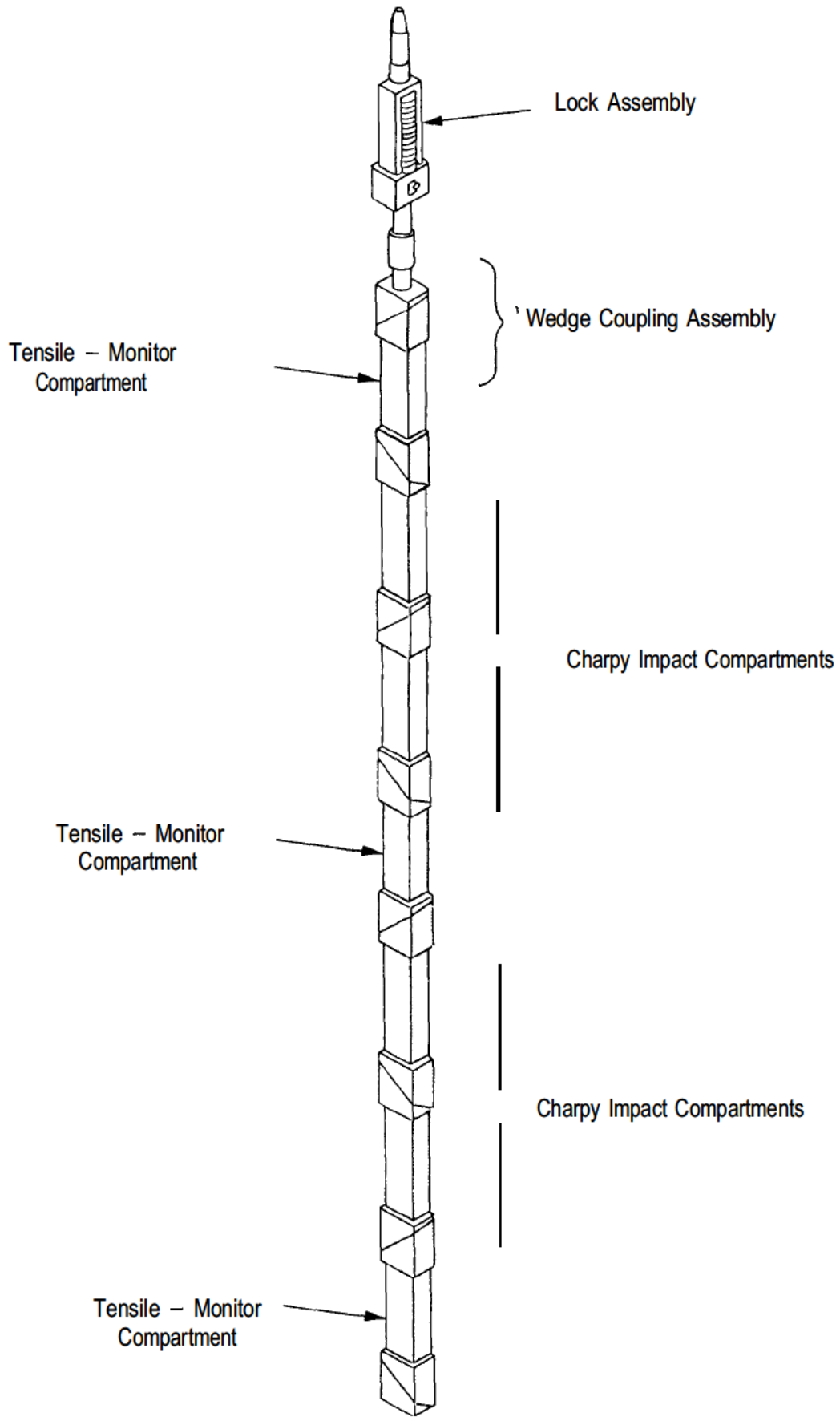
Figure  
5.3-3



LOUISIANA  
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Electric Station

CHARPY TEST RESULTS  
HEAT AFFECTED ZONE METAL PLATE  
M-1004-2

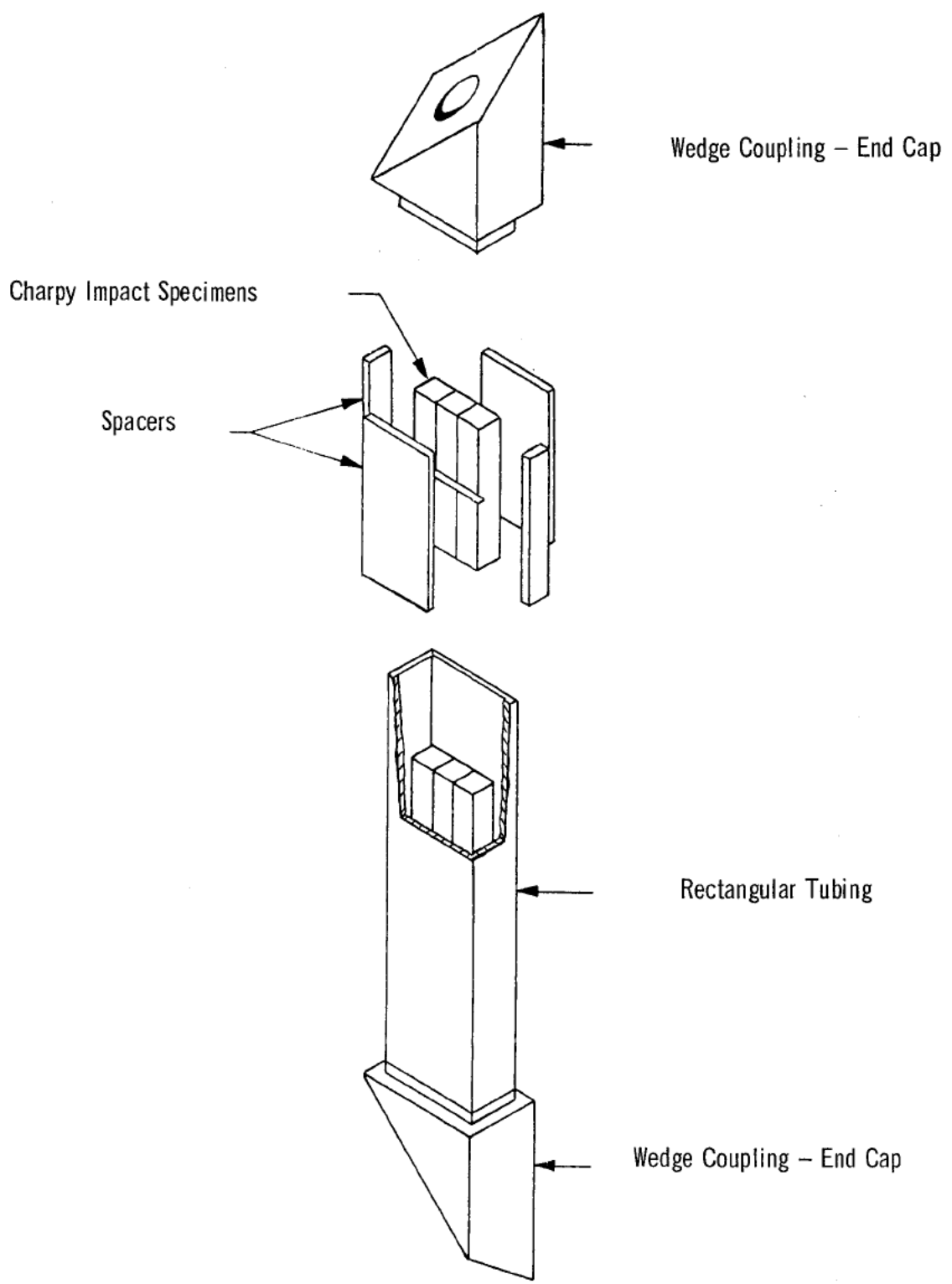
Figure  
5.3-4

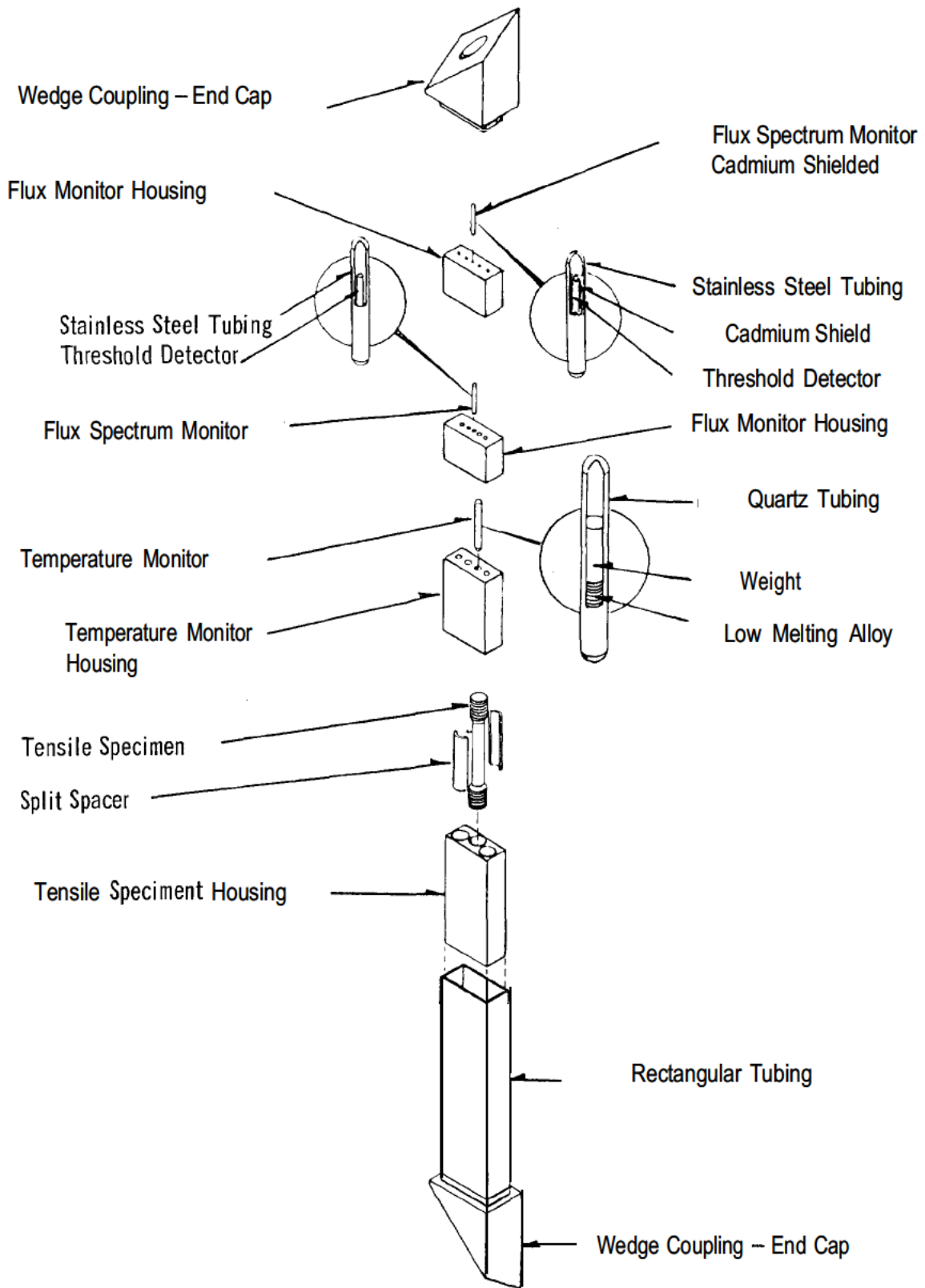


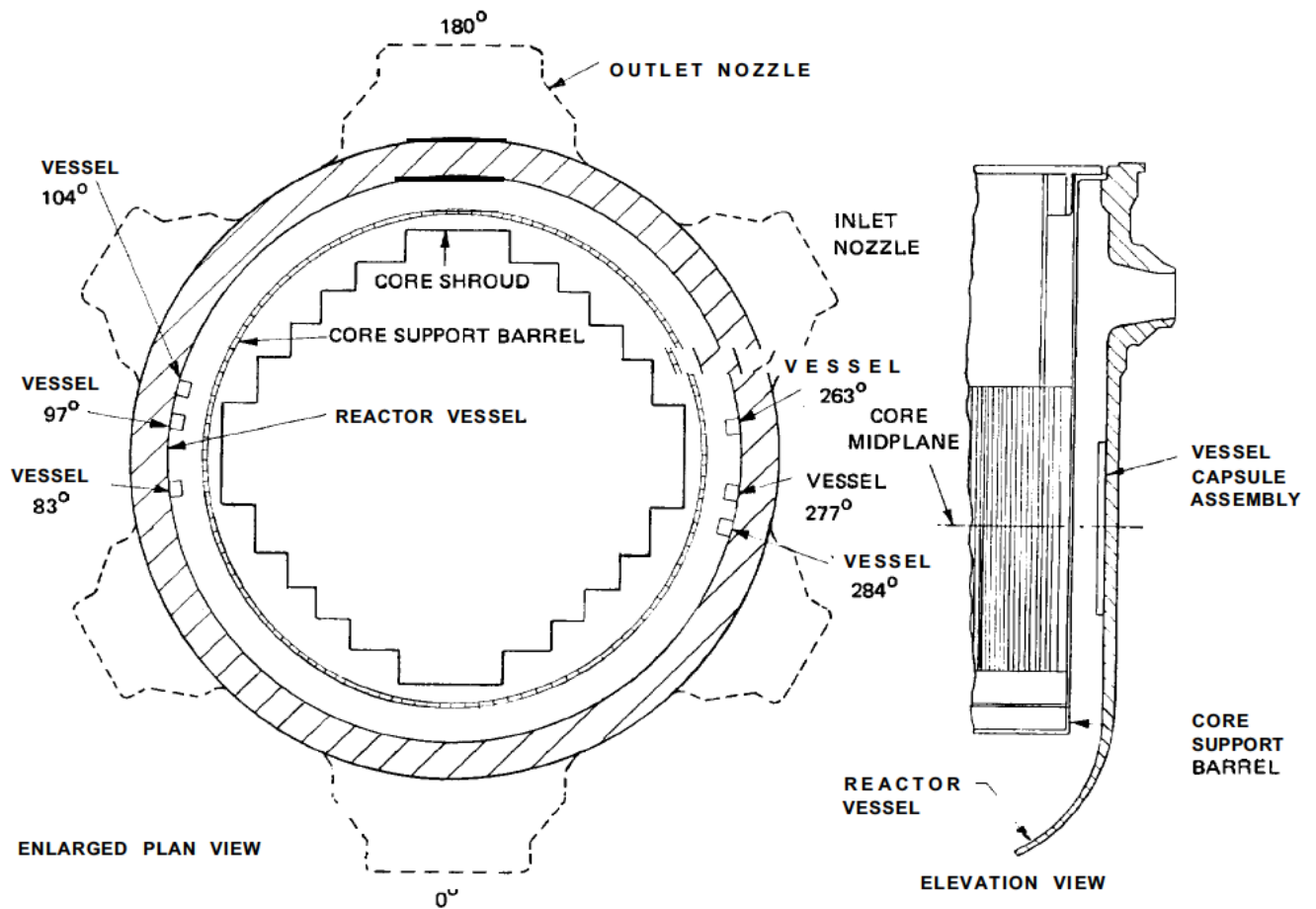
LOUISIANA  
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Electric Station

TYPICAL SURVEILLANCE CAPSULE ASSEMBLY

Figure  
5.3-5





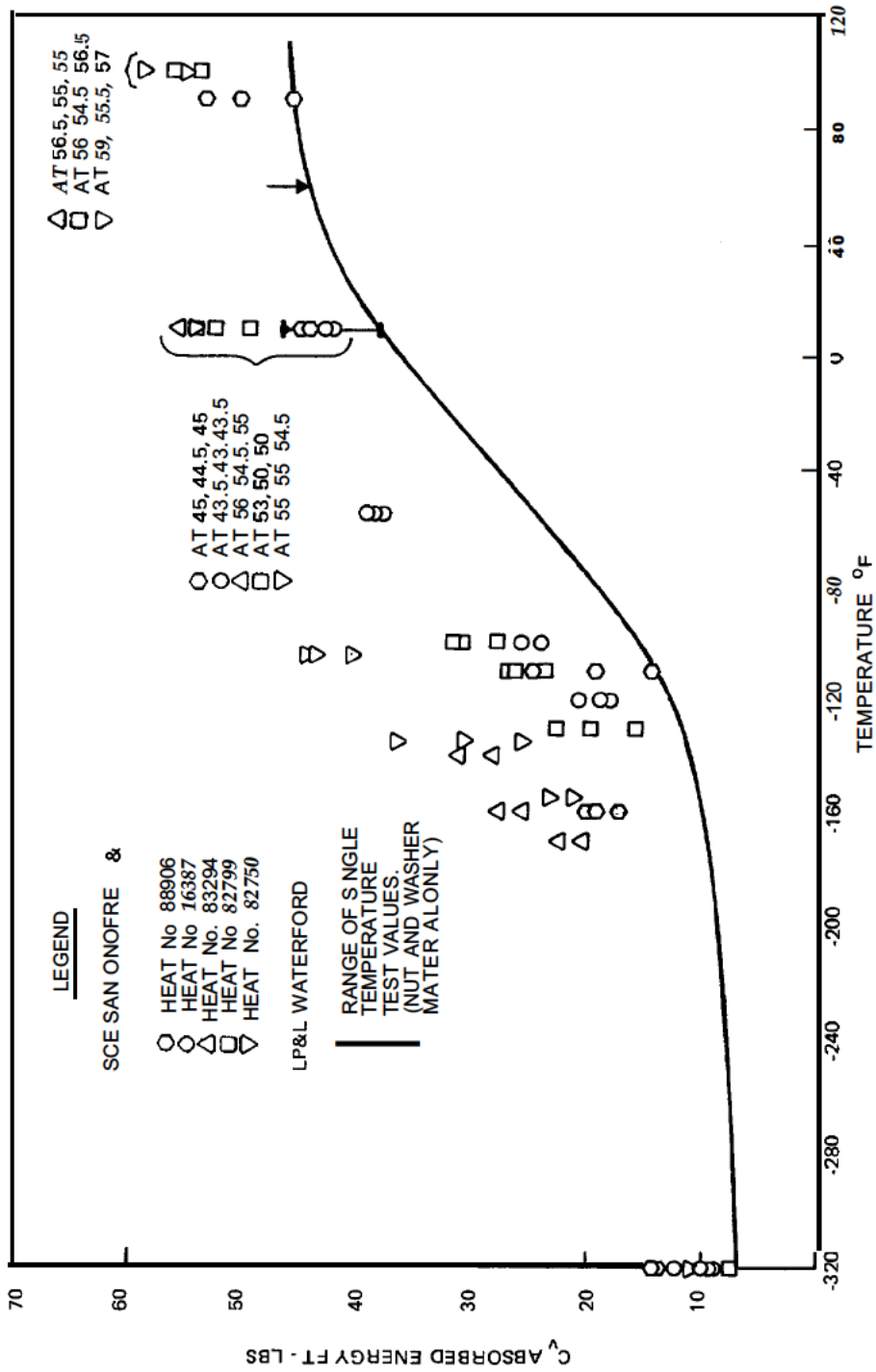


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Electric Station

LOCATIONS OF SURVEILLANCE CAPSULE ASSEMBLIES

Figure  
5.3-8



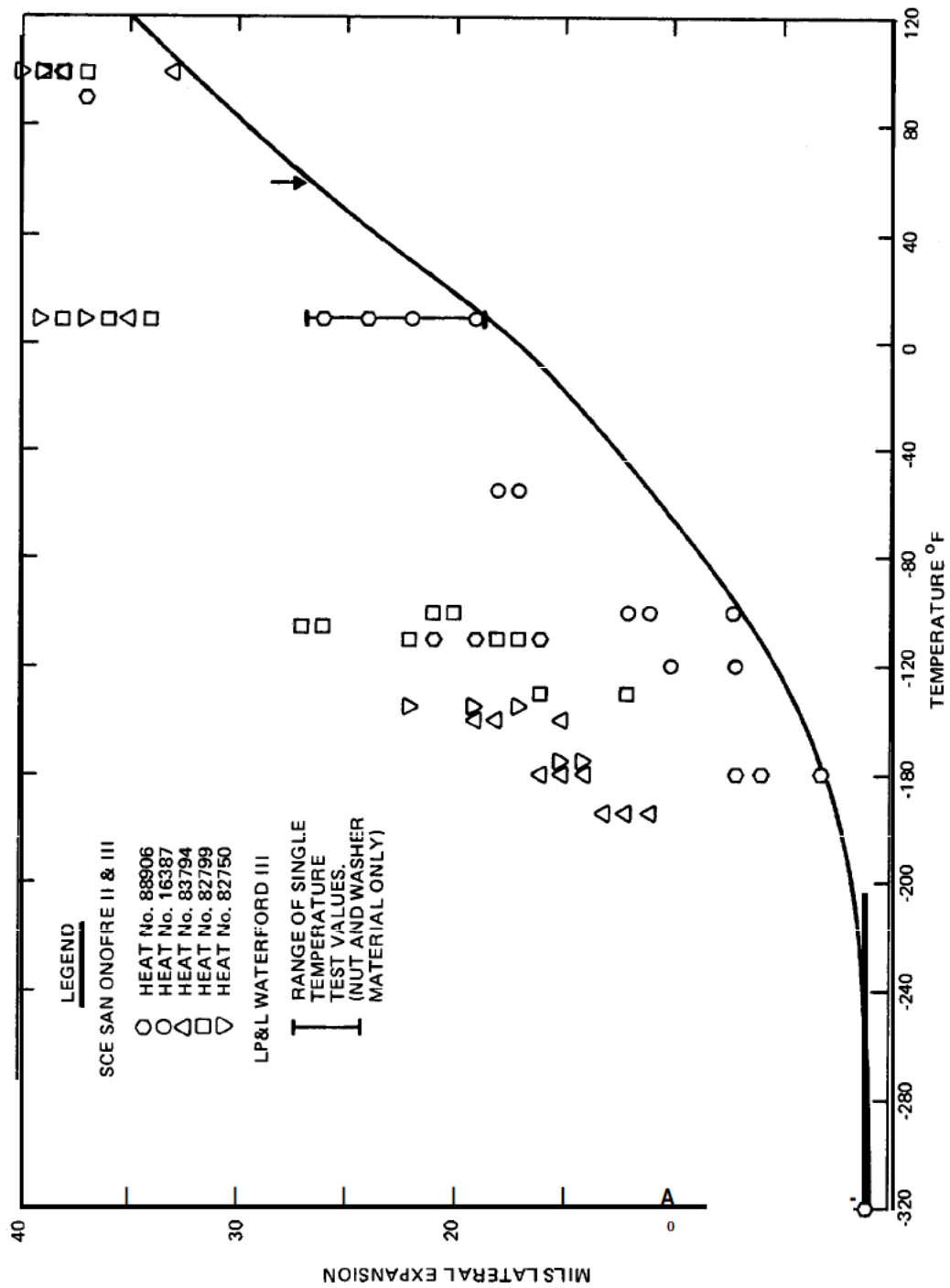


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 Waterford Steam  
 Electric Station

REACTOR VESSEL FASTENERS  
 CHARPY IMPACT ABSORBED ENERGY VS TEMPERATURE

Figure

5.3-9

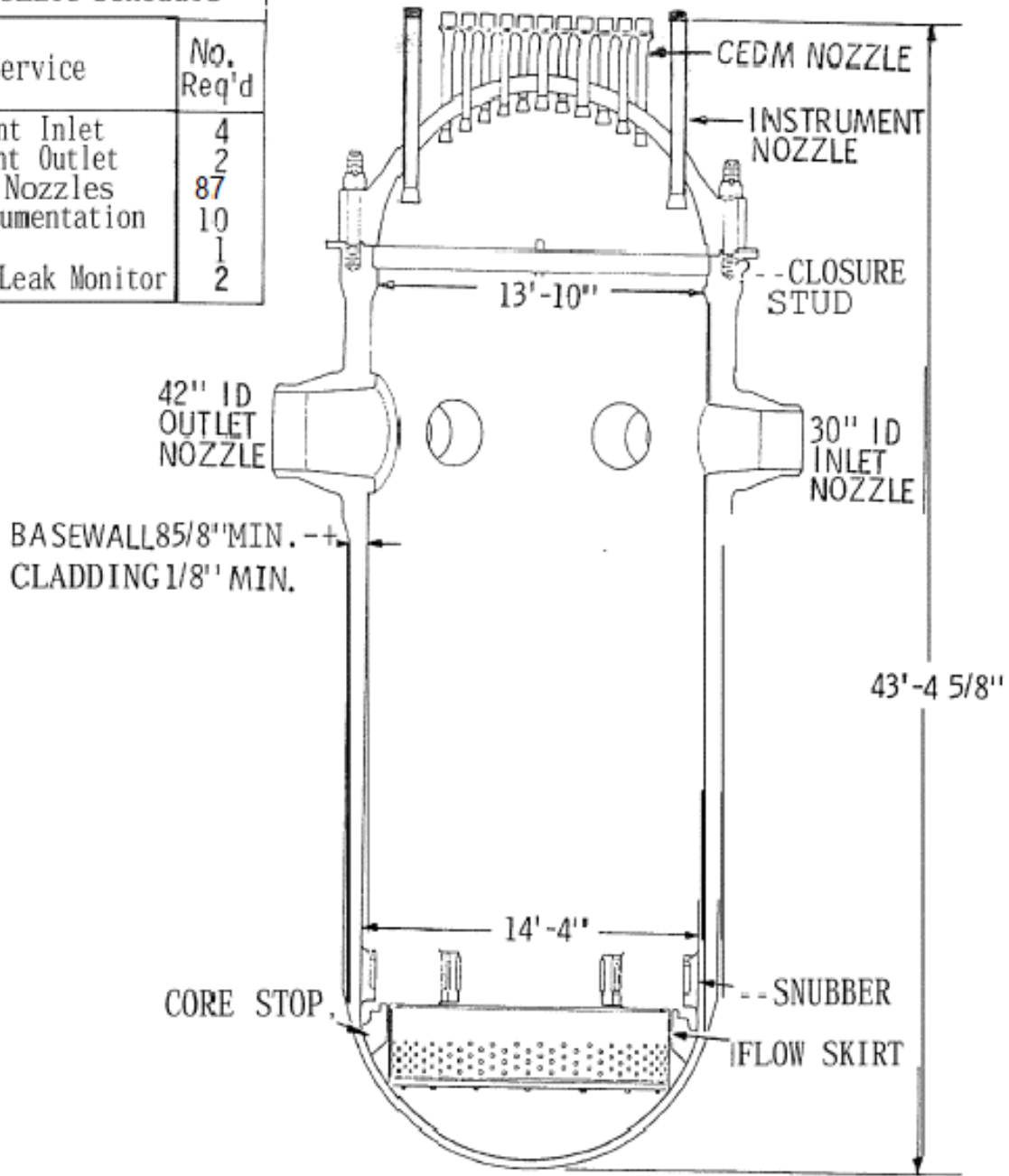


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Electric Station

REACTOR VESSEL FASTENERS  
LATERAL EXPANSION VS TEMPERATURE

Figure  
5.3-10

Nozzle Schedule	
Service	No. Req'd
Coolant Inlet	4
Coolant Outlet	2
CEDM Nozzles	87
Instrumentation	10
Vent	1
Seal Leak Monitor	2



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Electric Station

REACTOR VESSEL

Figure  
5.3-11

## WSES-FSAR-UNIT-3

### 5.4 COMPONENT AND SUBSYSTEM DESIGN

#### 5.4.1 REACTOR COOLANT PUMPS

##### 5.4.1.1 Design Bases

In addition to the design transients in Subsection 3.9.1.1, the reactor coolant pumps are designed to:

- a) Withstand 4000 cycles (pressure) of reactor coolant pump startup and stopping without exceeding code allowable stress limits. This transient is classified as a normal condition event.
- b) Provide sufficient moment of inertia to reduce the flow decay through the core upon loss of pump power, ensuring that fuel design limits are not exceeded.
- c) Prevent reverse rotation of the pump upon loss of pump power with the other pumps operating.
- d) Operate without cooling water for periods up to three minutes without incurring seal damage.

##### 5.4.1.2 Description

The reactor coolant is circulated by four vertical, single bottom suction, horizontal discharge centrifugal, motor driven pumps. (See Figures 5.4-1 and 5.4-2). The piping and instrumentation diagram for the reactor coolant pump is shown in Figure 5.1-3. Design parameters for the pumps are listed in Table 5.4-1. The predicted pump performance curve is shown in Figure 5.4-3.

##### 5.4.1.2.1 Reactor Coolant Pump Assembly

The reactor coolant pump assembly consists of the following:

- a) Pump Case
- b) Rotating Assembly  
(Containing the impeller which is welded to the shaft)
- c) Mechanical Seal Cartridge Assembly
- d) Motor Mount
- e) Motor Assembly

##### 5.4.1.2.1.1 Pump Case and Motor Mount

The reactor coolant pump motor is connected to and supported by the pump case through the motor mount. There are two openings on opposite sides of the motor mounts that provide access for assembly of the flanged rigid coupling between the motor and pump and for seal

## WSES-FSAR-UNIT-3

cartridge replacement. The pump case includes the seal heat exchanger, which cools the seal cartridge assembly.

### 5.4.1.2.1.2 Rotating Assembly

The pump rotating assembly consists of the impeller, a hydrostatic radial bearing, and the rotating elements of the seal cartridge assembly. The hydrostatic radial bearing maintains radial alignment of the shaft and is located just above the impeller. The upper radial bearings and an axial thrust bearing are located in the motor.

### 5.4.1.2.1.3 Mechanical Seal Cartridge Assembly

→(EC-6256, R302; EC-18520, R304)

The seal cartridge consists of four multiface type mechanical seals; three full-pressure seals mounted in series, and a fourth low pressure seal designed to withstand system operating pressure when the pump is not operating. A controlled bleedoff flow through the seals is used to cool the seals and to equalize the pressure drop across each seal. The seals are capable of operation without cooling water for up to ten minutes without incurring damage. The controlled bleedoff flow is collected in the volume control tank of the Chemical and Volume Control System (CVCS). Leakage past the vapor seal is routed to the Containment sump via the Floor Drain System.

←(EC-6256, R302; EC-18520, R304)

The seal cartridge concept reduces the time required for seal maintenance, thereby lowering personnel radiation exposure. The seal cartridge can be removed without draining the pump case. Details of the seal are shown in Figure 5.4-2.

### 5.4.1.2.1.4 Motor Assembly

The motor assembly includes the following:

- a) Air Cooler
- b) Motor Bearing Lubrication
- c) Oil Lift Pumps
- d) Motor Shaft
- e) Upper and Lower Radial Guide Bearings
- f) Axial Thrust Bearing
- g) Flywheel
- h) Anti-Reverse Rotation Device
- i) Motor

Cooling water for all motor heat exchangers is supplied from the Component Cooling Water System. Heat exchangers supplied include the upper bearing lube oil cooler, lower bearing oil cooler and motor air cooler.

## WSES-FSAR-UNIT-3

→(EC-8458, R307)

The oil lubrication system consists of two high pressure oil lift pumps with attached low pressure oil pumps, a lube oil cooler, motor lower bearing oil cooler, and slinger assembly all within the motor housing. During normal pump operation, the coil slinger maintains oil flow through the motor upper bearings. During pump startups and shutdowns, both the selected high pressure oil lift pump and its associated low pressure oil pumps are manually started to maintain sufficient oil pressure to the bearing and anti-reverse rotation device. If the oil pumps have not been started manually when the motor is stopped, the selected oil pump starts automatically, on a reactor coolant pump breaker trip. After the reactor coolant pump is at rated speed or has stopped, the operator must stop the lube oil pumps manually (see Subsection 5.4.1.5.2.3).

←(EC-8458, R307)

The motor bearing support system consists of a Kingsbury double acting thrust bearing and two radial bearings located above and below the motor rotor.

→(EC-8458, R307)

To improve pump coastdown characteristics in order to meet system requirements during a loss of pump power condition, the flywheel and motor-pump rotating assembly has a total moment of inertia of 106,500 lbm-ft<sup>2</sup>.

←(EC-8458, R307)

The reactor coolant pumps are designed with an anti-reverse rotation device (see Figure 5.4-4) to prevent reverse rotation of the pump. This device prevents reverse rotation after the pump stops from normal speed (1183 rpm) while the other pumps continue to operate.

The pump assembly is sized for continuous operation while pumping primary coolant with a specific gravity of 1.0. The motor is designed for 500 heatup cycles, during which the horsepower demand decreases from 9700 to 7800 over a period of approximately seven hours. The motors are also designed to start and accelerate to speed under full load at 80 percent of rated voltage. The pump and motor assembly is designed to operate in a radiation environment of  $10.5 \times 10^6$  rads (integrated) over a period of 40 years with nominal maintenance. Section 12.3 contains a further discussion of radiological considerations for these pumps.

### 5.4.1.3 Evaluation

→(EC-38245, R307)

The reactor coolant pumps are sized to deliver flow that equals or exceeds the design flow rate used in the thermal hydraulic analysis of the reactor coolant system. Analysis of steady state and anticipated transients is performed assuming the minimum design flow rate. Tests were performed to evaluate reactor coolant pump performance during the post core load hot functional testing to verify adequate flow.

←(EC-38245, R307)

→(DRN 03-2059, R14; EC-8458, R307)

The actual nominal RCS flowrate is approximately equal to 110% of the original design value of 99,000 gpm per pump. This results in a mass flowrate of approximately 163.5 Mlb/hr under normal operating conditions.

←(DRN 03-2059, R14; EC-8458, R307)

→(EC-30077, R 305; EC-38245, R307)

Leakage from the pump past the pump shaft is controlled by the shaft seal assembly. Coolant entering the seal chambers is cooled and collected in closed systems so that reactor coolant leakage to containment is essentially zero. The normal vapor stage leakoff is routed to the Containment Sump. In the event of a seal malfunction, instrumentation in the form of pressure transducers, a flow meter, and a temperature detector are provided to alert the operator to a potential problem.

←(EC-30077, R 305; EC-38245, R307)

→(DRN 05-150, R14)

Component cooling water to the reactor coolant pumps is not required to ensure (1) the integrity of the reactor coolant pressure boundary, (2) the capability to shutdown the reactor and maintain it in a safe shutdown condition, or (3) the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the guideline exposures of 10CFR50.67. Low cooling flow to each pump is alarmed

←(DRN 05-150, R14)

## WSES-FSAR-UNIT-3

in the main control room. Upon sounding of four alarms showing that flow is not available to all pumps, the operator trips the reactor and then trips the pumps allowing the system to be cooled down by natural circulation flow.

➔(DRN 00-1059, R11-A; EC-38245, R307)

The reactor coolant pumps, by design and field experience, are not susceptible to a loss-of-coolant accident (LOCA) due to seal failure resulting from loss of seal cooling water. The pumps are equipped with four series-arranged face seals, all of which are designed for 2500 psid. The  $\Delta P$  across any one of the three main seals during normal operation is 750 psi. The loss of any single seal would result in a  $\Delta P$  of approximately 1100 psi. A seal leakage chamber structurally designed for 2500 psia is provided to collect controlled seal leakage and conduct it to a closed system. The fourth face seal is provided as an integral part of the seal leakage chamber to prevent liquid or gaseous leakage from escaping to the atmosphere. The seal is designed to operate normally against a backpressure of 25 to 250 psia and is capable of holding against 2500 psia in the static condition and during coastdown following failure of the three series-arranged main seals. When holding against 2500 psia in the static condition, seal leakage should not exceed the normal operating seal leakage.

⬅(DRN 00-1059, R11-A)

Four reactor coolant pumps with seals of similar design have been operated for up to 40 minutes with no component cooling water flow. While there was some increase in controlled seal leakage (to the closed system), the mechanical seals were subsequently dismantled and refurbished without finding major damage. Therefore, a loss of component cooling water flow to the reactor coolant pumps for up to 40 minutes is not expected to result in a complete seal failure.

⬅(EC-38245, R307)

The component cooling water supply to the RCP's is isolated upon a Containment Spray Actuation Signal (CSAS). The RCP's are then stopped, resulting in no requirement for component cooling water. The CSAS is provided by a high-high containment pressure signal. Such a signal can be produced by only two accidents: (1) a LOCA, or (2) a major steam line break inside containment. In either case, the Plant Protective System trips the reactor and the operators trip the pumps.

In the event of a break in the reactor coolant pump suction piping, a high reverse flow through the pump is prevented by the anti-reverse rotation device, as described in Subsection 5.4.1.2.1.4. In the event of a discharge line break, increased flow through the pump tends to accelerate the pump impeller and flywheel in the forward direction. A detailed evaluation of this incident relating to the integrity of the flywheel is presented in Subsection 5.4.1.4.

### 5.4.1.4 Reactor Coolant Pump Flywheel Integrity

The design, fabrication and testing of the reactor coolant pump flywheel in relation to recommendations of Regulatory Guide 1.14 (formerly Safety Guide 14) and the results of an analysis of pump and flywheel performance during a LOCA are discussed in this section.

#### 5.4.1.4.1 Regulatory Guide 1.14, Rev. 00 (10/27/71)

The following design conditions and material specifications for the flywheels are consistent with the recommendations of Regulatory Guide 1.14, Rev. 00, "Reactor Coolant Pump Flywheel Integrity".

## WSES-FSAR-UNIT-3

### 5.4.1.4.1.1 Flywheel Material Specifications

The material used to manufacture the flywheel is pressure vessel quality, prepared by the vacuum melt and degassing process, ASTM-A-543 Grade B, Class I steel plate. Pump flywheel material is prepared and tested such that:

- a) The nil-ductility transition temperature of the flywheel is no greater than +10°F, as obtained from drop-weight tests performed in accordance with specification ASTM-E-208.
- b) The Cv upper shelf energy level, in the Wr direction, as obtained per ASTM-A-370, is no less than 50 ft lb. A minimum of three Cv specimens are tested from each plate.  
→(DRN 00-1059, R11-A; 05-1392, R14-A)
- c) The minimum fracture toughness of the material at the normal operating temperature of the flywheel is equivalent to a dynamic stress intensity factor ( $K_{IC}$  dynamic) and is equal to or greater than 100 Ksi  $\sqrt{in}$ . Compliance is demonstrated by one of three methods listed in NRC Regulatory Guide 1.14, Rev. 00, paragraph C.1.c.  
←(DRN 00-1059, R11-A; 05-1392, R14-A)
- d) Each finished flywheel is subjected to a 100 percent volumetric, ultrasonic examination from the flat surface using ASME Section III, Class I criteria. This inspection is performed on the flywheel after final machining and overspeed test.
- e) After flame cutting, at least 1/2 in. of stock is left on the outer and bore radii for machining to final dimensions.
- f) Finished machined bores, keyways and drilled holes in the flywheel are subjected to a magnetic particle or liquid penetrant examination before final assembly.

→(DRN 03-2059, R14)

### 5.4.1.4.1.2 Flywheel Design Criteria

The flywheels are designed to withstand normal operating conditions, anticipated transients, and the original design basis LOCA<sup>1</sup> combined with the safe shutdown earthquake. Thus the following criteria are met:

←(DRN 03-2059, R14)

- a) The combined primary stresses at normal operating speed do not exceed one-third of the minimum specified yield strength or one-third of the measured yield strength in the weak direction of the material.
- b) The combined primary stresses at design overspeed do not exceed two thirds of the minimum specified yield strength or two-thirds of the measured yield strength in the weak direction of the material. Design overspeed is defined as 125 percent of normal operating speed. See Subsection 5.4.1.4.2 for further discussion of pump overspeed following a discharge line break.

Each flywheel is tested for one minute at the design overspeed of 1500 rpm.

The flywheel is accessible for 100 percent in-place volumetric ultrasonic inspection. The motor assembly is designed to allow such inspection with a minimum of motor disassembly.

→(DRN 03-2059, R14)

<sup>1</sup>The effects of the original design basis LOCA (i.e., MCLBs, which have been eliminated via LBB, see Section 3.6.3) envelop the effects of the next limiting set of pipe breaks (BLPBs) on RCS flywheel design.

←(DRN 03-2059, R14)



## 5.4.1.4.2 Additional Data and Analysis

→(DRN 03-2059, R14)

As mentioned previously, a LOCA in the pump discharge leg of the reactor coolant piping tends to accelerate the pump in the forward direction during blowdown. A maximum speed of 1585 rpm has been calculated using the CE FLASH computer program, assuming a mechanistic break<sup>1</sup> located at the pump discharge with a simultaneous loss of electrical power to the pump motor. This is about 134 percent of normal operating speed. Even at pump speeds greater than 1585 rpm, no other failures of sufficient magnitude to increase the consequences of the initiating LOCA can be postulated except by assuming failure of the flywheel. Stress calculations have demonstrated that even at speeds up to 270 percent of normal operating speed with a sharp edged crack located to the base of the keyway, failure cannot be predicted. It is therefore concluded that adequate margin has been provided to preclude the development of a flywheel rupture by reactor coolant pump overspeed.

←(DRN 03-2059, R14)

5.4.1.5 Reactor Coolant Pump Instrumentation

The reactor coolant pumps and motors are equipped with the instrumentation necessary for proper operation and to warn of incipient failures. A description of the major channels follows. See Figure 5.1-3. Measurement channels are typical for each reactor coolant pump.

## 5.4.1.5.1 Temperature

## 5.4.1.5.1.1 Motor Stator Temperature

Each reactor coolant pump motor is provided with six thermocouples embedded in the stator windings. Indication of stator temperature is provided by the plant monitoring computer. During initial pump testing, the highest reading thermocouple will be selected for this temperature measurement channel. Should stator temperature exceed a predetermined limit, a high temperature alarm will be annunciated by the plant monitoring computer. High temperature is detrimental to motor winding insulation life, and may be caused by high ambient temperature, reduction in the cooling airflow to the stator, or inadequate time delay between successive starts of the motor.

## 5.4.1.5.1.2 Motor Bearing Temperatures

High temperature alarms for the oil lubricated bearings are annunciated by the plant monitoring computer at the main control board. These include:

- a) Motor upper guide bearing
- b) Upper and lower thrust bearings
- c) Motor lower guide bearing

High temperature is indicative of bearing or oil supply problems. The motor thrust bearing temperature is continuously monitored at the main control board. Temperatures of the motor guide bearings are fed to the plant monitoring computer.

→(DRN 03-2059, R14)

<sup>1</sup>The effects of the original design basis LOCA (i.e., MCLBs, which have been eliminated via LBB, see Section 3.6.3) envelop the effects of the next limiting set of pipe breaks (BLPBs) on RCS flywheel design.

←(DRN 03-2059, R14)

## WSES-FSAR-UNIT-3

### 5.4.1.5.1.3 Pump Controlled Bleedoff Temperature

The temperature of the controlled bleedoff flow is displayed in the main control room. An alarm signal is provided at the main control panel should the controlled bleedoff temperature exceed a high limit. A high temperature condition is an indication that the seal assembly or seal water cooler is not operating properly.

### 5.4.1.5.1.4 Lube Oil Cooler Inlet Temperature

→

The inlet temperature of the lube oil cooler is alarmed by the plant monitoring computer.

←

### 5.4.1.5.1.5 Lube Oil Cooler Outlet Temperature

→

The outlet temperature of the lube oil cooler is alarmed by the plant monitoring computer.

←

### 5.4.1.5.1.6 Anti-Reverse Rotation Device Bearing Temperature

→

Indication of the anti-reverse rotation device temperature is provided by the plant monitoring computer. A high temperature condition in the anti-reverse rotation device is alarmed by the plant monitoring computer and is an indication of lubrication problems.

←

### 5.4.1.5.1.7 Seal Water Cooler Component Cooling Water Outlet Temperature

The temperature of component cooling water is monitored at the seal water cooler outlet of each Reactor Coolant Pump (1A, 1B, 2A, 2B) with respective temperature elements. The temperature parameter is displayed in the main control room.

Two independent high temperature setpoints are used to annunciate an alarm in the main control room and to initiate an automatic isolation of the affected seal cooler by closing the component coolant water inlet and outlet valves. The high temperature alarm is an indication that leakage from the primary loop to the component coolant water has developed inside the seal water cooler or that the cooling water flow has decreased.

To countermand an isolation of seal coolers on a spurious signal, a control switch is provided in the main control room to allow the operator to override the high temperature signal and to reopen the component coolant water inlet and outlet valves to purge the seal water cooler for a preset time span. In that case, if the high temperature signal persists, the seal water cooler will be again automatically isolated at the end of the preset time span.

### 5.4.1.5.2 Pressure

#### 5.4.1.5.2.1 Pump Seal Pressures

→

The middle, upper and vapor seal cavities in each pump are provided with pressure elements that generate a signal proportional to the pressure within the cavity. The pressure in the seals is displayed in the main control room and are monitored by the plant monitoring computer. High and low pressure alarms for the middle and upper seal cavities and a high pressure alarm for the vapor seal cavity are annunciated by the plant monitoring computer.

←

Abnormally high or low pressure is an indication of seal malfunction.

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### 5.4.1.5.2.2 Lube Oil Cooler Pressure

A common low pressure alarm at the inlet to the lube oil cooler and anti-reverse rotation device is annunciated in the main control room.

### 5.4.1.5.2.3 High Pressure Oil Lift Pump Discharge Pressures

Pressure switches at each high pressure oil lift pump discharge actuate an alarm on low pressure in the main control room. In the event of a failure of one of the oil lift pumps, the second oil lift pump must be started. A separate measurement channel provides a control signal to the respective reactor coolant pump circuit, which prevents the starting of the reactor coolant pump if insufficient oil lift pressure exists. Another separate measurement channel provides local indication of pressure in the oil lift pump discharge header.

### 5.4.1.5.2.4 High Pressure Oil Lift Pump and Low Pressure Oil Pump Filters

→

Filters are used during motor start-up and shutdown at the suction and discharges of the high pressure oil lift and low pressure oil pumps. A high differential pressure across the filter indicates a clogged filter and is alarmed by the plant monitoring computer.

←

### 5.4.1.5.2.5 Reactor Coolant Pump Differential Pressure

Two independent differential pressure transmitters are provided on each reactor coolant pump. The differential pressure signal is indicated in the main control room. A calibration curve is used to relate pump differential pressure to pump flow.

### 5.4.1.5.2.6 Lube Oil Pressure to Anti-Rotation Device

Local pressure gages are provided in the oil line to the anti-rotation device for testing purposes only.

### 5.4.1.5.2.7 Reverse Rotation Indicator Switch

→

A differential pressure switch in the lube oil system mounted near the upper bearing bracket provides an indication that the reactor coolant pump motor is turning in the reverse direction. This switch alarms via the plant monitoring computer.

←

### 5.4.1.5.2.8 Seal Water Cooler Component Cooling Water Outlet Pressure

The component cooling water pressure is monitored at each of the reactor coolant pumps (RCP 1A, 1B, 2A, and 2B) seal water outlet. A high pressure signal annunciates an alarm in the main control room.

## WSES-FSAR-UNIT-3

### 5.4.1.5.3 Flow

#### 5.4.1.5.3.1 Pump Controlled Bleedoff Flow

A flowmeter is used to measure the controlled bleedoff flow from the bleedoff seal cavity to the CVCS. This instrument provides an indication of the flow rate and annunciates high and low flow alarms via the plant monitoring computer. Abnormal flow is an indication of improper seal performance.

#### 5.4.1.5.3.2 Motor Circulating Oil System Flow

A lube oil flow switch is provided at the inlet to the lube oil cooler. Should the lube oil flow to the cooler fall below a predetermined setpoint, a low flow alarm is actuated via the plant monitoring computer.

#### 5.4.1.5.3.3 Motor Low Pressure Oil Pump Flow

The lube oil flow to the anti-reverse rotation device is sensed by a lube oil flow switch. This switch actuates a low flow alarm via the plant monitoring computer when flow drops below the predetermined setpoint.

#### 5.4.1.5.3.4 Motor High Pressure Oil Lift Pump Flow

Flow in the discharge header of the two high pressure oil lift pumps is sensed by a lube oil flow switch. This switch actuates a low flow alarm via the plant monitoring computer when flow drops below the predetermined setpoint.

### 5.4.1.5.4 Level

A level sensing transmitter in each oil reservoir transmits a signal for level indication and high and low alarms via the plant monitoring computer.

### 5.4.1.5.5 Vibration

Motor vibration is sensed by a vibration switch attached to the pump motor casing. Excessive vibration is alarmed by the plant monitoring computer.

Review of Generic Letter 89-15 prompted the addition of new RC pump and motor monitoring. Though not required by that letter, additional "monitoring" devices have been added to correlate data to provide input and trending information on the status of motor, shaft, bearings and seals. The addition consists of x and y direction radial proximity probes and one axial thrust probe at the motor thrust bearing, x and y direction radial proximity probes at the bottom of the motor and x and y direction radial proximity probes at the pump shaft. The output of these devices does not interact with any plant controls, but provides additional "monitoring" capability.

Review of Generic Letter 89-15 also prompted the addition of two (2) keyphasors at the RC pump coupling spacer. These provide additional monitoring of shaft rotative speed and vibration phase angle. These devices do not interact with any plant controls, but provide additional "monitoring" capability.

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### 5.4.1.5.6 Speed

#### 5.4.1.5.6.3 Reactor Coolant Pump Speed Sensors

Reactor coolant pump shaft speed is transmitted to the Core Operating Limit Supervisory System (COLSS) and the Plant Protective System (PPS) for continuous monitoring and protection. See Sections 7.2 and 7.7 for a further description.

#### 5.4.1.6 Tests and Inspections

→(DRN 00-1059, R11-A)

The reactor coolant pump pressure boundary is nondestructively inspected as required by the ASME Code, Section III (see Table 5.2-1) for Class 1 components. The pump casing inspections include complete radiography and liquid penetrant testing. The pump casing, cover and seal cooling heat exchanger are subjected to a hydrostatic test at the manufacturing facility. The pump is hydrostatically pressure tested along with the RCS. In-service inspection will be performed during plant life in accordance with the ASME Code, Section XI as discussed in Subsection 5.2.4.

←(DRN 00-1059, R11-A)

All rotating parts of the pump are statically and dynamically balanced in two planes. Where possible, balancing is done for the entire assembly.

The pump assembly is performance tested in the vendor's shop in accordance with the Standards of the Hydraulic Institute to verify hydraulic performance, as well as the ability of the pumps to function as required by the specifications. The vibration levels are monitored during this test. Evidence of the pumps operating near a critical speed would be noted as excessive vibration.

Full scale seal testing is performed at rated pressure, temperature, water chemistry, and speed to demonstrate the capability of the seals to satisfactorily perform their design function.

### 5.4.2 STEAM GENERATORS

#### 5.4.2.1 Design Basis

→(DRN 03-2059, R14; 06-1060, R15; EC-8458, R307; LBDCR 16-037, R310)

Replacement Steam Generators, implemented in Cycle 19, are designed to transfer 3735 MWt from the RCS to the secondary system with 0% tube plugging, producing approximately  $16.6 \times 10^6$  lbm/hr of 837.2 psia saturated steam. At 10% tube plugging, the RSGs produce approximately  $16.6 \times 10^6$  lbm/hr of 818.0 psia saturated steam. Moisture separators and steam driers in the shell side of the steam

←(DRN 03-2059, R14; 06-1060, R15; EC-8458, R307; LBDCR 16-037, R310)

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➔(DRN 03-2059, R14; EC-8458, R307)

generator limit the moisture content of the steam to 0.10% during normal operation at full power. The steam generator design parameters are listed in Table 5.4-2. The steam generators, including the tubes, are designed for the RCS transients listed in Subsection 3.9.1.1 so that the code allowable stress limits are not exceeded for the specified number of cycles. All transients have been established based on conservative assumptions of operating conditions in consideration of supportive system design capabilities. The steam generators are capable of sustaining the following additional design transients without exceeding code allowable stress limits:

←(DRN 03-2059, R14)

- a) Ten secondary side hydrostatic tests with secondary side pressurized to 1-1/4 times the design pressure and the primary side pressurized so that the tube differential pressure does not exceed 1375 psia psi (test condition).

- b) Secondary side tube leak test

Cycles	Shell Side Pressure (PSIG)	Shell Side Temperature (°F)
400	200	60-250
200	400	60-250
120	600	60-250
80	840	60-250

- c) 5.700 cycles of adding feedwater to the steam generators through the main feedwater nozzle when in hot standby conditions (normal operations). The following conditions and cycles are based on actual operating conditions:

Cycles	Feedwater temp. (°F)	Flow change (gpm)	Rate for flow change
600	40	50 to 1100	1 second
948	40	0 to 450	1 second
1302	40	0 to 50	70 seconds
600	70	50 to 1100	1 second
948	70	0 to 450	1 second
1302	70	0 to 50	70 seconds

➔(DRN 00-1059, R11-A)

- d) 600 cycles of feeding during a three hour low power transient.

Cycles	Power change (%)	Feedwater temp change (°F)	Flow change (lb/hr)
300	0 to 15	40 to 260	25,000 to 1.04x10 <sup>6</sup>
300	15 to 0	260 to 40	1.04x10 <sup>6</sup> to 25,000

←(DRN 00-1059, R11-A)

- e) Four thousand pressure transients of 113 psi across the primary divider plate in either direction caused by starting and stopping reactor coolant pumps (normal condition).
- f) Twenty cycles of a Loss of Main Feedwater Transient.
- g) One cycle for a Design Bases Earthquake transient with a Steam Line Break.
- h) One cycle for a Design Bases Earthquake transient with a Feedwater Line Break.

←(EC-8458, R307)

The operating pressure and temperature limits for the steam generator primary side were determined in accordance with the ASME Code, Section III, Appendix G.

➔(EC-8458, R307)

The method of fastening tubes to the tube sheet conform with the requirements of Sections III and IX of the ASME Code. Tube expansion into the tube sheet is total with no voids or crevices occurring along the length of the tube in the tube sheet. Eight tube support plates of the flat-contact broached trifoil tube hole design are used in the RSG. The broached tube support plate is designed to reduce the tube-to-tube support plate crevice area while providing for maximum steam/water flow in the open areas adjacent to the tube. Flat tube contact geometry in the RSG provides additional dryout margin.

←(EC-8458, R307)

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→(EC-8458, R307)

The steam generator is designed to ensure that critical vibration frequencies are well out of the range expected during normal operation and during abnormal conditions. The tubing and tubing supports are designed and fabricated with considerations given to both secondary side flow induced vibration and reactor coolant pump induced vibrations. In addition, the steam generator assemblies are designed to withstand the blowdown forces resulting from severance at the steam nozzle.

←(EC-8458, R307)

Discussion of the techniques used to maintain cleanliness during final assembly and shipment are discussed in Subsection 5.2.3.

Onsite cleaning and cleanliness control procedures for the steam generator were consistent with the recommendations of Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants", (3/16/73) and ANSI N45.2-1-1973, "Cleaning of Fluid Systems and Associated Components For Nuclear Power Plants".

### 5.4.2.2 Description

→(DRN 03-2059, R14; EC-8458, R307)

The steam generator is illustrated in Figure 5.4-5. Moisture-separating equipment in the shell side of the steam generators limits moisture content of the exit steam to a maximum of 0.10%. Manways and handholes are provided for access to the steam generator internals.

←(DRN 03-2059, R14; EC-8458, R307)

Reactor coolant enters at the bottom of each steam generator through the single inlet nozzle, flows upward through the U-tubes, and leaves through the two outlet nozzles. A vertical divider plate separates the inlet and outlet plenums in the lower head.

→(DRN 02-674, R12; 03-130, R14; 06-290, R14-B; EC-8458, R307)

Feedwater enters the steam generator through the feedwater nozzle where it is distributed via a feedwater distribution ring. The feedwater ring is constructed with eccentric reducer fittings and perforated discharge nozzles which discharge at the top of the feeding to avoid regions where steam pockets can be formed. The feedwater ring is located above the elevation of the feed nozzle to minimize the time required to fill the feed nozzle during a cold water addition transient.

←(DRN 02-674, R12; 03-130, R14; 06-290, R14-B; EC-8458, R307)

The downcomer in the steam generator is an annular passage formed by the inner surface of the steam generator shell and the cylindrical shell that encloses the vertical U-tubes. Upon exiting from the bottom of the downcomer, the secondary flow is directed upward over the vertical U-tubes. Heat transferred from the primary side converts a portion of the secondary flow into steam.

→(EC-8458, R307)

Upon leaving the vertical U-tube heat transfer surface, the steam-water mixture enters the centrifugal-type separators. These impart a centrifugal motion to the mixture and separate the water particles from the steam. The water exits from the perforated separator housing and combines with the feedwater to repeat the cycle. Final drying of the steam is accomplished by passage of the steam through the single-tier banked dryers. Drain pipes located at the ends of each secondary separator dryer bank carry captured water downward into the recirculation pool. The pressure drop from the steam generator feedwater nozzles to the steam outlet nozzle is approximately 31.6 psi when operating at full power. The steam generator supports are described in Subsection 5.4.14. Secondary side overpressure protection is provided by 12 spring-loaded ASME Code safety valves mounted on the main steam lines as described in Subsection 5.4.13.

The RSGs have a 3.00 in NPS blowdown nozzle connected to the steam generator secondary water through a series of holes in the tubesheet secondary surface that intersect an internal blowdown passage.

←(EC-8458, R307)

→(EC-8458, R307)

The RSGs contain an integral flow limiting device, consisting of seven 8.53 inch I.D. venturis installed in the steam outlet nozzle. This flow restrictor reduces energy release to containment and loads on the steam generator internals during a postulated steamline break.

←(EC-8458, R307)

#### 5.4.2.3 Evaluation

##### 5.4.2.3.1 Steam Generator Tubes

##### 5.4.2.3.1.1 Chemistry Compatibility

→(EC-8458, R307)

The steam generators, tubed with Alloy 690 TT, 0.75 in. OD by 0.044 for Rows 1&2 and 0.043 for Rows 3 to 138 wall tubing, incorporates a general corrosion allowance that will provide for reliable operation over the plant design lifetime.

←(EC-8458, R307)

Localized corrosion has led to steam generator tube leakage in some operating plants. Examination of tube defects that have resulted in leakage has shown that two mechanisms are primarily responsible. These localized corrosion mechanisms are referred to as (1) stress assisted caustic cracking, and (2) wastage or beavering. Both of these types of corrosion have been related to steam generators that have operated on phosphate chemistry. The caustic stress corrosion type of failure is precluded by controlling bulk water chemistry to the specification limits shown in Subsection 10.3.5. Removal of solids from the secondary side of the steam generator is discussed in Subsection 10.4.8.

Localized wastage has been eliminated by removing phosphates from the chemistry control program.

Volatile chemistry (discussed in Subsection 10.3.5) has been successfully used in all C-E steam generators that have gone into operation since 1972.

##### 5.4.2.3.1.2 Mechanical Considerations

Because the reactor coolant pumps have a rotational speed of 1180 rpm, the imposition of exciting frequencies of 19 to 20 Hz and 95 to 100 Hz was considered in the design. The low frequency range is defined as a mechanical vibration resulting from the transmission of a mechanical impulse at the frequency of pump rotation. The upper frequency range is defined as a sinusoidal pressure vibration of  $\pm$  six psi in the reactor coolant piping that contains the pump. The pressure variation results from the impeller vanes interacting with the cut-water vane at the volute outlet during each revolution of the impeller.

##### 5.4.2.3.1.3 Hydraulic Stability of Feedwater System

Pressure pulses have been observed in some feedwater lines which were initiated by steam-water interaction causing ripple formation at a steam-water interface in the feedwater piping. This resulted in the formation of a water slug which isolated the steam in the feedwater line. As the isolated steam bubble condensed, the decreasing pressure would accelerate the water slug in the direction of the void. The kinetic energy in the slug would increase until the steam bubble collapsed. When the water slug impacted with the water filling the upstream side of the pipe, pressure pulses were generated. With a design incorporating only a small amount of entrapped steam, the steam bubble will collapse before the water slug can gain significant kinetic energy, and the intensity of the pressure pulses is reduced to negligible levels.



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→(EC-18652, R304)

In Waterford 3, the potential for a significant waterhammer transient in the Feedwater System is reduced by the construction of the feedwater ring, described in Subsection 5.4.2.2. Additional assurance that the feedwater ring will remain full of water is provided by automatic initiation of the Emergency Feedwater System (see Section 7.3). In addition, a downward sloping elbow at the steam generator feedwater nozzle connects to the feedwater piping. This design minimizes the drainable volume of the feedwater piping. Hence, when the feedwater ring and nozzle are drained and steam enters this region, the surface area of subcooled water exposed to saturated steam is minimized. As a result, only a small amount of steam can be trapped in the elbow and any water slug picked up during steam-water interaction is incapable of gaining significant kinetic energy before the steam bubble is collapsed.

←(EC-18652, R304)

The effectiveness of the similar modifications has already been demonstrated at Indian Point 2, Trojan and St. Lucie I. Tests at a European plant (Doel I in Belgium) have also confirmed the effectiveness of a short horizontal run in reducing waterhammer to negligible levels.

→(EC-18652, R304)

The feedring is subject to damage from pressure transients that can occur when refilling a steam generator whose water level has fallen below the feedring. Removal of the thermal liner o-ring and clamp on both steam generators and the acceptance of missing vent piping on Steam Generator #1 allows the feedring to drain after a loss of feedwater. These changes were evaluated on the basis of the Steam Generators remaining capable of performing their safety function with a damaged feedring.

←(EC-18652, R304)

### 5.4.2.3.1.4 Tube Wall Thinning

→(EC-8458, R307)

Tube wall thinning acceptance criteria is specified in the Technical Specifications. The Replacement Steam Generator tube structural integrity analysis is provided in WCAP-17263-P (Reference 1).

←(EC-8458, R307)

### 5.4.2.3.2 Potential Effects of Tube Rupture

→(DRN 01-1283, R12)

The steam generator tube rupture accident is a penetration of the barrier between the RCS and the Main Steam System. The integrity of this barrier is significant from the standpoint of radiological safety in that a leaking steam generator tube allows the transfer of reactor coolant into the Main Steam System. Radioactivity contained in the reactor coolant would mix with water in the shell side of the affected steam generator. This radioactivity would be transported by steam to the turbine and then to the condenser or directly to the condenser via the Steam Bypass System. Non-condensable radioactive gases in the condenser are removed by the Main Condenser Evacuation System and discharged to the atmosphere. Analysis of a steam generator tube rupture accident, assuming complete severance of a tube, is presented in Section 15.6.

←(DRN 01-1283, R12)

Experience with nuclear steam generators indicates that the probability of complete severance of a tube is remote. The material used to fabricate the vertical U-tube is a Ni-Cr-Fe alloy. A double-ended rupture has never occurred in a steam generator of this design. The more probable modes of failure, which result in smaller penetrations, are those involving the occurrence of pinholes or small cracks in the tubes, and of cracks in the seal welds between the tubes and tube sheet. Detection and control of steam generator tube leakage is described in Subsection 5.2.5.

### 5.4.2.3.3 Composition of Secondary Fluid

→(DRN 00-1059, R11-A; 02-218, R11-A)

The concentration of radioactivity in the secondary side of the steam generators is dependent upon the concentration of radionuclides in the reactor coolant, the primary-to-secondary leak rate, and the rate of steam generator blowdown. The expected specific activities in the secondary side of the steam generators during periods of normal operation are given in Section 11.1. Activities are based on operations with 0.1 percent failed fuel cladding, a total primary-to-secondary leakage of 100 lbm./day, and 60 gpm blowdown

←(DRN 00-1059, R11-A; 02-218, R11-A)

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→(DRN 00-1059, R11-A; 02-218, R11-A)

rate to the Steam Generator Blowdown System. An evaluation of the shell side radioactivity is presented in Section 11.2. Limits for radioactivity levels in the secondary side of the steam generators and the bases for these limits are provided in Technical Specifications, Chapter 16.

←(DRN 00-1059, R11-A; 02-218, R11-A)

The recirculation water within the steam generators will contain volatile additives necessary for proper chemistry control. These and other chemistry considerations of the Main Steam System are discussed in Subsection 10.3.5.

### 5.4.2.4 Tests and Inspections

#### 5.4.2.4.1 Fabrication Tests and Inspections

The steam generator is tested in accordance with ASME Boiler and Pressure Vessel Code, Section III. The nondestructive tests, some of which are not required by the code, performed during fabrication are given in Table 5.4-3.

During design and fabrication of the steam generator, additional operations beyond the requirements of the ASME Boiler and Pressure Vessel Code, Section III, were performed by the vendor. These included ultrasonic testing for defects in tube sheet clad and ultrasonic testing of weld clad for bond integrity.

Initial hydrostatic tests of the primary and secondary sides of the steam generator are conducted in accordance with ASME Code, Section III. Leak tests are also performed. Following satisfactory performance of the hydrostatic tests, magnetic-particle inspections are made on all accessible welds.

Steam generator performance is further verified during the initial startup tests. Provisions for onsite cleaning and cleanliness control are described in Subsection 5.2.3.

In-service Inspection will be performed during plant life in accordance with the ASME Code, Section XI as discussed in Subsection 5.2.4.

→(EC-8458, R307)

Inservice inspection of steam generator tubing will comply with the Technical Specifications.

### 5.4.2 REFERENCES

1. WCAP-17263-P, Revision 0, November 2010, Regulatory Guide 1.121 Analysis and Structural Integrity Performance Criterion Applications for the Waterford Unit 3 Model Delta 110 Replacement Steam Generators for a NSSS Power of 1869.6 MWt/SG.

←(EC-8458, R307)

### 5.4.3 REACTOR COOLANT PIPING

#### 5.4.3.1 Design Basis

→(DRN 06-546, R15)

The reactor coolant loop piping is designed and analyzed for normal operation including all transients discussed in Subsection 3.9.1. Loading combinations and stress criteria associated with faulted conditions are presented in Subsection 3.9.3.1. In addition, certain nozzles are subjected to local transients that are included in the design analysis of the areas affected. Thermal sleeves are installed in the

←(DRN 06-546, R15)

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surge nozzle, safety injection nozzles, and charging nozzle to accommodate these additional transients. Principal parameters are listed in Table 5.4-4. The ASME code and addenda to which the piping is designed is specified in Subsection 5.2.1.

In addition to being specified as seismic Category I, the following additional vibratory requirement is specified in the engineering specification. The various piping assemblies are designed so that no damage to the equipment is caused by the frequency ranges of 19 to 20 Hz and 95 to 100 Hz. The reasons for selecting these frequencies are the same as described in Subsection 5.4.2.3.1.2 for the steam generators. Additional presentations relating to seismic and dynamic analysis and criteria for the reactor coolant piping are contained in Subsections 3.7.2 and 3.9.2, respectively.

### 5.4.3.2 Description

Each of the two heat transfer loops contains five sections of pipe: one 42 in. ID pipe between the reactor vessel outlet nozzle and steam generator inlet nozzle, two 30 in. ID pipes from the steam generator's two outlet nozzles to the reactor coolant pumps suction nozzle, and two 30 in. ID pipes from the pumps discharge nozzle to the reactor vessel inlet nozzles. These pipes are referred to as leg hot leg, the suction legs, and the cold legs, respectively. The other major pieces of reactor coolant piping are the surge line, a 12 in. schedule 160 pipe between the pressurizer and the hot leg in Loop 1, and the spray line, a 4 in. schedule 120 pipe at the pressurizer end reduced to a 3 in. schedule 160 pipe between the pressurizer and the cold legs of Loops 1A and 1B.

To minimize the possibility of stress corrosion cracking, the reactor coolant piping is fabricated from SA 516 GR 70 base material mill clad with type 304L stainless steel. Lines such as the surge lines, spray lines, and other small lines are totally made of stainless steel. Nozzles are shop fabricated with safe ends to preclude dissimilar-metal field welds.

→(EC-19087, R305)

Where stainless steel or Ni-Cr-Fe nozzle or safe end material is used, the safe ends are welded to the assembly after final stress relief to prevent furnace sensitization. Other precautions used in the shop and during field assembly of the piping are described in Subsection 5.2.3. Pressurizer and hot leg nozzle welds that contain Ni-Cr-Fe materials susceptible to primary water stress corrosion cracking (PWSCC) have received structural weld overlays using Alloy 52M weld material (EC1830).

←(EC-19087, R305)

→(DRN 03-1268, R13)

The 42 in. and 30 in. pipe diameters are selected to obtain coolant velocities that provide a reasonable balance between erosion-corrosion, pressure drop, and system volume. The surge line is sized to limit the frictional pressure loss through it during the maximum in-surge so that the maximum allowable pressure of the RCS is not exceeded. The spray line is sized to ensure a minimum spray flow of 375 gpm with both spray valves open. Pressurizer parameters are listed in Table 5.4-6.

←(DRN 03-1268, R13)

To reduce the amount of field welding during plant fabrication, the 42 in. and 30 in. pipes are supplied in major pieces, complete with shop-installed instrumentation nozzles and connecting nozzles to the auxiliary systems. Where required, the nozzles are supplied with safe ends to facilitate field welding of the connecting piping. To reduce thermal shock, thermal sleeves are provided for all nozzles two in. or greater in diameter where fluid enters the main piping from an auxiliary system during normal operation.

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Flow restricting orifices (7/32 in. diameter by one in. long) are provided in the nozzles for the RCS sampling lines, the reactor coolant pipe hot leg pressure measurement nozzles, the pressurizer level and pressure instrument lines, and the reactor coolant pump differential pressure instrument lines to limit flow in the event of a break downstream of a nozzle.

➔(DRN 04-292, R13; 05-892, R14)

←(DRN 04-292, R13; 05-892, R14)

### 5.4.3.3 Evaluation

It is demonstrated by analysis that the reactor coolant piping is adequate for all normal operating and transient conditions of Subsection 3.9.1.1. In addition, the fully assembled RCS is subjected to the required hydrostatic tests. Fracture toughness of the reactor coolant piping is discussed in Subsection 5.2.3.

During the design phase, every effort is made to displace the frequency of the RCS piping from driving frequencies of concern by proper location of piping spring characteristics. The dynamic effects of system operation are also considered and piping restraints sized accordingly.

A discussion of the radiological considerations for the reactor coolant piping is provided in Section 12.3.

### 5.4.3.4 Tests and Inspections

Prior to and during fabrication of the reactor coolant piping, nondestructive testing, based on the requirements of the ASME Code (see Table 5.2-1) is applied. Table 5.4-5 summarizes the component inspection program during fabrication and construction. Tests for RCS integrity following normal opening, modification, or repair are specified in the Technical Specifications. To ascertain the integrity of the piping during plant life, necessary in-service inspections required by Section XI of the ASME Boiler and Pressure Vessel are performed where required on the reactor coolant piping. To facilitate such inspections, longitudinal weld seams have been oriented at the 90 and 270 degree locations where feasible. Removable insulation is installed to assure access to the welds. Inservice inspection of the RCS piping is further discussed in Subsection 5.2.4.

### 5.4.4 MAIN STEAM LINE FLOW RESTRICTIONS

➔(EC-8458, R307)

Main steam line flow restrictors are not required in the main steam line between the steam generator and the main steam isolation valves. A flow venturi (24 in. throat diameter) acts as a flow restriction for main steam line breaks downstream of the venturi. Refer to Figure 10.2-4 for location of the flow venturi.

Note: See Section 5.4.2.2 for a description of integral flow restrictors.

←(EC-8458, R307)

### 5.4.5 MAIN STEAM LINE ISOLATION SYSTEM

The Main Steam Line Isolation System is discussed in Sections 7.3 and 10.3.

### 5.4.6 REACTOR CORE ISOLATION COOLING SYSTEM

This Subsection is not applicable to Waterford 3.

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### 5.4.7 RESIDUAL HEAT REMOVAL SYSTEM

See Subsection 9.3.6.

### 5.4.8 REACTOR WATER CLEANUP SYSTEM

This Subsection is not applicable to Waterford 3.

### 5.4.9 MAINSTEAM LINE AND FEEDWATER PIPING

The Main Steam System piping is discussed in Subsection 10.3.6; Feedwater System piping is described in Subsection 10.4.7.

### 5.4.10 PRESSURIZER

#### 5.4.10.1 Design Bases

→(DRN 06-546, R15)

The pressurizer is designed and analyzed for the transients specified in Subsection 3.9.1.1 and the following additional requirements. During heatup and cooldown of the plant, the allowable rate of temperature change for the pressurizer is increased to 200°F/hr.

←(DRN 06-546, R15)

The pressurizer is designed to:

- a) Maintain RCS operating pressure
  - b) Compensate for changes in coolant volume during load changes
  - c) Contain sufficient volume to prevent draining the pressurizer as a result of a reactor trip
  - d) Limit the water volume to minimize the energy release during LOCA
  - e) Prevent uncovering of the heaters by the out-surge of water following load decreases; 10 percent step decrease and five percent per minute ramp decrease
  - f) Provide sufficient volume to accept the reactor coolant insurge resulting from a loss of load without the water level reaching the safety valve nozzles
  - g) Provide sufficient volume to yield acceptable pressure response to normal system volume changes during load change transients
- (DRN 00-1059, R11-A)
- h) Achieve a total coolant volume change and associated charging and letdown flows which are as small as practical and are compatible with the capacities of the volume control tank, charging pumps and letdown control valves during load-following transients
- ←(DRN 00-1059, R11-A)
- i) Ensure that the minimum pressure observed during transients is above the setpoint of the safety injection actuation signal, and that the maximum pressure is below the high-pressure trip.

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The heater capacity is selected to provide an adequate pressurizer heatup rate during plant startup.

In addition to being specified as seismic Category I, the following additional vibratory requirement is specified in the engineering specification. The pressurizer vessel, including the heaters, baffles, and supports shall be designed such that no damage to the equipment is caused by the frequency ranges of 19-20 Hz and 95-100 Hz. The lower frequency is defined as a mechanical vibration. The design basis for the higher frequency consists of a pressure pulse of five psi which diminishes internally within the vessel.

### 5.4.10.2 System Description

The pressurizer is shown in Figure 5.4-6 and the design parameters are given in Table 5.4-6

The pressurizer is a cylindrical carbon steel vessel with stainless steel clad internal surfaces. A spray nozzle on the top head is used in conjunction with heaters in the bottom head to provide level and pressure control. Overpressure protection is provided by two safety valves. The pressurizer is supported by a cylindrical skirt welded to the bottom head.

The pressurizer is designed and fabricated in accordance with the ASME Code requirements listed in Table 5.2-1. The interior surface of the cylindrical shell and upper head is clad with weld deposited stainless steel. The lower head is clad with a Ni-Cr-Fe alloy to facilitate welding of the Ni-Cr-Fe alloy heater sleeves to the shell. A stainless steel safe end is provided on the pressurizer nozzles, after vessel final stress relief, to facilitate field welds to the stainless steel piping.

The total volume of the pressurizer is established by consideration of the factors given in Subsection 5.4.10.1. To account for these factors and to provide adequate margin at all power levels, the water volume in the pressurizer is programmed as a function of average coolant temperature as shown in Figure 5.4-7 in conjunction with Figure 5.4-8. High or low water level error signals result in the control actions shown in Figure 5.4-9.

Pressure is maintained by controlling the temperature of the saturated liquid volume in the pressurizer. At full load conditions, slightly more than one-half of the pressurizer volume is occupied by saturated water, and the remainder by saturated steam. In order to maintain the programmed pressure, the corresponding saturation temperature must be maintained. To maintain this temperature, heaters are energized to compensate for heat losses through the vessel and to raise the continuous subcooled pressurizer spray flow to the saturation temperature.

During load changes, the pressurizer limits pressure variations caused by expansion or contraction of the reactor coolant. The average reactor coolant temperature is programmed to vary as a function of load as shown in Figure 5.4-8. A reduction in load is followed by a decrease in the average reactor coolant temperature to the programmed value for the lower power level. The resulting contraction of the coolant lowers the pressurizer water level, causing the Reactor Coolant System pressure to decrease. This pressure reduction is partially compensated by flashing of pressurizer water into steam. All pressurizer heaters are automatically energized on low system pressure, generating steam and further limiting pressure decrease. Should the water level in the pressurizer drop sufficiently below its

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setpoint, the letdown control valves close to a minimum value, and additional charging pumps in the Chemical and Volume Control System (CVCS) are automatically started to add coolant to the system and restore pressurizer level.

When load is increased, the average reactor coolant temperature is raised in accordance with the coolant temperature program. The expanding coolant from the reactor coolant piping hot leg enters the bottom of the pressurizer through the surge line, compressing the steam and raising system pressure. The increase in pressure is moderated by the condensation of steam during compression and by the decrease in bulk temperature in the liquid phase. Should the pressure increase be large enough, the pressurizer spray valves open, spraying coolant from the reactor coolant pump discharge (cold leg) into the pressurizer steam space. The relatively cold spray water condenses some of the steam in the steam space, limiting the system pressure increase. The programmed pressurizer water level is a power dependent function. A high level error signal, produced by an insurge, causes the letdown control valves to open, releasing coolant to the CVCS and restoring the pressurizer to the programmed level. Small pressure and primary coolant volume variations are accommodated by the steam volume that absorbs flow into the pressurizer and by the water volume that allows flow out of the pressurizer.

The pressurizer heaters are single unit, direct immersion heaters that protrude vertically into the pressurizer through sleeves welded in the lower head. Each heater is internally restrained from high amplitude vibrations and can be individually removed for maintenance during plant shutdown.

Approximately one-fifth of the heaters are connected to proportional controllers that adjust the heat input as required to compensate for steady-state losses and to maintain the desired steam pressure in the pressurizer. The remaining backup heaters are connected to on-off controllers. These heaters, normally deenergized, are turned on by either a low-pressurizer pressure signal or high-level error signal. This latter feature is provided since load increases result in an in-surge of relatively cold coolant into the pressurizer, thereby decreasing the bulk water temperature. The CVCS acts to restore level, resulting in a transient pressure below normal operating pressure. To minimize the extent of this transient, the backup heaters are energized, contributing more heat to the water. An interlock prevents operation of the backup heaters in the event of concurrent high level error and high-pressurizer pressure signals. A low-low pressurizer level signal deenergizes all heaters to protect the heaters should they become uncovered.

→(LBDCR 15-028, R308A)

In order to determine the pressurizer heater capacity required to maintain natural circulation in the hot standby condition after a loss of offsite power, it was conservatively assumed that the ambient heat loss rate through the pressurizer was 400,000 BTU/hr. The measured heat loss from startup testing was only 356,000 BTU/hr. With an assumed 400,000 BTU/hr heat loss and a safety valve leakage of up to 0.5 gpm, single phase natural circulation can be maintained at hot standby conditions with a 50°F subcooled margin indefinitely by energizing 150kW of heater capacity thirty minutes after the loss of offsite power. Loss of subcooling, however, does not imply loss of natural circulation. The natural circulation cooldown analysis (refer to FSAR Section 9.3.6.3.3.1), performed to comply with Branch Technical Position 5-4, Design Requirements of the Residual Heat Removal System, does not credit the operation of any pressurizer heaters. Therefore, the operator action to energize the Pressurizer Heaters is not a time critical operator action)

←(LBDCR 15-028, R308A)

→(DRN 00-1673)

A redundant group of pressurizer proportional heaters and three redundant groups of backup heaters have been made available to be placed manually on the emergency diesel generator after a loss of offsite power. Each bank of heaters has access to only one Class 1E division power supply.

←(DRN 00-1673)

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→(LBDCR 15-028, R308A)

Part of the closing circuitry to the breakers that provide power to the 480V non-safety switchgear buses 3A32 and 3B32 (that power the Pressurizer Heaters) share a specific common circuit breaker, CVCEBKR014AB-13. CVCEBKR014AB-13 powers the interlock 52z relay, SSDEREL2348-D (SSDEREL2398-D). The interlock 52z relay checks for completion of load stripping on the respective 480V non-safety switchgear buses 3A32 (3B32) at the onset of a Loss of Offsite Power. If the load stripping is complete, the interlock 52z relay closes a contact in the closing circuitry to the breakers that provide power to the 480V non-safety switchgear buses 3A32 and 3B32 to allow the breakers to close automatically when the sequencer load block contact in the closing circuitry is closed.

Alternatively, if the specific common circuit breaker, CVCEBKR014AB-13, is Open, then the breakers that provide that power to the 480V non-safety switchgear buses 3A32 and 3B32 will not close automatically at the onset of a Loss of Offsite Power. To close the breakers that power each Pressurizer Heater electrical switchgear 3A32 (3B32), local manual action in the respective train Switchgear room is necessary.

Reenergization of the necessary heaters from the emergency onsite power can be accomplished manually from the control room. At the onset of a Loss of Offsite Power concurrent with the specific common circuit breaker, CVCEBKR014AB-13, being Open, the reenergization of the 480V non-safety switchgear buses 3A32 and 3B32 (that power the Pressurizer Heaters) will require action to be performed outside of the Control Room. To close the breakers that power the 480V nonsafety switchgear buses 3A32 and 3B32, local manual operator action in the respective train Switchgear room is necessary. Once each 32 switchgear bus is reenergized, the necessary Pressurizer Heaters powered from that bus can be reenergized from the Control Room.

The natural circulation cooldown analysis (refer to FSAR Section 9.3.6.3.3.1), performed to comply with Branch Technical Position 5-4, Design Requirements of the Residual Heat Removal System, does not credit the operation of any pressurizer heaters. Therefore, the operator action to close the breakers that power each Pressurizer Heater electrical switchgear 3A32 (3B32), located outside of the control room, is not a time critical operator action. Procedures ensure that the addition of these loads after a loss of offsite power will not exceed the rating of the emergency diesel generator. The heaters are powered from the 480V non-safety switchgear buses 3A32 and 3B32. The safety-related Class 1E breakers providing power to these buses from the 4.16kV ESF buses (3A3-S and 3B3-S) will trip upon LOOP or SIAS. In this manner, the Class 1E interfaces for main power and control power to the pressurizer heaters are protected by safety-grade circuit breakers. This scheme also ensures that in case of an SIAS the non-Class 1E pressurizer heaters are shed from their emergency power sources via Class 1E circuit breakers.

←(LBDCR 15-028, R308A)

Pressurizer spray is supplied from each of the reactor coolant pump cold legs in loop one to the pressurizer spray nozzle. Automatic spray control valves control the amount of spray as a function of pressurizer pressure; both of the spray control valves function in response to the signal from the controller. These components are sized to use the differential pressure between the pump discharge and the pressurizer to pass the amount of spray required to maintain the pressurizer steam pressure during normal load following transients. A small continuous flow is maintained through the spray lines at all times to keep the spray lines and the surge line warm to reduce thermal shock during plant transients. This continuous flow also serves to keep the chemistry and boric acid concentration of the pressurizer water the same as that of the coolant in the heat transfer loops.

An auxiliary spray line is provided from the charging pumps to permit pressurizer spray during plant heatup, or to allow depressurization and cooling if the reactor coolant pumps are shut down. The capability to depressurize using auxiliary spray with one charging isolation valve failed open was demonstrated by testing. With a charging to cold leg isolation valve open, some of the charging pump flow is diverted from the auxiliary sprayflow path to the cold legs. With two charging pumps running (88 gpm) an auxiliary spray flow rate of 37 gpm was achieved. This resulted in a depressurization rate of 24 psi/min. This is sufficient to depressurize the RCS during any design basis accident where auxiliary spray is required



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In the event of an abnormal transient that causes a sustained increase in pressurizer pressure at a rate exceeding the control capacity of the spray, a high-pressurizer pressure reactor trip will be initiated.

### 5.4.10.3 Safety Evaluation

It is shown by analysis made in accordance with the requirements of the ASME Code, Section III that the pressurizer is adequate for all normal operating and transient conditions expected during the life of the plant. Following fabrication, the pressurizer was hydrostatically and nondestructively tested in accordance with the ASME Code, Section III.

During hot functional testing, the transient performance of the pressurizer is checked by determining its normal heat losses and maximum pressurization and depressurization rates. This information is used in setting the pressure controllers.

Overpressure protection of the RCS is provided by two ASME Code spring loaded safety valves.

A discussion of the radiological considerations for the pressurizer is provided in Section 12.3.

### 5.4.10.4 Inspection Testing and Requirements

Prior to and during fabrication of the pressurizer, nondestructive testing is performed in accordance with the requirements of the ASME Code Section III. The pressurizer inspection program is summarized in Table 5.4-7.

Further assurance of the structural integrity of the pressurizer during plant life will, be obtained from the inservice inspection performed in accordance with the ASME Code, Section XI and described in Section 5.2.

## 5.4.11 QUENCH TANK (PRESSURIZER RELIEF TANK)

### 5.4.11.1 Design Basis

The quench tank is designed to receive and condense the normal discharges from the primary (pressurizer) safety valves and to prevent the discharge from being released to containment.

→(DRN 03-2059; 05-316, R14; EC-8458, R307)

The tank is sized to receive and condense a total steam release of 1232 lbm. The maximum normal discharge that the quench tank must withstand occurs during the loss of condenser vacuum event (described in Section 15.2.1), which is approximately 1200 lbm.

←(DRN 03-2059; 05-316, R14; EC-8458, R307)

The quench tank is mounted on structural steel framing supported from the secondary shield wall.

### 5.4.11.2 System Description

The quench tank, shown in Figure 5.4-10, is an austenitic stainless steel vessel suitable for prolonged contact with borated, demineralized water. Nozzles are provided for the safety valve discharge line, vents, drains, instrumentation, makeup water, nitrogen addition, and the rupture disc. The tank prevents the steam released from the primary (pressurizer) safety valves from being released to the containment atmosphere. The steam is discharged under water by the sparger and condensed. Demineralized water is manually added from the Reactor Auxiliary Building to cool the tank water after a steam discharge. A rupture disc venting to the containment atmosphere is provided for overpressure protection. Noncondensable gases within the quench tank are vented to the containment vent header through an air-operated valve.

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The tank is designed and fabricated in accordance with the ASME Code, Section VIII. The design parameters are given in Table 5.4-8.

The sparger, spray header, nozzles and rupture disc fittings are stainless steel.

### 5.4.11.3 Safety Evaluation

The quench tank and associated blowdown system are sized such that the maximum safety valve backpressure, 500 psig, is not reached during any anticipated transient steam release. The quench tank rupture disc has a relief capacity greater than the combined relief capacities of the primary (pressurizer) safety valves.

The quench tank is classified as non-nuclear safety class as defined in Section 3.2. The failure of any of these components in no way compromises the integrity of the RCS pressure boundary or safe shutdown of the plant, nor does it in any way jeopardize the safety of the public.

### 5.4.11.4 Instrumentation Requirements

The quench tank is equipped with level, pressure, and temperature instrumentation.

A brief explanation of each instrument is made below.

The location of each of the sensors and functional requirements of the instruments are indicated on Figure 5.1-3.

#### 5.4.11.4.1 Temperature

The temperature measurement channel consists of a precision resistance temperature detector (RTD), a temperature transmitter and associated indicators and alarms. The resistance of the RTD varies as a function of the temperature of the RTD environment. This change in RTD resistance is sensed by the temperature transmitter and converted to a change in a low dc current, which is used as a signal to the remote temperature indicators and alarms.

Quench tank temperature indication and high temperature alarm are provided in the main control room. A high temperature alarm may be indicative of primary safety valve leakage to the tank.

#### 5.4.11.4.2 Level

Level instrumentation consists of a level transmitter and associated indicating and alarming equipment. The level transmitter measures the pressure difference between a reference column of water and the vessel water level. This pressure difference is converted to a small dc current which is proportional to the level of water in the vessel. This dc current output is used as a signal to the remote level indicators and alarms.

Quench tank level indication and high and low level alarms are provided in the main control room. A low level alarm is indicative of the sparger being uncovered or of insufficient tank fluid volume to quench the design basis accident steam release. A high level alarm alerts the operator of insufficient volume in the quench tank to accept the pressurizer steam release without becoming overpressurized and causing the rupture disc to burst.

#### 5.4.11.4.3 Pressure

A pressure measurement instrument consists of an electric force balance pressure transmitter and associated indicators and alarms. The transmitter produces dc current output that is proportional to the pressure sensed by the instrument. This dc current output is used as a signal to the remote pressure indicators and alarms.

Quench tank pressure indication and high pressure alarm are provided in the main control room. The high pressure alarm alerts the operator to the situation prior to the bursting of the rupture disc.

5.4.11.5                    Inspection and Testing Requirements

The quench tank is designed to handle the design basis steam releases as described in Subsection 5.4.11.2. However, should the rupture disc burst, the primary coolant released to the containment would be minimal when compared to the LOCA for which the engineered safety features are designed to accommodate. Therefore, no inspection or testing requirements are imposed on the quench tank.

Since, the entire system is located within the containment structure, the quench tank must be vented to containment atmosphere, during the containment leakage test, to avoid collapse of the tank.

5.4.12                    VALVES

5.4.12.1                Design Basis

The safety-related functions of valves within the reactor coolant pressure boundary are to act as pressure retaining vessels and leaktight barriers during normal plant operation, accidents and seismic disturbances.

These valves are designed in accordance with the applicable ASME Code, Section III, or the Draft ASME Code for Pumps and Valves, Class I requirements and must withstand the effects of the system design transients (see Subsection 3.9.1.1) plus other transients associated with the valves location or service requirements. The valves meet seismic Category I requirements. Backseats are specified on manual and motor-operated gate and globe valves to further minimize potential leakage. Functional requirements for each valve are detailed in the individual valve specifications.

Materials of construction are specified to assure compatibility with the environment and contained fluids.

5.4.12.2                Design Description

→(DRN 99-811)

All valves in the Reactor Coolant System are constructed primarily of stainless steel. Other materials in contact with the coolant, such as hard facing and packing, are compatible materials. Fasteners, packing gland assemblies, and yoke fasteners are also constructed of stainless steel to eliminate corrosion problems. Valve packing glands have provisions to adjust packing compression to reduce or eliminate leakage. These features keep uncontrolled leakage from these valves at essentially zero.

←(DRN 99-811)

5.4.12.3                Design Evaluation

All valves within the reactor coolant pressure boundary are stress analyzed in accordance with the applicable ASME Code, taking into consideration cyclic loadings. Refer to Section 3.9.

5.4.12.4 Tests and Inspections

The valves are hydrostatically tested and leak tested across the seats and across the packing in accordance with individual valve specifications and the applicable ASME Code. Valves greater than 1 inch inner diameter are dimensionally checked, including measurements to determine wall thickness.

In-service inspection will be performed during the plant life in accordance with ASME Code Section XI, as discussed in Subsection 5.2.4.

## 5.4.13 SAFETY AND RELIEF VALVES

5.4.13.1 Design Basis

There are no power-operated relief valves in the RCS. The primary safety valves on the pressurizer are designed to protect the system, as required by the ASME Code, Section III.

The design basis for establishing the relieving capacity of the primary safety valves is presented in Appendix 5.2A. For the postulated transients presented in Chapter 15, the results demonstrate that the pressurizer will not go "solid" and that the relieving capacity of the safety valves is sufficient to provide overpressure protection in accordance with Section III of the ASME Code.

Safety valves on the steam side of each steam generator are designed to protect the steam system, as required by the ASME Code, Section III. They are conservatively sized to pass a steady flow equivalent to the maximum expected power level at the design pressure of the Main Steam System.

5.4.13.2 Description

The RCS has two safety valves to provide overpressure protection. The safety valve is illustrated in Figure 5.4-11. The design parameters are given in Table 5.4-9. They are direct acting, spring-loaded safety valves meeting ASME Code requirements. They have an enclosed bonnet and have a balanced bellows for superimposed backpressure. The safety valves pass sufficient pressurizer steam to limit the RCS pressure to 2750 psia (110 percent of design) following a complete loss of turbine generator load without simultaneous reactor trip. A delayed reactor trip is assumed on a high-pressurizer pressure signal. To determine maximum steam flow through the primary safety valves, the main steam safety valves are assumed to be operational. Values for the system parameters, delay times, and core moderator coefficient are given in Chapter 15.

Overpressure protection for the shell side of the steam generators and the main steam line up to the inlet of the turbine stop valve is provided by the secondary safety valves.

5.4.13.3 Evaluation

Overpressure protection is discussed in Subsection 5.2.2. The ASME Code Report on Overpressure Protection, is included as Appendix 5.2A.

→(DRN 00-1059; 02-88)

←(DRN 00-1059; 02-88)

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→(DRN 00-1059, R11-A; 02-88, R11-A)

### 5.4.13.4 Tests and Inspections

←(DRN 00-1059, R11-A; 02-88, R11-A)

The valves are inspected during fabrication in accordance with ASME Code, Section III requirements. The inlet is hydrostatically tested and the outlet is pneumatically tested. Seat leakage is checked by a steam test at 93 percent of set pressure. Set pressures are adjusted using steam.

In-service inspection will be performed during plant life in accordance with the ASME Code Section XI as discussed in Subsection 5.2.4.

## 5.4.14 COMPONENTS SUPPORT

→(DRN 03-2059, R14)

### 5.4.14.1 Design Bases

The criteria applied in the design of the RCS supports are that the specific function of the supported equipment be achieved during all normal earthquake, and pipe break conditions. Specifically, the supports are designed in accordance with the design loading combinations which are applied in the design of ASME Code Class I supports. These design loading combinations are categorized with respect to their plant operating conditions, which are identified as normal, upset, emergency and faulted (as defined in the ASME Code, Section III, for Class I components). The following design loading combinations are applied:

←(DRN 03-2059, R14)

#### a) Loading Combination 1 (Upset)

The concurrent loadings associated with either the normal plant condition or the upset plant condition and the vibratory motion of the operational basis earthquake (OBE).

#### b) Loading Combination 2 (Emergency)

The concurrent loadings associated with the plant emergency condition.

#### c) Loading Combination 3 (Faulted)

→(DRN 03-2059, R14)

The combined loadings associated with the normal plant condition, the vibratory motion of the SSE, and the dynamic system loadings associated with postulated pipe ruptures as discussed in Subsections 3.6.2, 3.6.3, and 6.2.1.2. This loading combination is assumed to occur at 100 percent power steady state operation.

←(DRN 03-2059, R14)

For supports that are an integral part of RCS components, the design stress limits of the parent component are used as follows:

- a) The design stress limits applied in conjunction with loading combinations 1 and 2 are in accordance with the rules of ASME Code Section III, Paragraph NB-3223 for upset conditions and Paragraph NB-3224 for emergency conditions, respectively.

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Nonintegral supports are designed and constructed to the stress limits of ASME Section III, Subsection NF. The design stress limits used are as follows:

- a) The design stress limits applied in conjunction with loading combinations 1 and 2 for plate type supports and linear type supports are in accordance with the rules of Subsubarticles NF-3220 and NF-3230, respectively.

For these supports the design stress limits applied in conjunction with loading combination 3 are dependent upon the type of system or subsystem analyses used to establish design loadings. Elastic system or subsystem analysis used to determine the design loadings produced by loading combination 3 and the associated design stress limits applied in the design of supports are in accordance with the rules of the ASME Code, Section III, Appendix F, Subsubarticle F-1323, or plastic instability load criteria are applied in accordance with paragraphs F-1370 (d) and F-1370 (e) of Appendix F, including the effects of the resulting plastic deformation.

### 5.4.14.2 Description

Figure 5.4-12 illustrates the RCS support points. A description of each supported component follows.

#### 5.4.14.2.1 Reactor Vessel Supports

The reactor vessel integral supports consist of four pads welded to the underside of the vessel inlet nozzles. Vertically, the pads rest on lubricated bearing plates, and contain studs that act as holddown devices for the vessel. Horizontal keyways interface with the pads. The arrangement of the vessel supports allows radial growth of the reactor vessel due to thermal expansion while maintaining it centered. The supports are designed to accept normal loads and seismic and pipe rupture accident loads.

Reactor vessel supports are shown in Figure 5.4-13 and Figure 3.8-34.

#### 5.4.14.2.2 Steam Generator Supports

→(DRN 00-1059, R11-A; 03-2059, R14)

The steam generator weight is supported at the bottom by a sliding base bolted to an integrally attached conical skirt. The sliding base rests on low friction bearings which allow unrestrained thermal expansion of the RCS. Two embedded keys mate with keyways within the sliding base to guide the movement of the steam generator during expansion and contraction of the RCS and, together with anchor bolts, prevent excessive movement of the bottom of the steam generator during seismic events and following a pipe break.

←(DRN 00-1059, R11-A; 03-2059, R14)

A system of keys and snubbers located on the steam drum guide the top of the steam generator during expansion and contraction of the RCS and provide support during seismic events and following a primary side or secondary side pipe break.

→(EC-8438, R307)

The mechanical/structural loads associated with the dynamic effects of a LOCA in the RCS hot leg and cold leg piping have been eliminated with the application of Leak-Before-Break (LBB) (Reference Section 3.6.3). Based on removal of the RCS dynamic pipe break loads, the shim plate pack (stop) on the reactor side of the keyway (underneath the RCS hot leg) was permanently removed from the SG-1 and SG-2 sliding base. Additionally, the SG-1 and SG-2 sliding base supports were modified to remove the shim plate from the perimeter of the SG support skirt flange. These modifications were performed as part of the changeout of the steam generators.

←(EC-8438, R307)

Steam generator supports are shown in Figure 5.4-14.

#### 5.4.14.2.3 Reactor Coolant Pump Supports

→(DRN 03-2059, R14)

Each reactor coolant pump is provided with four vertical support columns, four horizontal support columns, and one horizontal snubber. The rigid structural columns provide support for the pumps during normal operation, earthquake conditions, and any postulated pipe break in either the pump suction or discharge line<sup>2</sup>. An illustration of the pump supports is shown in Figure 5.4-15.

←(DRN 03-2059, R14)

For the case of a pipe break in the pump discharge line, three structural stops are provided to limit the pump motion. Pipe stop structures that limit pipe motion are also provided to prevent overloading of the pump support columns due to a pipe rupture at either the steam generator or reactor vessel nozzles.

#### 5.4.14.2.4 Pressurizer Supports

→(DRN 03-2059, R14)

The pressurizer is supported by a cylindrical skirt welded to the bottom of the pressurizer and bolted to the support structure. The skirt is designed to withstand dead weight and normal operating loads as well as the loads due to earthquakes and postulated pipe break. An illustration of the pressurizer supports is shown in Figure 5.4-16.

#### 5.4.14.3 Evaluation

The structural integrity of the reactor coolant component supports is confirmed by analyses, using the design basis presented in Subsection 5.4.14.1. The analyses employ dead weight, thermal, seismic and pipe break loadings, combined in the various load combinations. Dead weight and thermal loads are determined by static analysis of the RCS. The method of determining seismic loadings is described in Subsection 3.7.2. RCS response to postulated branch line pipe breaks (BLPBs) for extended power uprate to 3716 MWt is determined using non-linear time history analysis of a full three-dimensional ANSYS model of the RCS. The effects of BLPBs on the RCS component supports include the effects of pipe tension release, jet impingement, RV blowdown, SG subcompartment and component-to-support gaps. Subsequent evaluations demonstrate the adequacy of the RCS supports under extended power uprate conditions to 3716 MWt, per the acceptance criteria provided in Section 3.9.3.

Structures are provided to mate with the component supports to restrain and support RCS components. The loads at the support/structure interface locations are examined under normal, OBE, SSE, and pipe break conditions (see Section 3.8). Seismic and accident loads are determined by the same methods referred to in the first paragraph of this section, taking into account the structural characteristics at the support/structure interfaces. The design of each support is compatible with the design radiation levels given in Section 3.11.

←(DRN 03-2059, R14)

#### 5.4.14.4 Testing and Inspection

Tests were conducted on materials similar to that being used for the reactor vessel and steam generator sliding supports to demonstrate that the maximum static coefficient of friction does not exceed 0.15 at a design loading of 5000 psi. Tests on sliding supports and the steam generator basis are in accordance with ASME Section III. In addition, all sliding supports are 100 percent liquid penetrant inspected at the vendor's shop. The steam generator base is ultrasonically and magnetic particle inspected.

The specifications for the steam generator snubbers require that they be tested in the vendor's shop at the rated load capacity in both tension and compression. The piston creep during these tests must comply with specification limits. Tests are also specified for initiation of snubber action in both tension and compression.

→(DRN 03-2059, R14)

<sup>2</sup>These pipe breaks, which have been eliminated via LBB (Section 3.6.3) and which have been replaced by postulated BLPBs for extended power uprate to 3716 MWt, continue to provide the enveloping design loads on the RCP supports.

←(DRN 03-2059, R14)

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Supports integral to the RCS components receive quality assurance inspections in accordance with the ASME Code, Section III, during fabrication.

During preoperational testing of the RCS, the support displacements will be monitored for agreement with calculated displacements and/or clearances. Subsequent examinations of supports of RCS components will be in accordance with the ASME Code, Section XI.

### 5.4.15 REACTOR COOLANT GAS VENTING SYSTEM

#### 5.4.15.1 Design Bases

The Reactor Coolant Gas Venting System (RCGVS) is designed to allow for remote venting of non-condensable gases, which may collect in the RCS, via the reactor vessel head vent or pressurizer steam space vent during post-accident situations. The system may be used for normal RCS venting when required during plant outages. This system has been designed to meet the requirements of NUREG-0737.

The design criteria for the RCGVS are as follows:

- a) The system permits remote (control room) venting of the reactor vessel head or the pressurizer.
- b) The system is designed for a single active failure with active components powered from their respective redundant emergency power sources. The system has parallel vent paths with valves powered from alternate power sources. A single active failure in the power and control portion of the vent system will not prevent isolation of the entire vent system when required.
- c) The system is designed primarily to vent noncondensable gas.
- d) The vent flow capability is based upon the following considerations:
  - 1) The vent rate is sufficient to vent 4800 standard cubic feet of hydrogen per minute.
  - 2) Coolant liquid loss through the vent will not exceed the makeup capacity of one charging pump in the event of a Safety Class 2 pipe break or inadvertent valve operation, thus limiting leakage to less than the LOCA definition of 10CFR50.
  - 3) The vent mass rate will not result in heat loss from the RCS in excess of the normal pressurizer heater capacity.
- e) Vent paths are provided to the quench tank which allows for cooling of gases and condensing water vapor by releasing the vented gases-below the water level in the tank. The vent paths are safety grade and, being a part of the RCS, meet the same qualifications as the existing RCS.
- f) A method of leakage detection is provided to identify and ensure that leakage in the



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system is identifiable. This allows continued power operations at leak rates greater than 1 gpm but less than 10 gpm (refer to plant Technical Specifications).

- g) The solenoid operated valves are powered from safety grade 120V ac power supplies. Power is removed from the fail closed valves by utilizing key-locked control switches to minimize the possibility of inadvertent operation during normal operation.
- h) Valve position indication (open-closed) is provided in the control room for all remotely operated valves.
- i) The system is designed so that each vent valve may be tested for operability during plant operation. Testing can be performed in accordance with Subsection IWV of Section XI of the ASME code for Category B valves.

See Figure 5.4-17 for flow diagram arrangement.

### 5.4.15.2 Description/Principal Modes of Operation

#### 5.4.15.2.1 Startup

Venting of the RCS prior to plant startup can be accomplished using the RCGVS. The reactor vessel can be vented by opening either 2RC-E2559A or 2RC-E2560B and the pressurizer can be vented by opening either 2RC-E2557A or 2RC-E2558B. Venting is routed to the quench tank. Air flow can then be routed to the Waste Management System via the containment vent header.

#### 5.4.15.2.2 Normal Operation

This system is not intended for use during normal power operation and administrative controls are provided to minimize the possibility of inadvertent operation. In addition, power is removed from all valves during normal plant operation.

During normal operation, leakage detection is maintained by use of pressure instrumentation. A rise in pressure will indicate leakage past valves 2RC-E2557A, 2RC-E2558B, 2RC-E2559A or 2RC-E2560B. The ability to identify leakage above 1 gpm from the RCS may be accomplished in using either of two methods:

- a) If the pressure increase is slow enough, the leakage rate can be determined by observing the rate of pressure increase per unit time.
- b) The leakage would be diverted to the quench tank and the increase in quench tank level can be translated into a leakage rate.

#### 5.4.15.2.3 Refueling Shutdown

→(EC-5000080002, R301)

- a) During refueling shutdown, valves 2RC-E2557A and/or 2RC-E2558B, 2RC-E2559A and/or 2RC-E2560B are opened to align the reactor vessel to the pressurizer which is vented to atmosphere. Opening 1RC-V2590 A/B (RC-10111) and 1RC-V2506 (RC-101) provides an alternate vessel vent path. Either path prevents vacuum formation during reduction of reactor vessel level.

←(EC-5000080002, R301)

- b) During shutdown for refueling, valves 2RC-E2557A, 2RC-E2558B, 2RC-E2559A, 2RC-E2560B, 2RC-2561A and 2RC-E2562B are to be tested per In-service Testing requirements.

5.4.15.2.4 Accident Conditions

Prior to system operation, the quantity of non-condensable gases in the RCS can be estimated. This can be accomplished through evaluation of the inadequate core cooling instrumentation that will be provided in accordance with the requirements of NUREG-0737 II.F.2 (Section 1.9.30) for the case of gas in the reactor vessel and by the response of system pressure control methods or departure from saturation conditions in the pressurizer for the case of gas in the pressurizer. Additional indirect methods involving observing pressurizer level response to a given pressure change can be used as a backup method of gas volume determination.

The vent system is aligned to vent from the reactor vessel or pressurizer to the quench tank. Small quantities of gas may be vented to the quench tank without rupturing the quench tank rupture disc. This permits gas removal from the RCS without contaminating the containment atmosphere. Gas may be discharged from the quench tank to the Waste Management System.

5.4.15.3 Evaluation

The RCGVS may be required to operate during post-accident situations to remove noncondensable gases from the RCS. To assure operability under those conditions, the components of the system required to perform venting operations have been environmentally qualified to operate under post-accident containment conditions. They are provided with emergency power and parallel valves powered from redundant power sources. Parallel valves assure a vent flow path to containment in the event of single active failure.

The RCGVS is supplied with flow limiting orifices which limit mass loss from the RCS to an amount less than the makeup capacity of a charging pump.

The components, piping and supports in the RCGVS are specified as seismic Category I. All valves have been qualified for operability during and following a seismic event.

5.4.15.4 Required Components

a) Piping and Valves

All piping and valves are constructed of austenitic stainless steel and are safety grade as required. The piping size is 1 inch Schedule 160. The six Safety Class 2 solenoid operated valves are designed to fail closed to minimize inadvertent operation. The Safety Class I orifices, one of which is included in the present RCS design, are sized to meet the specified flow requirements.

b) Instrumentation and Controls

The system is designed to be controlled remotely from the main control room. All safety-related instrumentation is powered from emergency power sources. Position indication (open/closed) is provided for all remotely operated valves and displayed in the control room. Pressure instrumentation is also provided to monitor system performance and any valve leakage. Pressure indication is located in the control room. Quench tank level is also located in the control room.

## WSES-FSAR-UNIT-3

### 5.4.16 REFUELING WATER LEVEL INDICATING SYSTEM (RWLIS)

#### 5.4.16.1 Design Bases

→(DRN 00-1059)

The REFUELING WATER LEVEL INDICATING SYSTEM (RWLIS) is designed to perform the following functions.

←(DRN 00-1059)

- a) Monitor the water level in the Reactor Coolant System and the refueling pool during refueling operations.
- b) Monitor the water level in the RCS hot leg during maintenance evolutions requiring the RCS water levels at elevations within the range of the hot leg.
- c) Provide indication of the level locally in the containment building and remotely in the control room.
- d) Provide an alarm in the control room when the water level in the refueling pool exceeds a predetermined level or when the water level drops below a predetermined level in the range of the hot leg.

Refer to Figure 5.4-18 for a diagram of the system showing its connection to the RCS and the major components.

The design pressure and temperature of the RWLIS are 100 psig and 250°F. With the design conditions, the system's normal operational conditions are modes 5 and 6 with the RCS vented and pressurizer level greater than 5% and less than 75% as indicated by the pressurizer cold calibrated level instrumentation. The system is isolated from the RCS during plant operations. The RWLIS can be operated with the vessel head in place.

#### 5.4.16.2 System Description

The RWLIS is comprised of a narrow and a wide range differential pressure transmitter attached through stainless steel piping to a primary system high point near the top of the pressurizer and to the hot leg drain on the RCS (see Figure 5.4-18). The signals from these transmitters are provided to a local indicator inside containment, and to a remote indicator in the control room.

The RWLIS is designed to operate with the reactor vessel head installed or removed. With the exception of the local RWLIS indicator in containment, the system is designed for permanent installation. The RWLIS will compensate for slight positive and negative pressure variations in the RCS by utilizing a differential level instrument arrangement to compare actual RCS water level to a reference leg vented to an optimum system high point.

The low level sensing line is permanently attached to the drain line just below hot leg #1 in the vicinity of the shutdown cooling suction line.

The reference leg is permanently attached to the existing upper level tap of the pressurizer wide range level instrument.

## WSES-FSAR-UNIT-3

The transmitters and the local indicator are mounted on an instrument stand outside the biological shield. The location was chosen to minimize tubing runs and to allow easy draining of the reference leg to existing RCS level drain pipe to maintain the reference leg dry during normal RWLIS operation.

The level signals are conducted to the control room panel and annunciator window. The RWLIS signals are also provided to the Plant Monitoring Computer.

### 5.4.16.3 Evaluation

The valves which maintain the systems' isolation during plant operation are designed Class I and Class II for the isolation at the hot leg and the pressurizer, respectively, and are part of the Reactor Coolant Pressure Boundary. These valves are discussed in Subsection 5.4.12. The remainder of the RWLIS is installed as Class III. The tube routing and instrument stands are installed as Seismic Class 1. The RWLIS is not installed as safety-related instrumentation.

### 5.4.16.4 Tests and Inspections

Inservice inspection of the boundary valves is performed during the plant life in accordance with ASME Section XI as discussed in Subsection 5.2.4.

Operating procedures for the RWLIS require that the system be calibrated prior to draindown to ensure accurate level indication.

## 5.4.17 REACTOR COOLANT SHUTDOWN LEVEL MEASUREMENT SYSTEM (RCSLMS)

### 5.4.17.1 Design Bases

The Reactor Coolant Shutdown Level Measurement System (RCSLMS) is designed to perform the following functions:

- a) Monitor the water level in the Reactor Coolant System (RCS) during non-power operation.
- b) Monitor the water level in the RCS hot leg during maintenance evolutions requiring the RCS water levels at elevations within the range of the hot leg.
- c) Provide indication of the level locally in the containment building and remotely in the control room.
- d) Provide an alarm in the control room when water level drops below a predetermined level in the range of the hot leg.

Refer to Figure 5.4-19 for a diagram of the system showing its connection to the RCS and the major components.

The design pressure and temperature of the RCSLMS are 100 psig and 300°F. With the design conditions, the system's normal operational conditions are modes 5 and 6 with the RCS vented and pressurizer level greater than 5% and less than 75% as indicated by the pressurizer cold

calibrated level instrumentation. The system is isolated from the RCS during plant operations by removing the temporary connections.

#### 5.4.17.2 System Description

→(DRN 01-3975)

The RCSLMS is comprised of the Mansell Level Monitoring System (level instrument) and a skid mounted sight glass. This RCSLMS is connected by stainless steel tubing to a primary point at the top of the pressurizer and to the hot legs on the RCS (see Figure 5.4-19). The sight glass provides local indication. The Level instrument is installed on a temporary basis during refueling operations. The level instrument provides indication to the existing control room annunciators and indicators. Indication also exists on the level instrument computer system that is temporarily mounted in the Control Room during refueling operations to provide for system status of the instrument and level of the RCS.

→(DRN 00-1059)

The level instrument is comprised of two channels, each containing a reference (low-pressure) transducer assembly and a high-pressure transducer assembly that provides absolute pressure measurements to the computer system. The computer system computes the level of the RCS and displays the level in the control room. The computer system has a channel selectable output of 0-10VDC to the RCSLMS process analog control cards for conditioning and transmission of the signal to the control room indicators.

←(DRN 00-1059; 01-3975)

The RCSLMS is designed to function during non-power operations, providing a redundant measurement of RCS level. The system is designed to have temporary connections which are made only during non-power operation. The system is isolated and disconnected during normal operation.

#### 5.4.17.3 Evaluation

During power operation, the RCSLMS is disconnected, providing isolation from the RCS pressure boundary at the hot leg and pressurizer. The temporary connections are made after the reactor pressure has dropped below 300 psig in a section of piping which is Cat. 7 NNS and has valving to provide isolation for making the connections. These valves are part of Reactor Coolant Pressure Boundary and are discussed in Subsection 5.4.12. The tubing, tube supports and instrument skid are installed as Seismic Class 1. The RCSLMS is installed as non-safety-related instrumentation.

#### 5.4.17.4 Tests and Inspections

Visual inspection of welds and hydrostatic testing of the added piping and valves shall be performed in accordance with ANSI B31.1-1977.

Operating procedures for the RCSLMS require that the system be calibrated prior to dwdrain to ensure accurate level indication.

REACTOR COOLANT PUMP PARAMETERS

<u>Parameter</u>	<u>Value</u>
Number of Pumps	4
Type	Vertical, controlled leakage, centrifugal
Shaft seals, type, quantity	Mechanical, 4
Byron Jackson SU seal:	
Materials, stationary face	Carbon A GR CCP-72 or US Graphite Graphitar G114
Rotating face body	ASTM-A-351 Gr CF8
Rotating face ring	Titanium carbide, Kennametal K-162 B
Byron Jackson N-9000 seal	
Materials, stationary face	Morganite CNFJ or US Graphite Graphitar G114
Rotating Face	Tungsten Carbide, Kennametal KZ-801
Atomic Energy of Canada Limited (AECL) CAN4 seal:	
Materials, stationary face	Zr, -2.5 Wt. % Nb Carbon Graphite
Rotating face body	XM-19, annealed (Nitronic 50)
Rotating face ring	Tungsten Carbide
Design pressure, psig	2485
Design temperature, °F	650
Normal operating pressure, psig	2235
→(DRN 03-2059, R14)	
Normal operating temperature, °F	543
Rated flow, gpm (@ 553°F)	99,000
←(DRN 03-2059, R14)	

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TABLE 5.4-1 (Sheet 2 of 2)

REACTOR COOLANT PUMP PARAMETERS

<u>Parameter</u>	<u>Value</u>
Motor	
Voltage, volts	6600
Frequency, Hz/φ	60/3
Horsepower/speed, hot hp/rpm	7800/1183
Horsepower/speed, cold, hp/rpm	9700/1183
Service factor	1.0

TABLE 5.4-2 (Sheet 1 of 2)

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→(EC-8458, R307)

STEAM GENERATOR PARAMETERS<sup>(a)(b)</sup>

←(EC-8458, R307)

<u>Parameter</u>	<u>Value</u>
Number of units	2
→(DRN 03-2059, R14; EC-8458, R307)	
Heat transfer rate, each, Btu/hr.	6.3793 x 10 <sup>9</sup>
←(EC-8458, R307)	
<b>Primary side</b>	
Design pressure/temperature (psig/°F)	2485/650
→(DRN 06-1060, R15; EC-8458, R307)	
Coolant inlet temperature, °F	602
←(DRN 06-1060, R15; EC-8458, R307)	
Coolant outlet temperature, °F	543
→(DRN 06-1060, R15; EC-8458, R307)	
Coolant flow rate, each, lb/hr	81.8 x 10 <sup>6</sup>
←(DRN 03-2059, R14; 06-1060, R15)	
Coolant volume at 68 °F each, ft <sup>3</sup>	2051
←(EC-8458, R307)	
Tube size, OD, in.	3/4
→(EC-8458, R307)	
Tube thickness, nominal, in.	0.044 for Rows 1 and 2 0.043 for rows 3 to 138
←(EC-8458, R307)	
<b>Secondary side</b>	
Design pressure/temperature (psia/°F)	1100/560
→(DRN 03-2059, R14; 06-1060, R15; EC-8458, R307)	
Steam pressure, psia	837.2
←(DRN 06-1060, R15)	
Steam flowrate (at 0.10% moisture) each, lb/hr	8.3 x 10 <sup>6</sup>
←(EC-8458, R307)	
→(DRN 06-1060, R15; EC-8458, R307)	
Feedwater temperature at full power, °F	449.8
←(DRN 06-1060, R15)	
Moisture carryover, weight maximum, %	0.10
←(DRN 03-2059, R14; EC-8458, R307)	
Primary inlet nozzle, No./ID, in.	1/42
→(DRN 03-2059, R14; EC-8458, R307)	

a. Based on full power post-EPU (Extended Power Uprate) conditions with Replacement Steam Generators.

b. The parameters presented in this table are based on a feedwater temperature of 449.8°F. The best estimate feedwater temperature is 447.7°F. An evaluation was performed to assess the impact of the difference in feedwater temperatures which concluded the difference between the two values is insignificant.

←(DRN 03-2059, R14)

→(DRN 06-1060, R15)

←(DRN 06-1060, R15; EC-8458, R307)



→(EC-8458, R307)

STEAM GENERATOR PARAMETERS<sup>(a)(b)</sup>

←(EC-8458, R307)

<u>Parameter</u>	<u>Value</u>
Primary outlet nozzle, No./ID, in.	2/30
→(EC-8458, R307)	
Steam nozzle, No./ID, in.	1/34
←(EC-8458, R307)	
Feedwater nozzles, No./size/schedule	1/18/80
→(DRN 03-2059, R14; EC-8458, R307)	
Overall heat transfer coefficient (nominal)	1436
←(EC-8458, R307)	
Btu/hr-ft <sup>2</sup> -°F	
Blowdown flow (gpm nominal, per SG)	165
←(DRN 03-2059, R14)	

STEAM GENERATOR FABRICATION TESTING

<u>Component</u>	<u>Test (a)</u>
Tube sheet	
Forging	UT, MT
Cladding	UT, PT
Primary head	
→(EC-8458, R307) Forging	UT, MT
←(EC-8458, R307) Cladding	UT, PT
Secondary shell and head	
→(EC-8458, R307) Forging	UT, MT
←(EC-8458, R307) Tubes	UT, ET
Nozzles (forging)	UT, MT
Studs	UT, MT
Welds	
→(EC-8458, R307) Shell, circumferential	RT, MT, UT ISI welds only
←(EC-8458, R307) Cladding	UT, PT
←(EC-8458, R307) Feedwater Nozzles to shell	RT, MT, UT
←(EC-8458, R307) Tube-to-tube sheet	PT
Instrument connections	MT
→(EC-8458, R307) All accessible welds - after hydrostatic test	MT
Primary side pressure nozzles	PT
Pedestal to Head	RT, MT
←(EC-8458, R307)	
Support Lugs	MT, PT
a. UT = Ultrasonic testing MT = Magnetic-particle testing RT = Radiographic testing	
→(EC-8458, R307) ET = Eddy-current testing PT = Dye-penetrant testing	
←(EC-8458, R307)	

REACTOR COOLANT PIPING PARAMETERS

Parameter	Value
Number of loops (steam generators)	2
<small>→(DRN 03-2059, R14; EC-8458, R307)</small> Design Flow per loop, lb/hr <small>←(DRN 03-2059, R14; EC-8458, R307)</small>	81.8 x 10 <sup>6</sup>
Pipe size	
Reactor outlet ID/wall, in.	
Pipe	42/3-3/4 w/o clad
Elbow	42/4-1/8 w/o clad
Reactor inlet, ID/wall, in.	
Pipe	30/3 w/o clad
Elbow	30/3 w/o clad
Pump suction	
Elbow	30/3 w/o clad
Pipe	30/2-1/2 w/o clad
Surge line, (nominal pipe size in./ schedule)	12/160
Sprayline (nominal pipe size in./ schedule)	4/120 3/160
Design pressure, psia	2500
Design temperature, °F (hot & cold legs) (surge line)	650 700

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TABLE 5.4-5

REACTOR COOLANT PIPING TESTS

<u>Component</u>	<u>Test (a)</u>
Fittings (castings)	RT, PT
Piping (castings)	RT, PT, or MT
Pipe and elbows	UT, MT
Carbon steel plate	UT, PT
Welds	
Circumferential	RT, PT, or MT
Nozzles to pipe run	RT, MT, UT
Instrument connections to pipe	PT or MT
Cladding	UT, PT
Safe ends to nozzles	RT, PT

a. Key

UT = Ultrasonic testing

MT = Magnetic-particle testing

PT = Dye-penetrant testing

RT = Radiographic testing

WSES-FSAR-UNIT-3

TABLE 5.4-6 (Sheet 1 of 2)

Revision 309 (06/16)

PRESSURIZER PARAMETERS

<u>Property</u>	<u>Parameter</u>
Design pressure, psia	2500
Design temperature, °F	700
Normal operating pressure, psia	2250
Normal operating temperature, °F	653
→(DRN 06-904, R15)	
Internal free volume, ft <sup>3</sup>	1519
←(DRN 06-904, R15)	
Normal (full power) operating water volume, ft <sup>3</sup>	800
Normal steam volume full power, ft <sup>3</sup>	700
→(DRN 00-1673, R10; 01-1361, R12; 05-892, R14, LBDCR 16-017, R309)	
Installed heater capacity, kW (Nominal)	1350
←(DRN 00-1673, R10; 01-1361, R12; 05-892, R14, LBDCR 16-017, R309)	
→(DRN 03-1268, R13)	
Spray flow, maximum, gpm	490
←(DRN 03-1268, R13)	
Spray flow, continuous, gpm	1.5
<b>Nozzles</b>	
Surge line (1 ea) nominal, in.	12, schedule 160
Safety valves (3), ID, in.	6, schedule 160
Spray (1) nominal, in.	4, schedule 120
→(DRN 00-1631, R10; 05-892, R14, LBDCR 15-004, R309, LBDCR 16-017, R 309)	
Heaters (30), OD, in.	1.25
(F4 Plugged / F3 & D4 CAPPED)	
←(DRN 00-1631, R10; 05-892, R14, LBDCR 15-004, R309, LBDCR 16-017, R309)	
<b>Instruments,</b>	
Level (4) nominal, in.	3/4, schedule 160
Temperature (1) nominal, in.	1, schedule 160
Pressure (2) nominal, in.	3/4, schedule 160
<b>Dimensions</b>	
Overall length, including skirt and spray nozzle, in.	441
Outside diameter, in.	106-1/2
Inside diameter, in.	96
Cladding thickness, in. (minimum)	1/8

WSES-FSAR-UNIT-3

TABLE 5.4-6 (Sheet 2 of 2)

Revision 15 (03/07)

PRESSURIZER PARAMETERS

<u>Property</u>	<u>Parameter</u>
→(DRN 06-904, R15) Dry weight, including heaters, lb	203,288
←(DRN 06-904, R15) Flooded weight, including heaters, lb	297,712

WSES-FSAR-UNIT-3

TABLE 5.4-7

PRESSURIZER TESTS

Component	Test(a)
Heads	
Plates	UT, MT
Cladding	UT, PT
Shell	
Plates	UT, MT
Cladding	UT, PT
Heaters	
Tubing	UT, PT
Centering of elements	RT
End plug	UT, PT
Nozzle (Forgings)	UT, MT
Studs	UT, MT
Welds	
Shell longitudinal	RT, MT, UT
Shell circumferential	RT, MT, UT
Cladding	UT, PT
Nozzles	RT, MT
Nozzle safe ends	RT, PT
Instrument connections	PT
Support skirt	MT
Temporary attachment after removal	MT
All welds after hydrostatic test	MT or PT
Heater assembly, end plug weld	RT, PT

a. Key:

UT = ultrasonic testing, MT = magnetic particle testing,  
 PT = dye-penetrant testing, RT = radiographic testing

WSES-FSAR-UNIT-3

TABLE 5.4-8

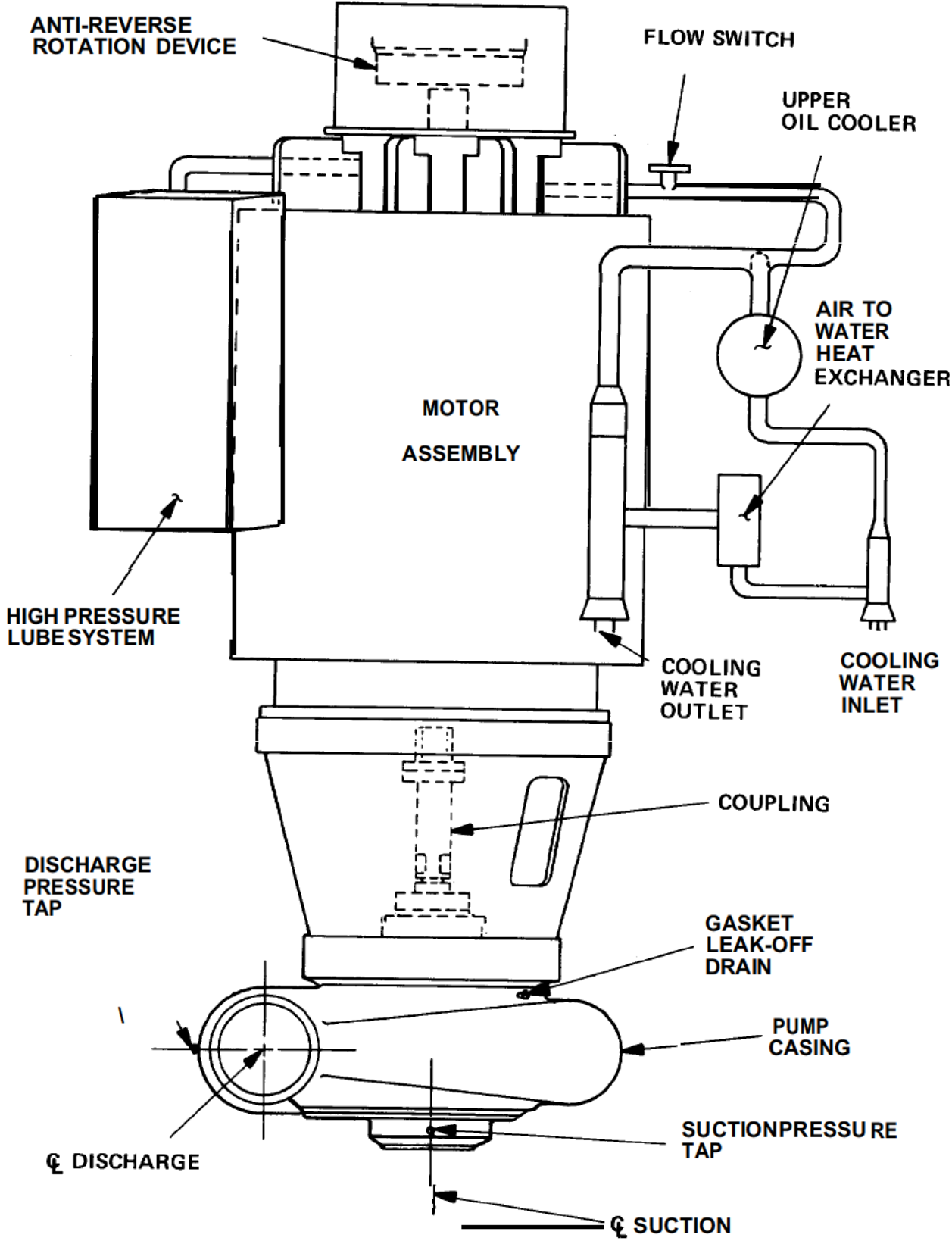
QUENCH TANK PARAMETERS

<u>Property</u>	<u>Parameter</u>
Design pressure, psig (internal/external)	130/15
Design temperature, °F ,	350
Normal operating pressure, psig	3
Normal operating temperature, °F	120
Minimum internal volume, gal	2400
Blanket gas	Nitrogen
Nozzles	
Pressurizer discharge (1) nominal, in.	12 Sch. 40 S
Demineralized water (1), in./rating	2/3000 lb SW Coupling
Rupture disc (1) in.	20 flanged
Drain (1), in./rating	2/3000 lb SW Coupling
Temperature instrument (1), in./rating	1/3000 lb SW Coupling
Level instrument (2), in./rating	1/3000 lb SW Coupling
Vent (1), in./rating	2/3000 lb SW Coupling
Vessel material	ASTM-SA-240 Type 304
Dimensions	
Overall length, in.	151.75
Outside diameter, in.	72
Dry weight, lb	6700
Flooded weight, lb	26,850
Code and date	ASME Section VIII Div. 1 through Summer 1970 Addenda



PRIMARY SAFETY VALVE PARAMETERS

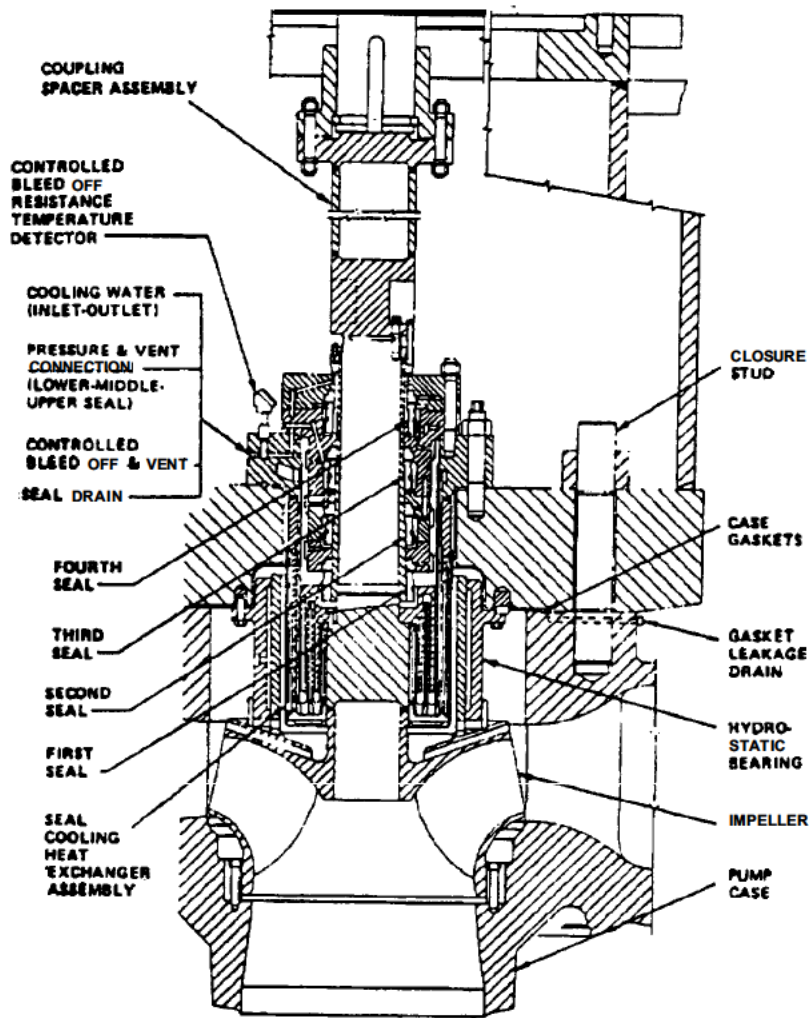
Property	Parameter
Design pressure, psia	2500
→(DRN 06-872, R15)	
Design temperature, °F	700
←(DRN 06-872, R15)	
Fluid	Saturated steam, 2000 ppm H <sub>3</sub> BO <sub>3</sub> , pH=5.0
Set pressure, psia	2500 +/-3
Capacity, lb/hr. at set pressure, each	460,000
Type	Spring loaded safety- balanced bellows. Enclosed bonnet.
Accumulation, percent	3
Backpressure Max buildup/max. superimposed, psig	500/130
Blowdown, percent	5
Materials	
Body	ASME SA 182, GR. F316
Disc	ASTM A637, GR. 688
Nozzle	ASME SA 182, GR347



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REACTOR COOLANT PUMP OUTLINE

Figure  
5.4-1



NOTE :

MARGANITTE CNFJ OR US GRAPHITE  
 GRAPHITAR GR114 OR ZR.2.5 WT %  
 Nb CARBON GRAPHITE SEALS MAY BE USED

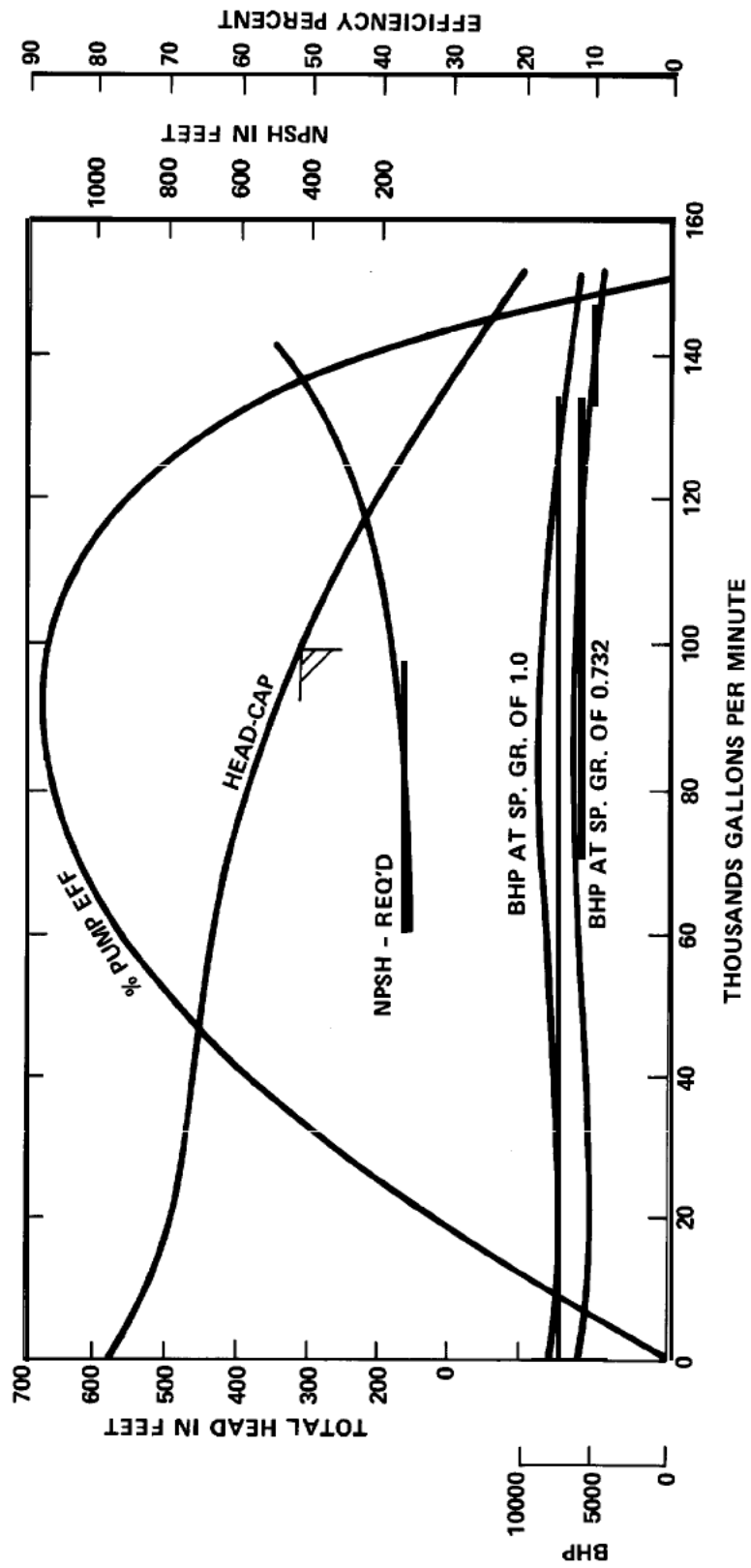
THIS DRAWING IS A TYPICAL REACTOR COOLANT PUMP ASSEMBLY AND IS NOT AN EXACT REPRESENTATION OF THE EQUIPMENT IN USE. SEE DRAWING 1564-1550 FOR THE IN-PLANT REACTOR COOLANT PUMP ASSEMBLY.

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Waterford Steam  
 Electric Station #3

REACTOR COOLANT PUMP & SEAL

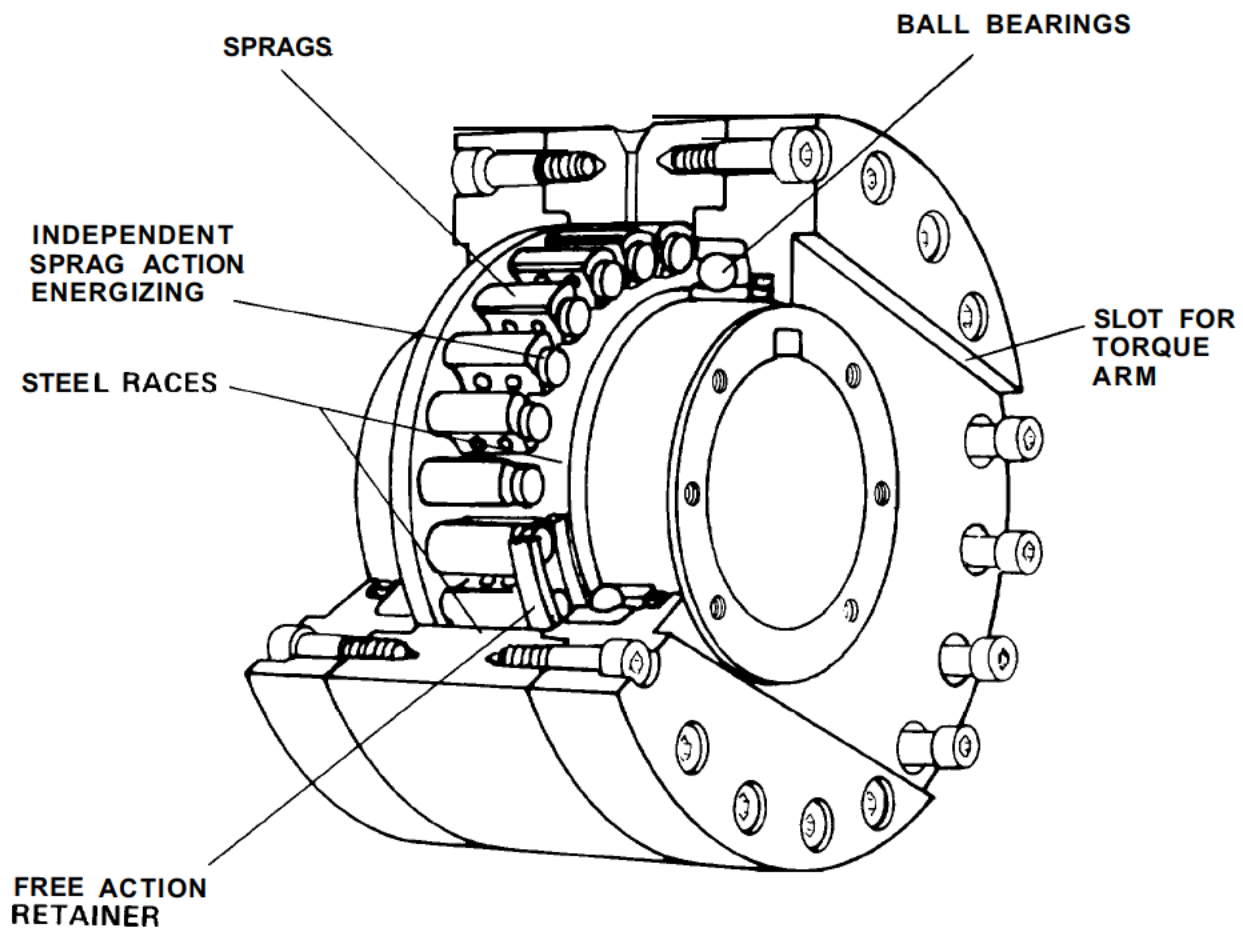
Figure  
 5.4-2



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REACTOR COOLANT PUMP PERFORMANCE

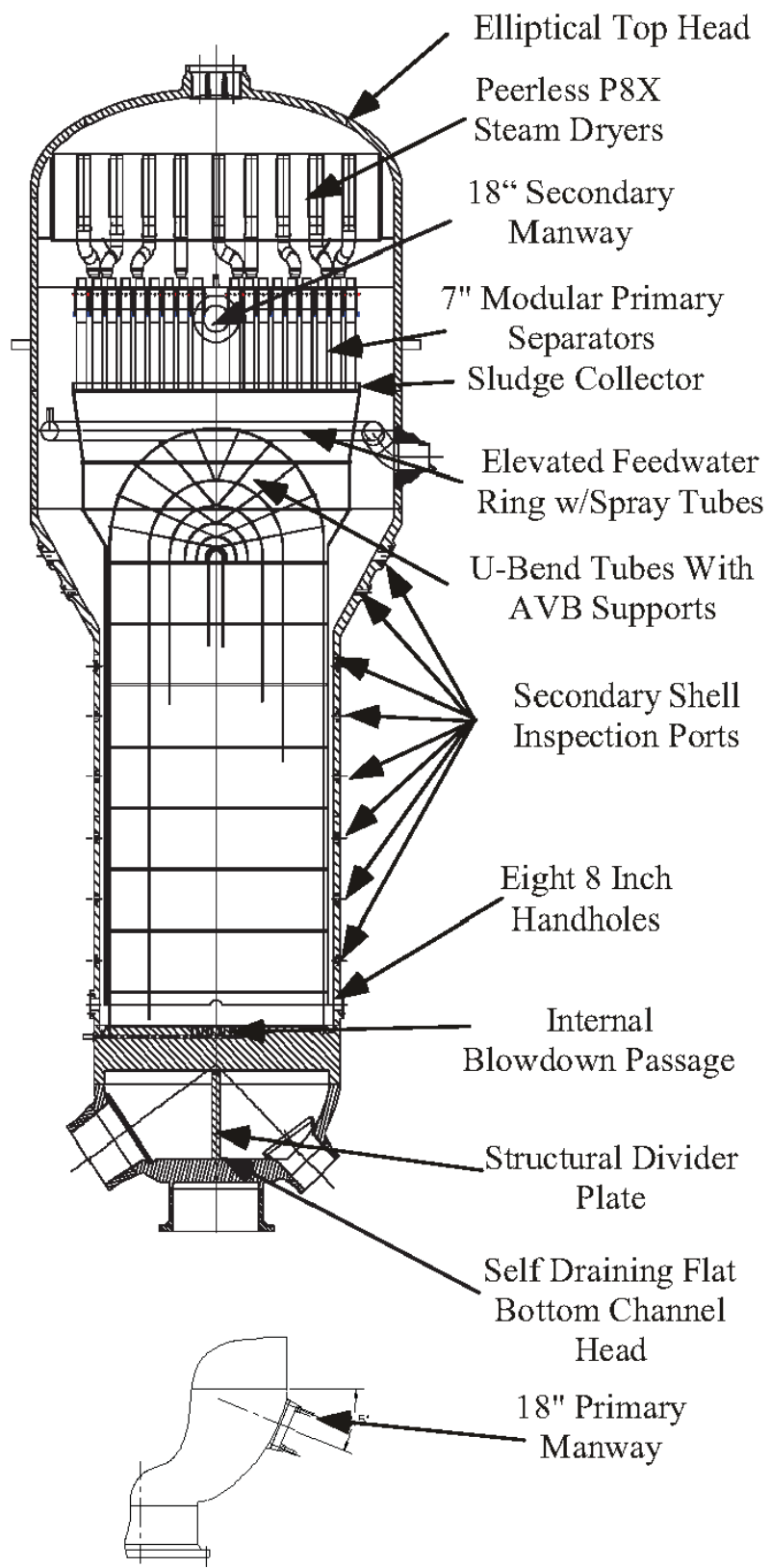
Figure  
5.4-3



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ANTI-REVERSE ROTATION DEVICE

Figure  
5.4-4



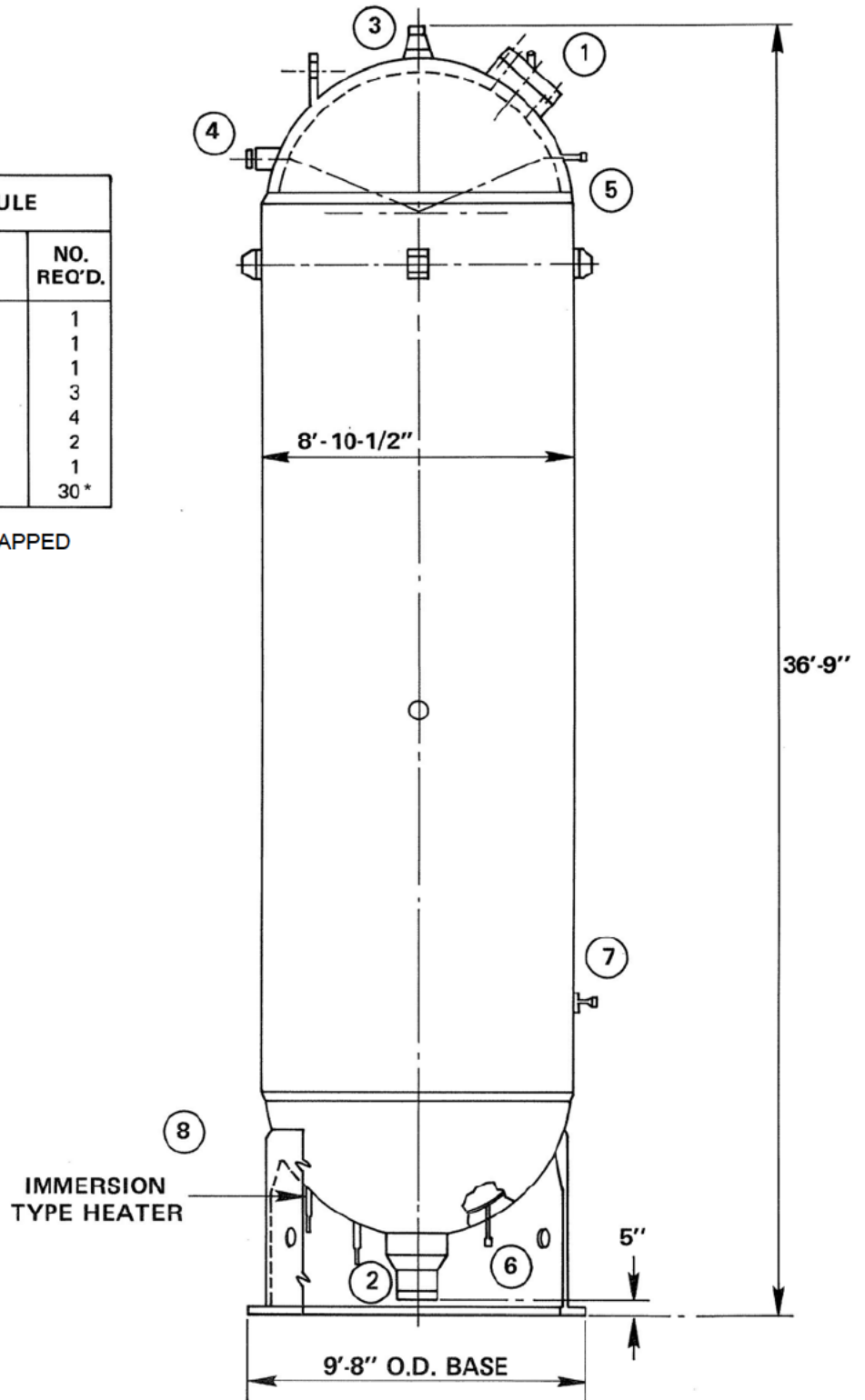
Elliptical Top Head  
 Peerless P8X  
 Steam Dryers  
 18" Secondary  
 Manway  
 7" Modular Primary  
 Separators  
 Sludge Collector  
 Elevated Feedwater  
 Ring w/Spray Tubes  
 U-Bend Tubes With  
 AVB Supports  
 Secondary Shell  
 Inspection Ports  
 Eight 8 Inch  
 Handholes  
 Internal  
 Blowdown Passage  
 Structural Divider  
 Plate  
 Self Draining Flat  
 Bottom Channel  
 Head  
 18" Primary  
 Manway

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Waterford Steam Electric Station #3	STEAM GENERATOR	Figure 5.4-5
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NOZZLE SCHEDULE		
NO.	SERVICE	NO. REQ'D.
1	Manway	1
2	Surge	1
3	Spray	1
4	Safety Valve	3
5	Instrument	4
6	Instrument	2
7	Temperature	1
8	Heater	30*

\*F4 PLUGGED / F3 CAPPED

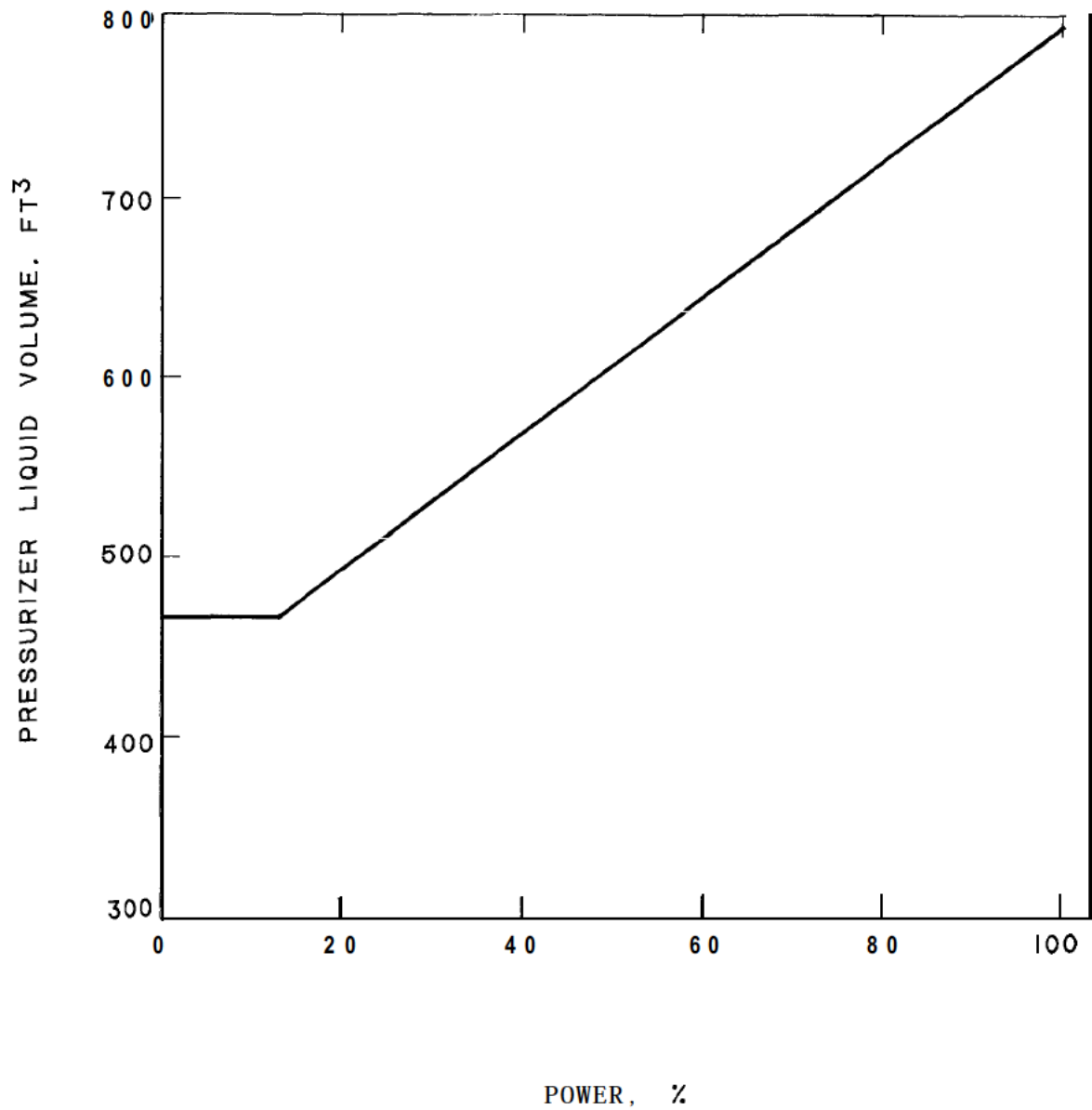


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Waterford Steam  
Electric Station #3

PRESSURIZER

Figure  
5.4-6

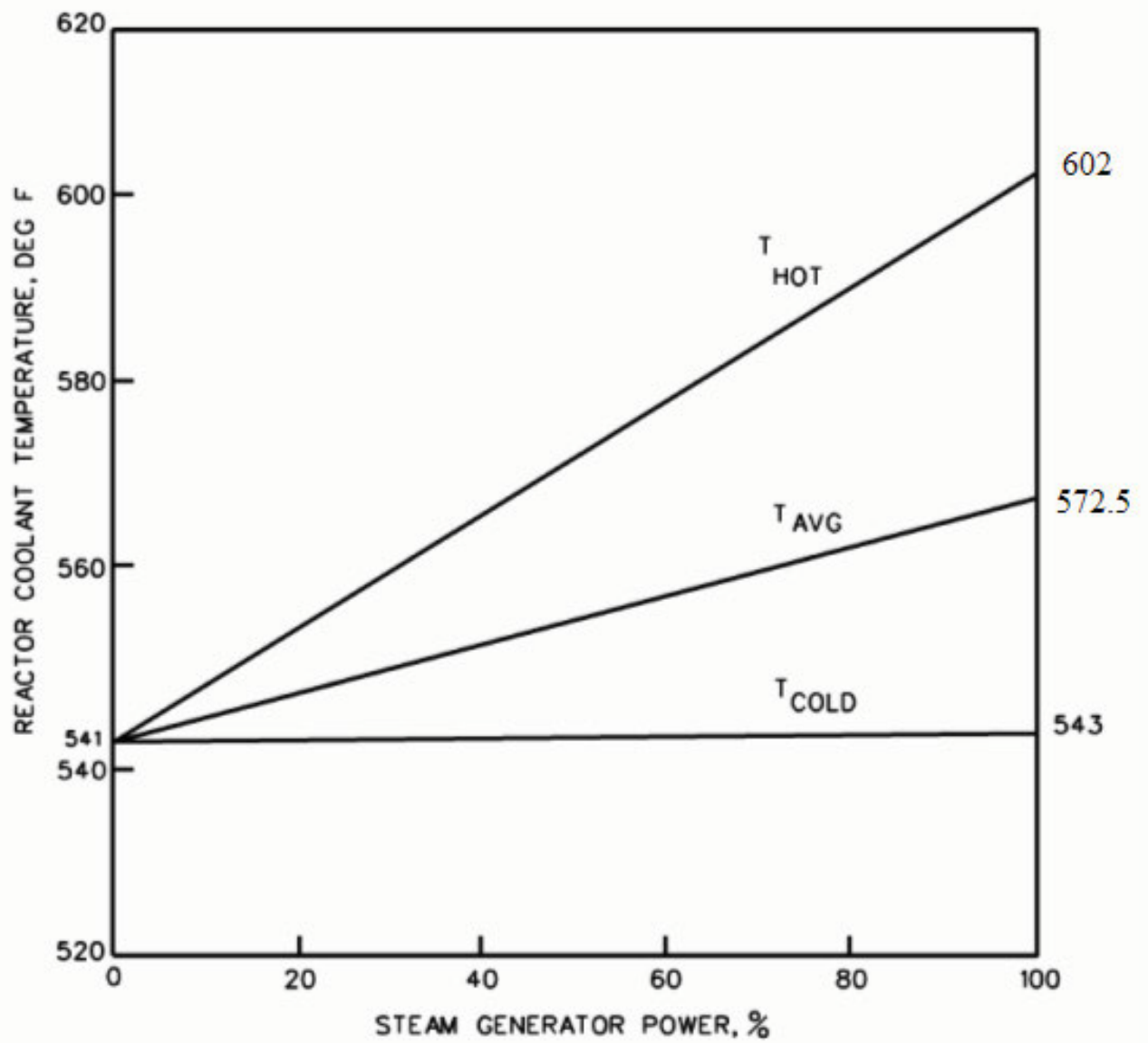


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Electric Station #3

PRESSURIZER LIQUID VOLUME PROGRAM

Figure  
5.4-7





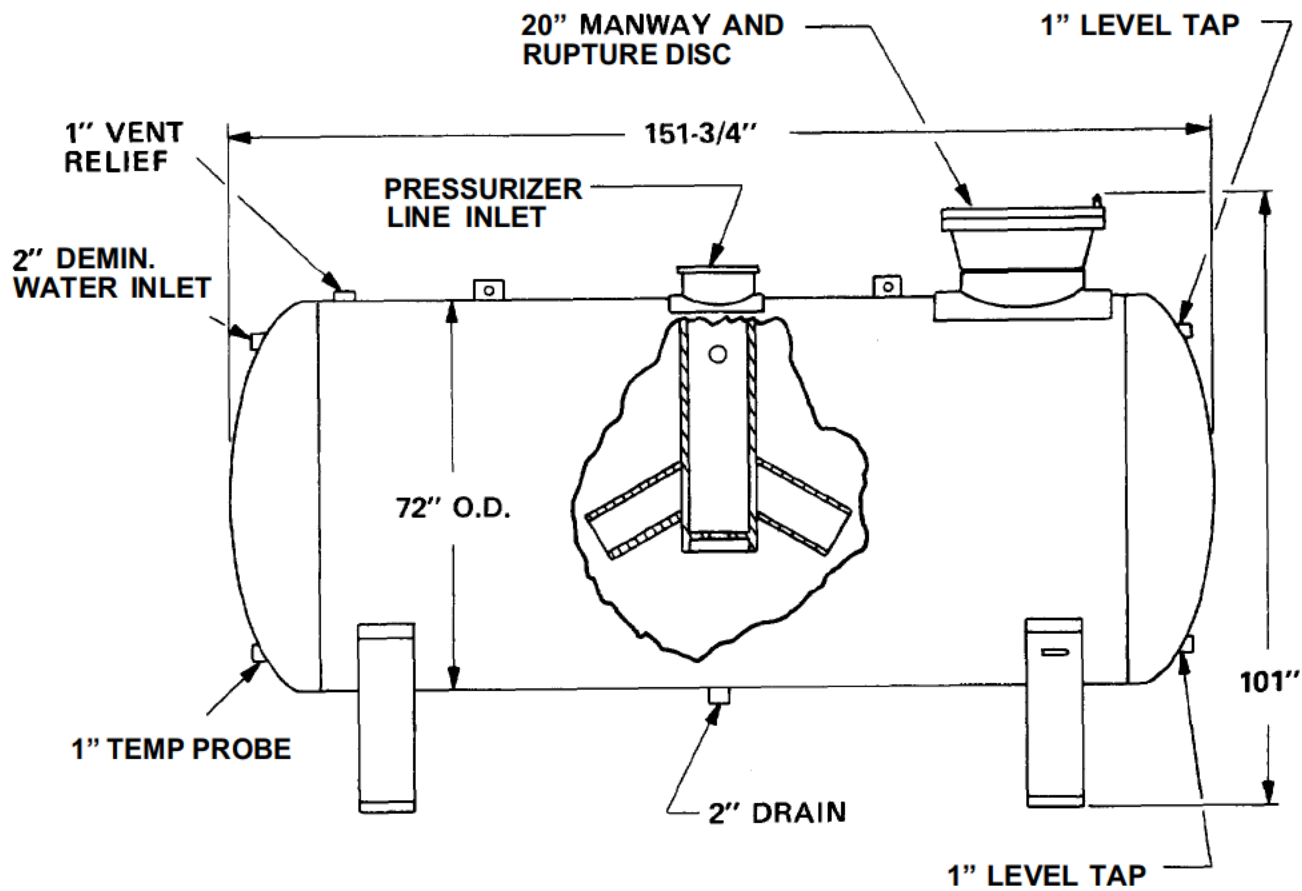
Revision 307 (07/13)

PRESSURIZER LEVEL ERROR (INCHES) <sup>(b)</sup>	ACTION <sup>(a)</sup>
+36	HIGH LEVEL ERROR ALARM
+14.4	ENERGIZE ALL PRESSURIZER HEATERS & BACKUP SIGNAL TO STOP 2 BACKUP CP'S
+11.4	ALL PRESSURIZER HEATERS OFF
-4	STOP BACKUP CP 2
-6	STOP BACKUP CP3
-9	START BACKUP CP2
-14	START BACKUP CP3
-21.6	LOW LEVEL ERROR ALARM AND BACKUP SIGNAL TO START ALL CP'S

(a) CP = CHARGING PUMP

(b) INSTRUMENT SETPOINT TOLERANCES ARE NOT  
INCLUDED IN THE VALUES SHOWN ABOVE.

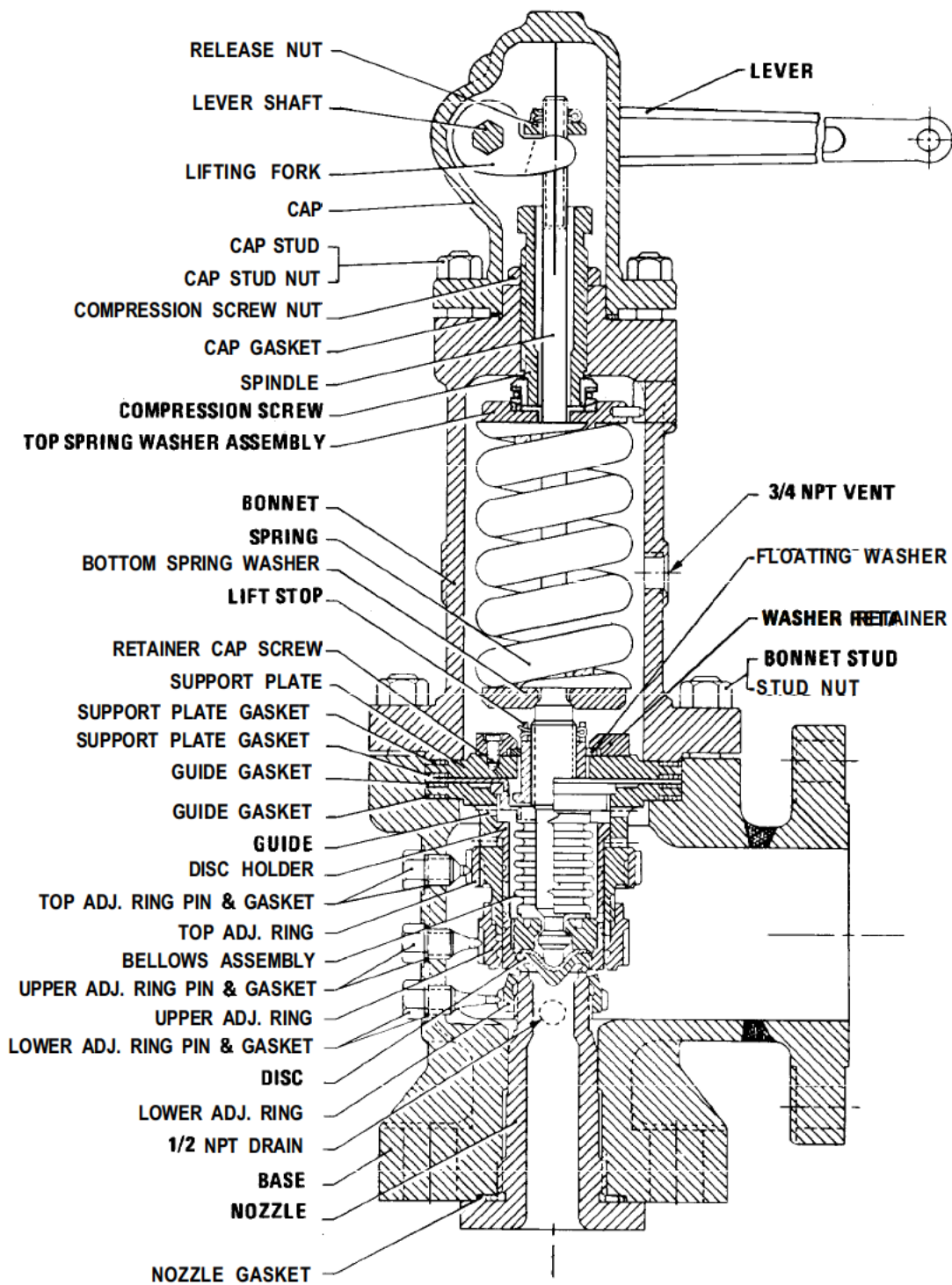
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QUENCH TANK

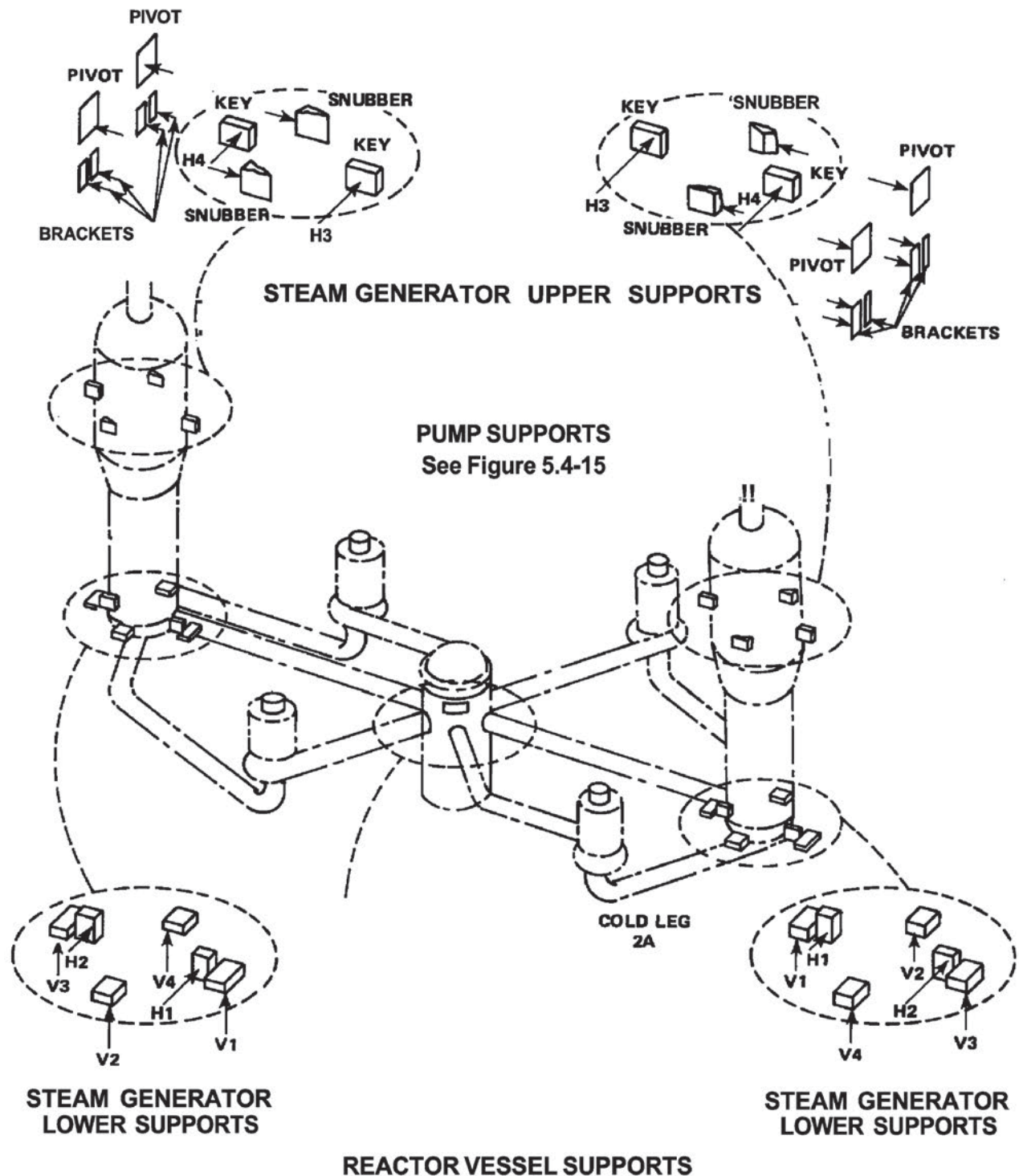
Figure  
 5.4-1 0



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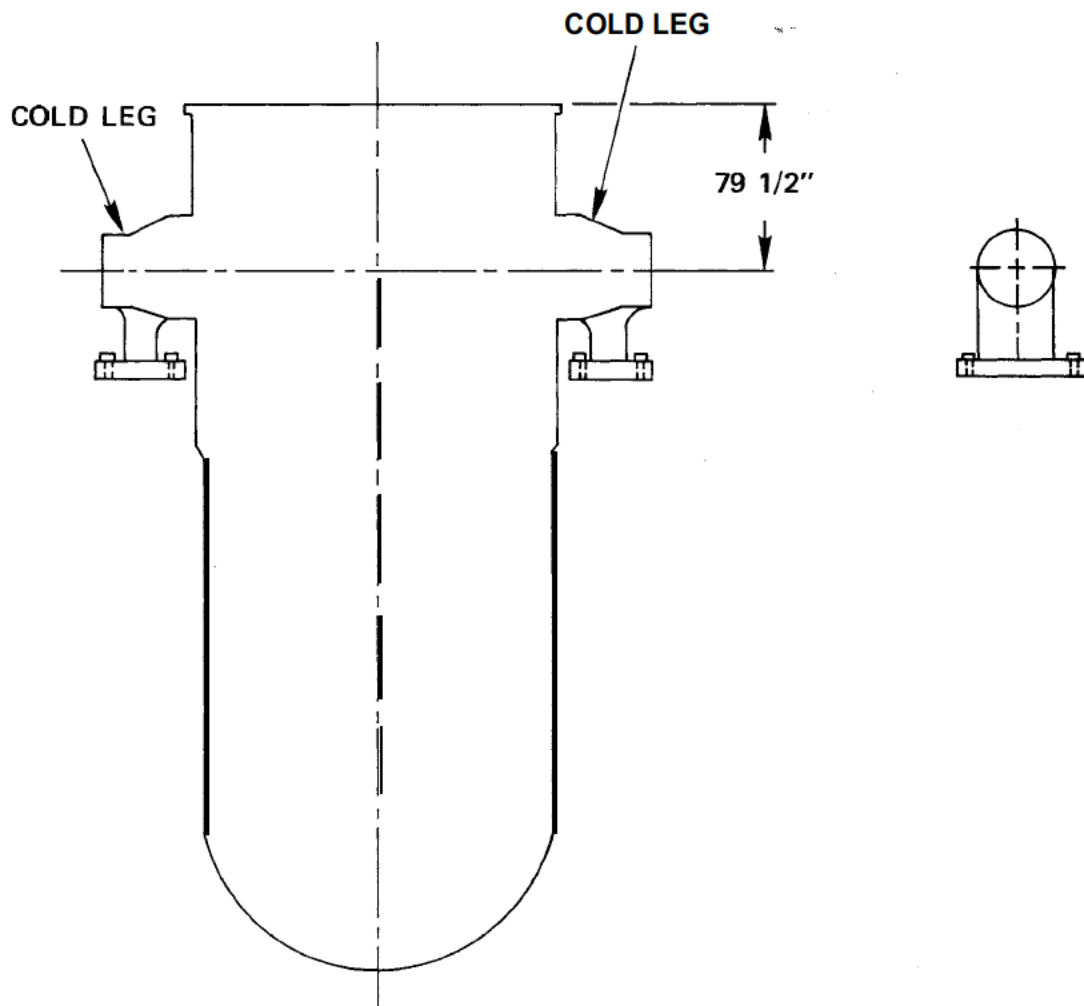
PRIMARY SAFETY VALVE

Figure  
 5.4-11



THE SHIM PLATE PACK (STOP) ON THE REACTOR SIDE OF THE KEYWAY WAS PERMANENTLY REMOVED FROM THE STEAM GENERATOR LOWER SUPPORTS SLIDING BASE DURING STEAM GENERATOR REPLACEMENT.

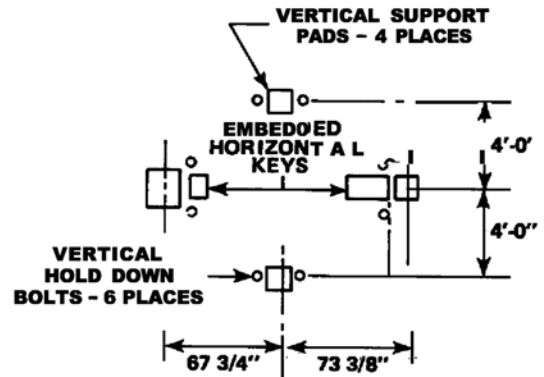
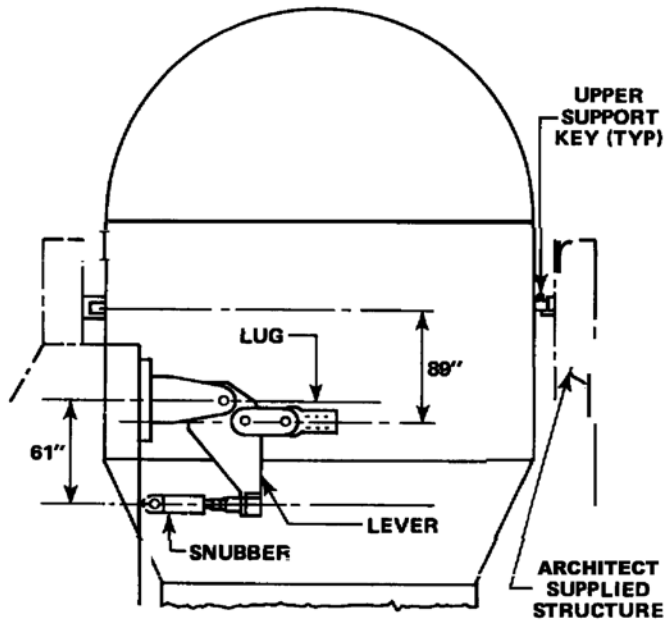
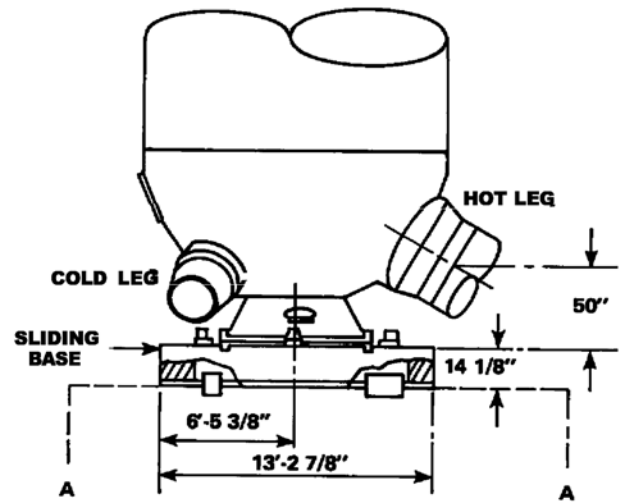
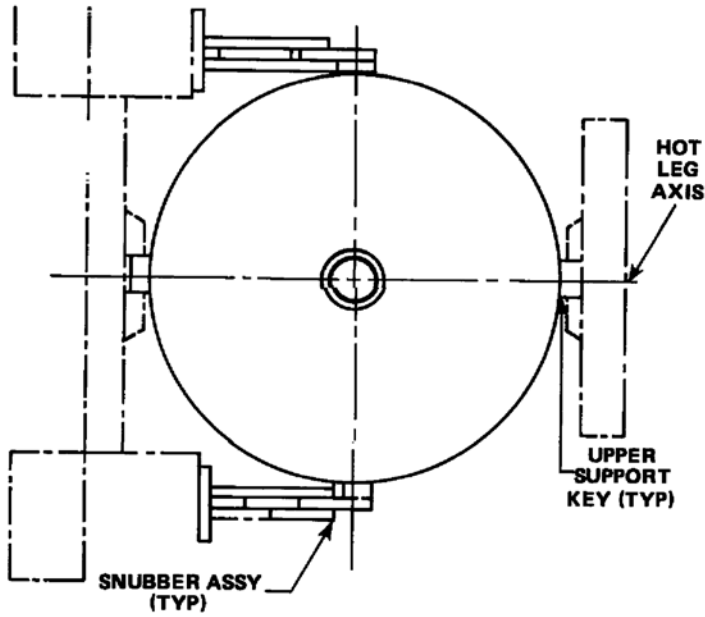
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REACTOR VESSEL SUPPORTS

Figure  
5.4-13



**UPPER SUPPORTS**

**LOWER SUPPORTS**

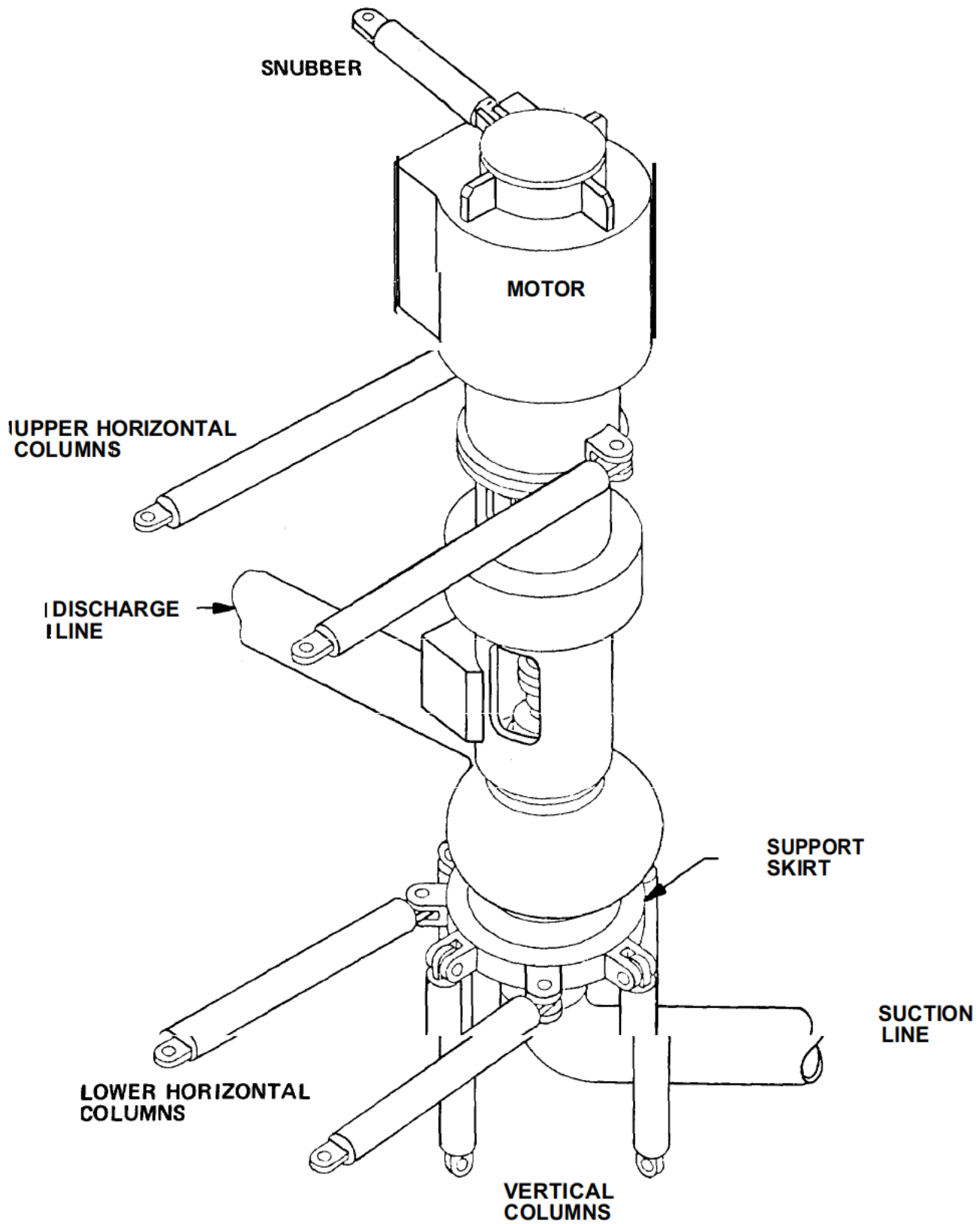
THE SHIM PLATE PACK (STOP) ON THE REACTOR SIDE OF THE KEYWAY WAS PERMANENTLY REMOVED FROM THE STEAM GENERATOR LOWER SUPPORTS SLIDING BASE DURING STEAM GENERATOR REPLACEMENT.

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STEAM GENERATOR SUPPORTS

Figure  
5.4-14

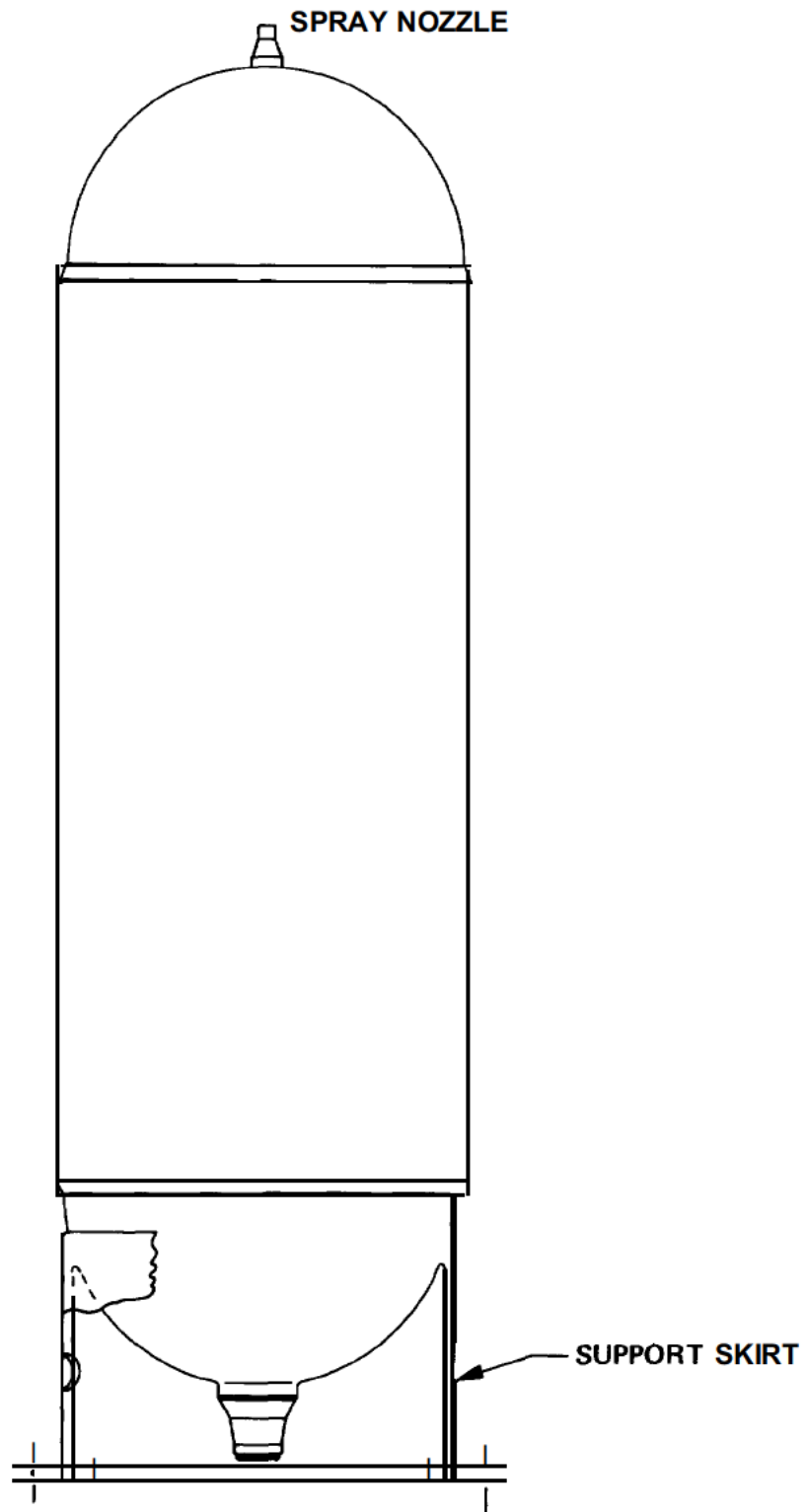


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REACTOR COOLANT PUMP SUPPORTS

Figure  
5.4-15

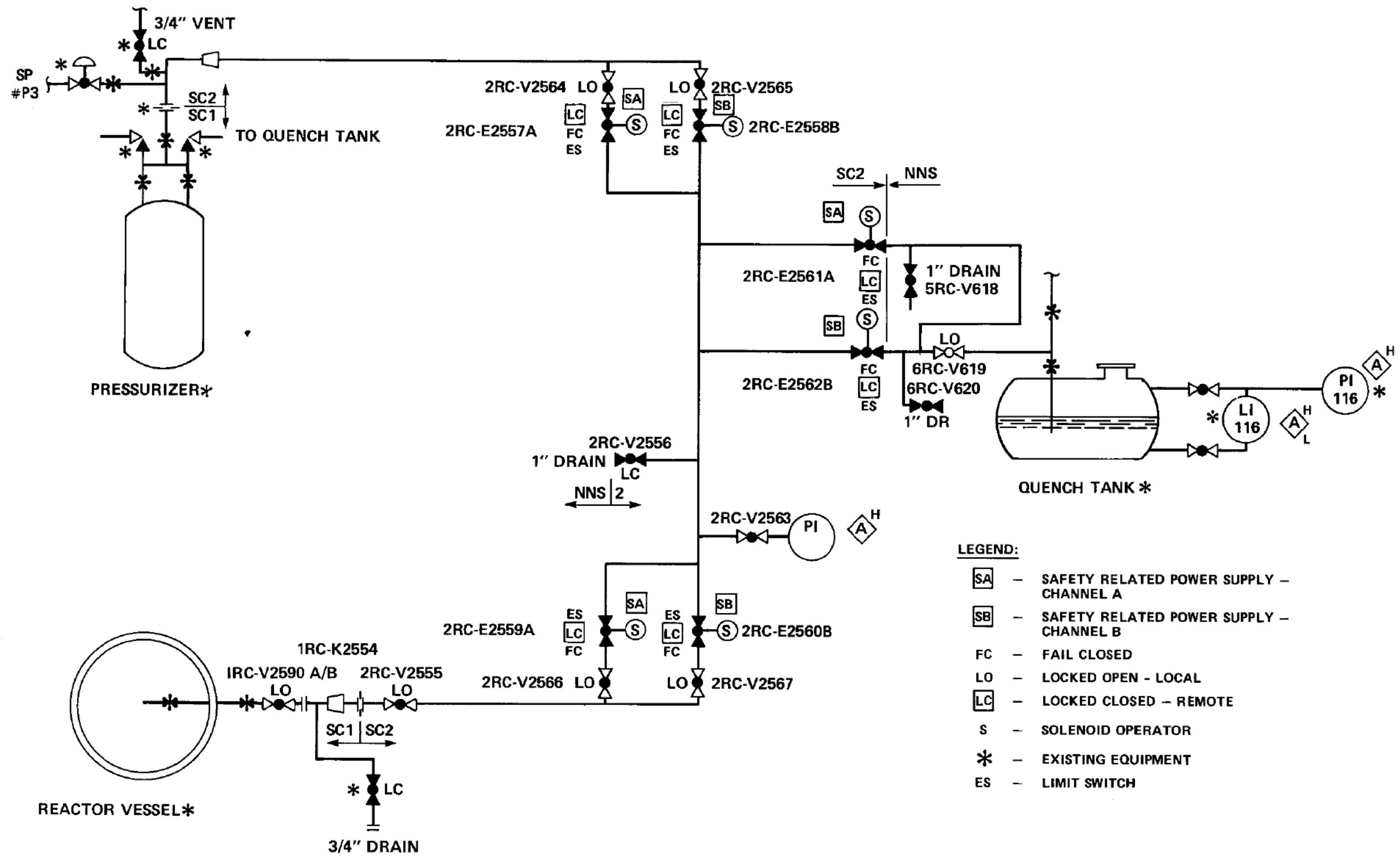




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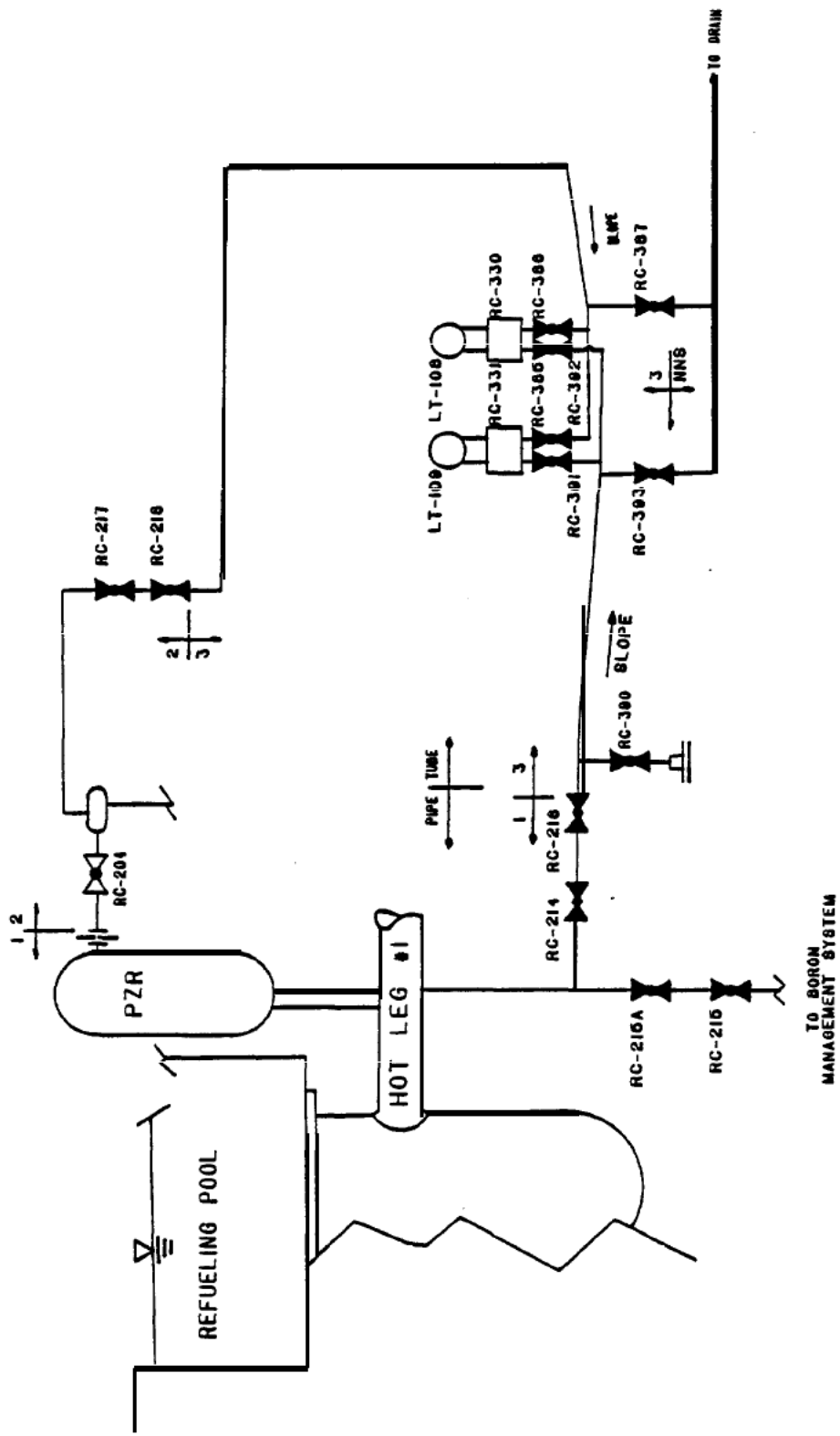
PRESSURIZER SUPPORTS

Figure  
5.4-16



- LEGEND:**
- [SA] - SAFETY RELATED POWER SUPPLY - CHANNEL A
  - [SB] - SAFETY RELATED POWER SUPPLY - CHANNEL B
  - FC - FAIL CLOSED
  - LO - LOCKED OPEN - LOCAL
  - [LC] - LOCKED CLOSED - REMOTE
  - S - SOLENOID OPERATOR
  - \*
  - ES - LIMIT SWITCH

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 REACTOR COOLANT SYSTEM  
 VENTING SYSTEM  
 FIGURE 5.4-17



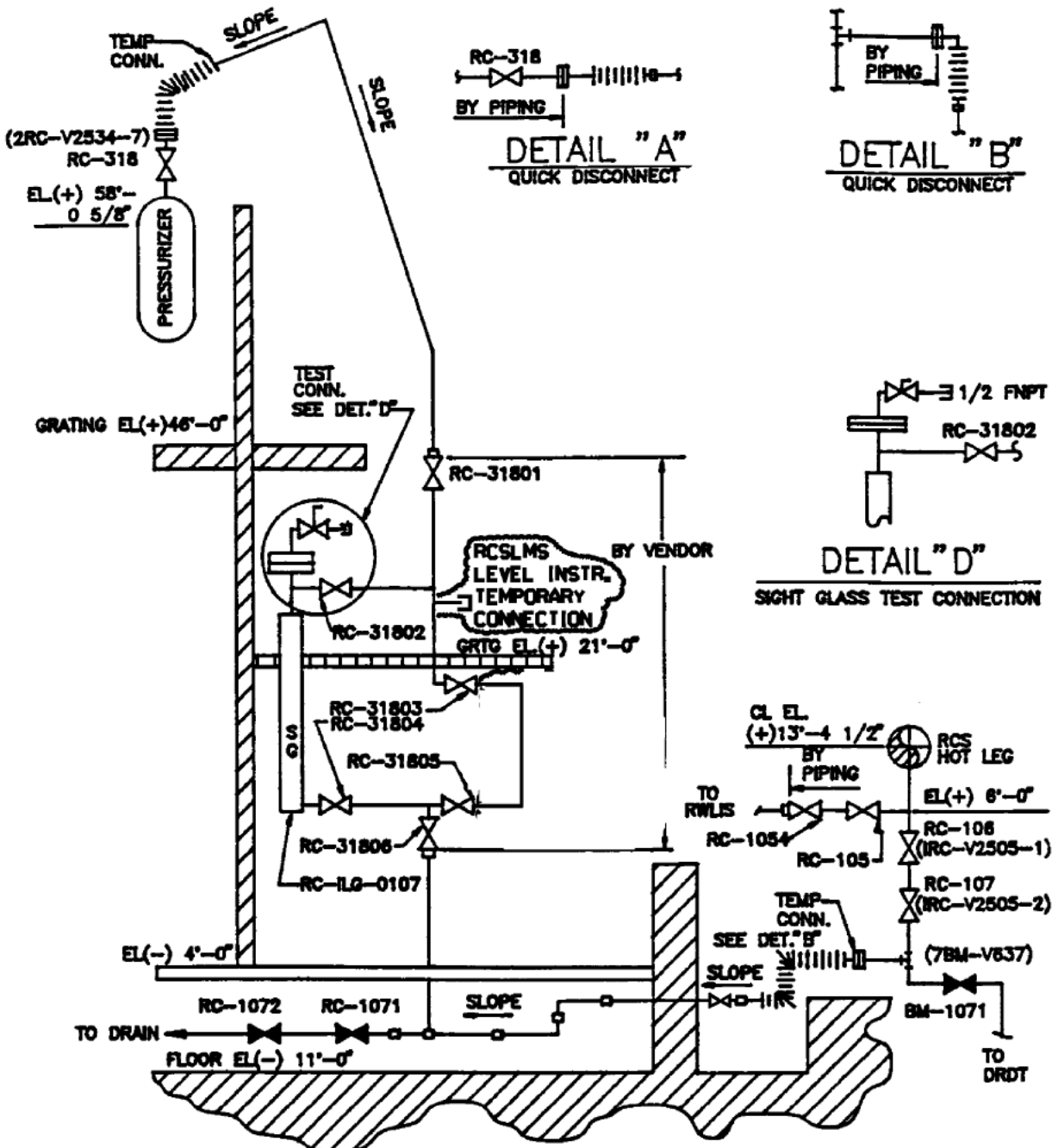
Note: Combustion Engineering valve nomenclature utilized.

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REFUELING WATER LEVEL  
INDICATING SYSTEM

Figure  
5.4-18



**REACTOR COOLANT SHUTDOWN  
LEVEL MEASUREMENT SYSTEM  
LEVEL INSTR.**

REVISION 12 (10/02)

<p>Waterford Steam Electric Station #3</p>	<p>REACTOR COOLANT SHUTDOWN LEVEL MEASUREMENT SYSTEM</p>	<p>Figure 5.4-19</p>
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APPENDIX 5.4A

DYNAMIC ANALYSIS FOR THE WATERFORD 3 REACTOR VESSEL

SUPPORT LOADS UNDER LOCA CONDITIONS

→(DRN 03-2059, R14)

This appendix was written with respect to main coolant loop breaks (MCLBs), which were subsequently eliminated from consideration of mechanical (dynamic) effects via leak-before-break (LBB) arguments. Elimination of MCLBs by LBB and replacement with branch line pipe breaks (BLPBs) is discussed in Section 3.6.2. LBB methodology is discussed in Section 3.6.3.

This appendix has been retained for the historical record because the Reactor Vessel (RV) support loads, as well as RV shell response motions used in RV internals evaluations, that were generated from this analysis, bound the respective RV responses generated by the branch line pipe break (BLPB) analysis for power uprate to 3716 MWt.

←(DRN 03-2059, R14)

#### 5.4A.1 PURPOSE

This appendix presents the results of dynamic structural analyses of the WATERFORD 3 reactor vessel and supports subjected to loads induced by pipe breaks at the reactor vessel inlet and outlet nozzles and at the steam generator inlet nozzle. Two generations of dynamic analysis are presented; the original analysis using the original model and results of Reference 1 thermohydraulic methodology, the second analysis using an expanded model and results of Reference 2 methodology. Simultaneous effects of pipe break thrust and external and internal horizontal and vertical asymmetric pressure loads were applied to the reactor vessel and internals as a consequence of each postulated pipe rupture. A non-linear time history dynamic analysis of a three dimensional mathematical model of the Reactor Coolant System including details of the reactor internals, pressure vessel, supports and piping was performed for each postulated pipe break to demonstrate the adequacy of the reactor vessel supports.

#### 5.4A.2 CONCLUSION

The calculated loads on reactor vessel supports do not exceed specified loads except for one load in the original analysis. For all supports, all calculated loads have been evaluated and found to be acceptable.

#### 5.4A.3 BACKGROUND

The reactor internals, including the fuel and supporting structures, are suspended from the closure flange region of the reactor vessel and are surrounded by the cylindrical "core support barrel" (CSB) as shown on Figure 5.4A-1. The CSB and reactor vessel are essentially concentric cylinders throughout the length of the CSB.

Upon postulation of a break in a primary coolant pipe, as on Figure 5.4A-2, several rapidly occurring events cause internal and external transient loads to act upon the reactor vessel. For a reactor vessel inlet break, asymmetric pressure changes take place in the annulus between the CSB and the vessel. Decompression occurs on the side of the vessel nearest the pipe break before pressure on the opposite side changes. The momentary difference in pressure across the CSB induces lateral loads in opposite directions on the CSB and the reactor vessel as shown on Figure 5.4A-3. Vertical loads are also applied to the internals and to the vessel due to the vertical flow resistance through the core and asymmetric axial decompression of the vessel. Simultaneously, as fluid escapes through the break, the annulus between the reactor vessel and biological shield wall becomes asymmetrically pressurized. The difference in pressure across the reactor vessel causes additional horizontal and vertical external loads on the vessel as shown on Figure 5.4A-4. In addition, the vessel is loaded by the effects of initial tension release and blowdown thrust on the broken pipe (Figure 5.4A-5).

The loads occur simultaneously as shown on Figure 5.4A-6. For a reactor vessel outlet break, the same type of loadings as shown on Figure 5.4A-6 occur, but the internal loads are more predominately vertical due to smaller break size and more rapid decompression of the upper plenum. For each postulated break the time history maximum reactor vessel support reactions due to the complete set of horizontal and vertical loads are calculated.

## WSES-FSAR-UNIT-3

For a steam generator (SG) inlet break, the same type of loadings on the reactor vessel (RV) as shown on Figure 5.4A-6 occur, but because of the position of the break, the initial tension release and blowdown thrust contain a vertical component of load that enhances rocking effects on the vessel. Because this break occurs in the SG subcompartment and not in the RV cavity, there are no differential pressure loads on the reactor vessel.

### 5.4A.4 MODELS



Different models were developed for the two generations of dynamic analysis. In the original analysis, condensed structural models of the RCS and reactor internals were created from highly detailed representations of each component by maintaining response characteristics and interface response compatibility. A detailed model of the reactor coolant system is shown on Figure 5.4A-7 and a detailed model of the reactor internals is shown on Figure 5.4A-8. The internals model and the RCS model were condensed and coupled to form the final model used for the analysis of the vessel supports. (See Figures 5.4A-9 and 5.4A-10.) The condensed model of the internals is intended to represent the effects of the reactor internals on the reactor vessel support reactions and is not intended to be used to analyze the internals themselves. The adequacy of the reduced model of the internals for this purpose was verified by analysis of planar models of a coupled system having a detailed internals model and a simplified representation of the vessel and supports. Figures 5.4A-11 and 5.4A-12 show excellent agreement between the reactor vessel support reactions using the detailed and the reduced internals models. This model was used for the analysis of the 350 in.<sup>2</sup> R.V. inlet nozzle and 100 in.<sup>2</sup> R.V. outlet nozzle circumferential guillotines. The criteria used to define break and crack locations and configuration for high energy piping systems is discussed in FSAR Section 3.6.2.1.



The details of the coupled model are summarized in Table 5.4A-2. The model is three dimensional and has 680 total static degrees of freedom and 61 mass degrees of freedom. The reactor vessel and all internal components are modeled as colinear elements and gaps are present at internal and support interfaces. Details of the reactor vessel support arrangements are shown on Figure 5.4A-13.

The models for the second set of dynamic asymmetric loads analyses contain a similarly detailed reactor vessel and condensed reactor internals model. However, they also contain detailed steam generator (SG), pump (RCP), RCS piping and RCS support modelling and are a combination of models shown by Figures 5.4A-7 and 5.4A-8. These expanded models were used for a comparative analysis of the 350 in.<sup>2</sup> break (see Figure 5.4A-5) and for the analysis of the 600 in.<sup>2</sup> S.G. inlet nozzle circumferential guillotine. For the 600 in.<sup>2</sup> SG inlet nozzle break, an elasto-plastic multi-mass model of the severed hot leg with a gapped pipe restraint was added to the model. The model used for the 350 in.<sup>2</sup> break has 1912 total static degrees of freedom and 108 mass degrees of freedom. The model used for the 600 in.<sup>2</sup> break has 1961 total static degrees of freedom and 131 mass degrees of freedom.

The physical definition of the structure was given as input to the STRUDL<sup>(3)</sup> Computer Code which generated the condensed stiffness matrix used in the dynamic analysis. Hydrodynamic effects, including both virtual mass and annular effects, were accounted for in the coupling between the RV and CSB and between the CSB and the core shroud. The resulting mass matrix is non-diagonal because of the inclusion of the annular hydrodynamic effects (Reference 4).

All RCS support stiffnesses and pipe restraint stiffnesses used were plant specific.

## 5.4A.5 FORCING FUNCTIONS

The original model shown on Figures 5.4A-9 and 5.4-10 and described in Table 5.4A-2 was subjected to the loads that resulted from a postulated rupture at the inlet and outlet RV nozzles (350 in.<sup>2</sup> and 100 in.<sup>2</sup> breaks respectively). The location and size of breaks have been determined using the methods of Reference 5 and are summarized in Table 5.4A-3. For each postulated break a thermohydraulic analysis was performed according to the procedures and models developed in Reference 1. The resulting pressure and flow parameters were used to calculate three dimensional time history forcing functions acting on the reactor vessel and internals at the locations shown on Figures 5.4A-9 and 5.4A-10. Analyses to calculate the asymmetric subcompartment pressures in the reactor vessel cavity were performed by Ebasco using mass energy release data provided by CE. The resulting pressures were used to calculate the three dimensional time history forces applied to the exterior of the vessel. The forces were applied to the model at the locations shown on Figure 5.4A-10. Internal and external forces were applied simultaneously in three directions to the Vessel-Internals System. The dynamic analysis to determine the response to the system is performed using the DAGS and FORCE (6,7) codes to determine the maximum reactions at the vessel support to locations shown on Figure 5.4A-14.

→(DRN 00-1059)

The expanded models, shown on Figures 5.4A-7 and 5.4A-10 and described in Tables 5.4A-4 and 5.4A-5, were subjected to the loads that resulted from a postulated rupture at each of the RV inlet and SG inlet nozzles (350 in.<sup>2</sup> and 600 in.<sup>2</sup> breaks, respectively). For each of these postulated breaks, a thermohydraulic analysis was performed according to the procedures and models developed in Reference 2. All dynamic analysis methodology used was the same as in the original analysis.

←(DRN 00-1059)

## 5.4A.6 RESULTS

Results of maximum reactor vessel supports loads are given in Table 5.4A-1 for all postulated breaks.

It can be seen from Table 5.4A-1 that the 350 in.<sup>2</sup> inlet break produces the controlling values for all support force components. The table also shows the comparison results between the original and revised analyses for the controlling 350 in.<sup>2</sup> break. This comparison demonstrates that the original analysis provided conservative results. In addition, the table demonstrates that no updated support load exceeds any originally specified load. All calculated support loads in the table have been evaluated and found to be acceptable.

Figure 5.4A-15 shows vessel vertical support reactions both with and without the internal loads for the original 350 in.<sup>2</sup> RV inlet nozzle guillotine analysis. The major effect of the internal loads on the vertical supports is to increase the maximum compressive reaction by a factor of three.

Figure 5.4A-16 shows horizontal support reaction with and without the internal loads for the same analysis. The major effect of the internal loads on the horizontal supports is to shift the phasing of the load response. The maximum horizontal support load with internal applied loads is not greater than the maximum horizontal support load without internal

→(DRN 00-1059)

←(DRN 00-1059)



applied loads, but they occur at different times. The effect is due to the combination of a relatively low ratio of horizontal concrete stiffness to vertical concrete stiffness (under the support pads). This combination causes the pads to take vertical load in resisting the overturning effects of the horizontal forces on the vessel.

Inside the vessel, interface gaps are found to open and close many times. It can be seen from Figure 5.4A-15 that the total vertical effect of all loads on the system can cause the vessel to overcome its compressive dead weight and thermal growth loads on the support pad, traverse the vertical support gap, and engage the anchor bolts. The vessel does lift off, from as many as all four pads simultaneously, but most often it remains in contact, oscillating on one or more compressed pads. The horizontal effect of all loads on the system causes the vessel to traverse the horizontal support gaps, impact the supports and remain in contact oscillating about a deflected, closed gap position.

REFERENCES: SECTION 5.4A

1. CENPD-42, Topical Report on Dynamic Analysis of Reactor Vessel Internals, December 1973.
2. CENPD-252-P-A, Topical Report, "Method for the Analysis of Blowdown Induced Forces in a Reactor Vessel," July 1979.
3. ICES STRUDL II. The Structural Design Language Engineers Users Manual, M.I.T. Press., Cambridge, Massachusetts, 1968.
4. Fritz, R. J. and Kiss, E., The Vibration Response of a Cantilevered Cylinder Surrounded by an Annular Fluid KAPL-M-6539m Knolls Atomic Power Laboratory, Schenectady, New York.
5. CENPD-168A, Topical Report on Design Basis Pipe Breaks, June 1977.
6. Lien, J. S., R. P. Kassawara, H. B. Smith, Dynamic Analysis of Piecewise Liner Structures, ASCE Specialty Conference, New Orleans, Louisiana, 1975.
7. Kassawara, R. P., and Peck, D. A., Dynamic Analysis of Structural Systems Excited at Multiple Support Locations, SCE Specialty Conference, Chicago, Illinois, 1973.

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TABLE 5.4A-1

RESULTS OF LOCA ANALYSIS OF REACTOR VESSEL SUPPORTS

Maximum LOCA Absolute Reactor Vessel Support Loads

Accident	Force Component (See Figure 5.4A-14)		
	H	V (pad)	V (bolts)
100 in <sup>2</sup> RV Outlet Nozzle Guillotine <sup>(1)</sup>	2721	514	
350 in <sup>2</sup> RV Inlet Nozzle Guillotine <sup>(1)</sup>	3759	2106	
350 in <sup>2</sup> RV Inlet Nozzle Guillotine <sup>(2)</sup>	3625	1403	
600 in <sup>2</sup> SG Inlet Nozzle Guillotine <sup>(2)</sup>	1566	145	
Design Maximum LOCA Loads	5426	5200	4530

(1) Using condensed RCS model and internal loads from Reference 1 methodology

(2) Using full RCS model and internal loads from Reference 2 methodology

WSES-FSAR-UNIT-3

TABLE 5.4A-2

REACTOR COOLANT SYSTEM MODEL

<u>Component</u>	<u>Mass Point Number</u>	<u>Degrees of Freedom</u>	<u>Hydrodynamic Coupling</u>	<u>Active Directions</u>	
Reactor	9918	XYZ	9918-18	XZ	
	9913		9999-16		
Vessel	9909	XZ	9909-9		
	9905		9905-5		
	9914		9914-14		
	9999				
Core Support Barrel	18	XYZ	Gap Locations	Degrees of Freedom	Initial Condition
	9				
	13		9918-18	Y	Closed with Compressive Preload
	5		9918-38		
	14		18-38		
16	28-29				
Lower Support Structure	2	XYZ	9914-14	XZ	Open, Gap ±
			9999-16	X	
Fuel Assembly	20	XYZ			
	24				
	28				
Core Shroud	22	XZ			
	26				
Upper Guide Structure	6	XYZ			
	10		XZ		
	38	XYZ			
	32				
	29				

WSES-FSAR-UNIT-3

TABLE 5.4A-3

PIPE BREAK SUMMARY

<u>Pipe</u>	<u>Location</u>	<u>Break Type Description</u>	<u>Flow Area (Sq. in..)</u>
	R.V. terminal end	circumferential break	100
hot leg	S.G. terminal end	circumferential break	600
	R.V. terminal end	circumferential break	350
discharge leg	Pump terminal end	circumferential break	480
	Pump terminal end	circumferential break	430
suction leg	Pump elbow	slot $\pm 90^\circ$ from elbow crotch	532
	S.G. elbow	slot $\pm 90^\circ$ from elbow crotch	532
	S.G. terminal end	circumferential break	592

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Note: Flow areas listed are total area available for flow from both sides of a given break location.

WSES-FSAR-UNIT-3

TABLE 5.4A-4

EXPANDED REACTOR COOLANT SYSTEM MODEL

350 IN.<sup>2</sup> RV INLET NOZZLE BREAK

In addition to those mass degrees of freedom shown in Table 5.4A-2, the following were included for this break:

<u>Component</u>	<u>Mass Point Number</u>	<u>Degrees of Freedom</u>
R.C. Pumps (in intact legs)*	1103	XYZ
	2103	
	5103	
	1101	XZ
	2101	
	5101	
Intact R.C. Piping*	1760	XYZ
	2760	
	5760	
	1580	XYZ
	2580	
	5580	
SG	800	XYZ
	3800	
	409	XZ
	3409	
	404	XYZ
	3404	

\* For break postulated at the 2B RV inlet nozzle.

WSES-FSAR-UNIT-3

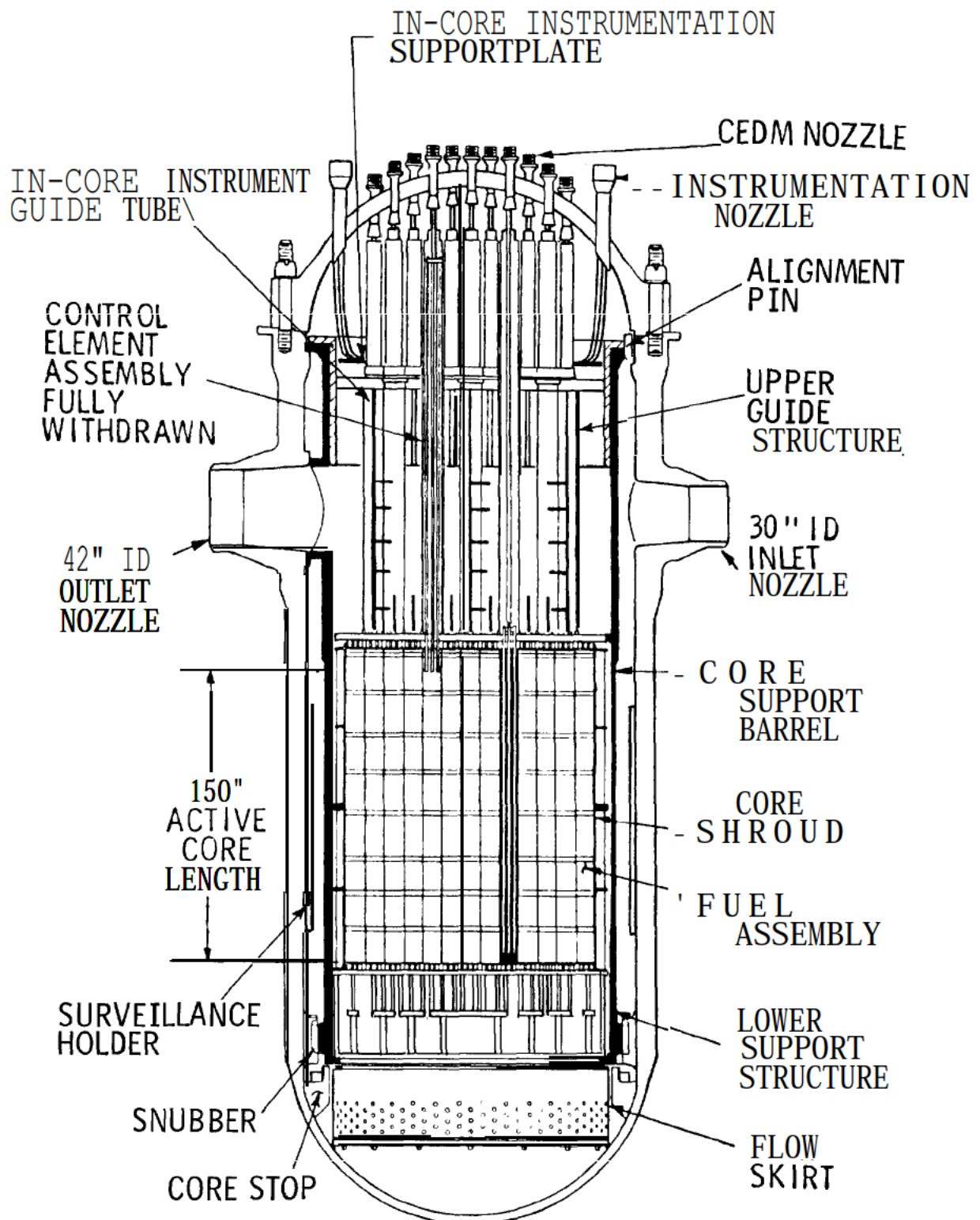
TABLE 5.4A-5

EXPANDED REACTOR COOLANT SYSTEM MODEL

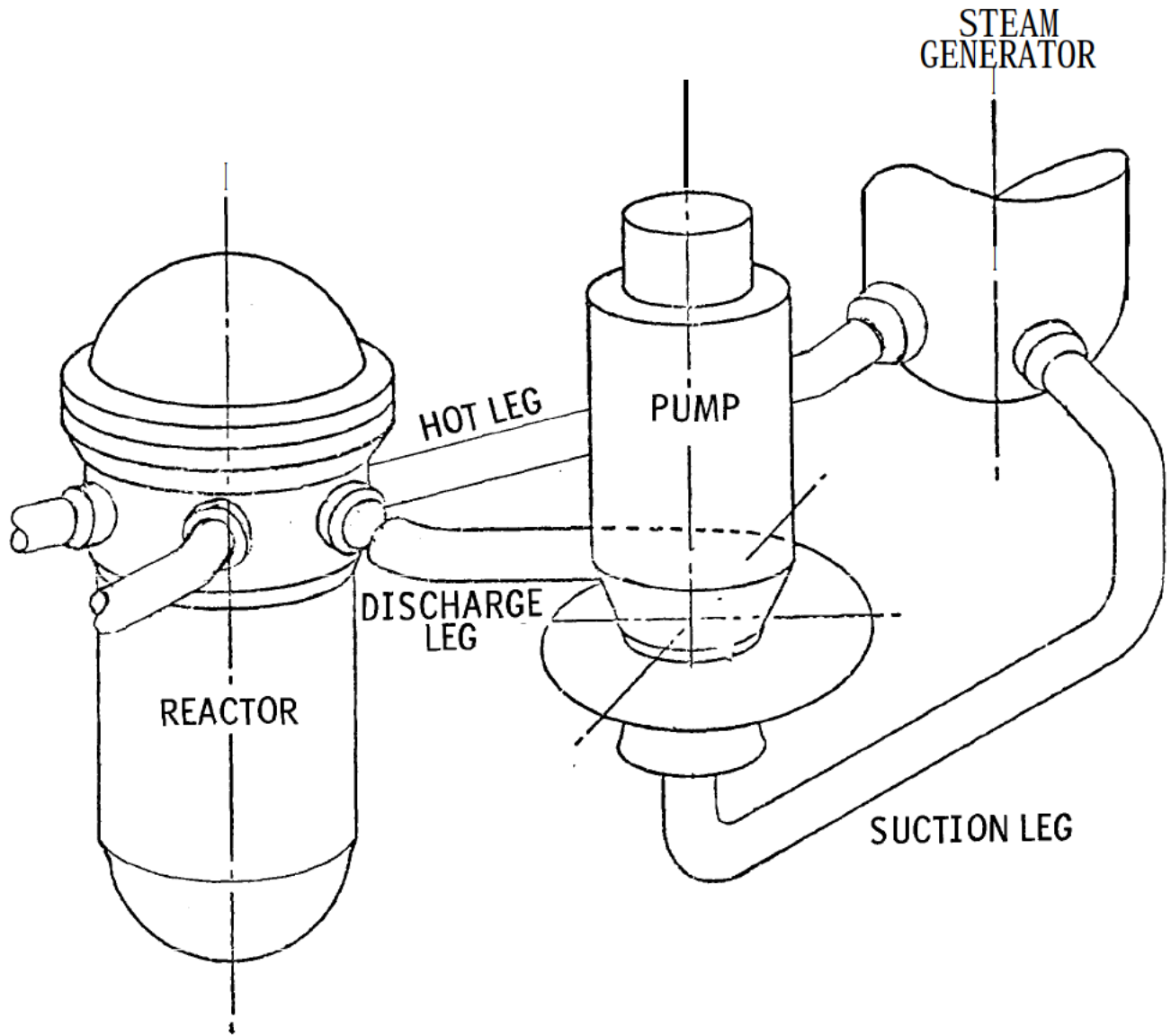
600 IN.2 S.G. INLET NOZZEL BREAK

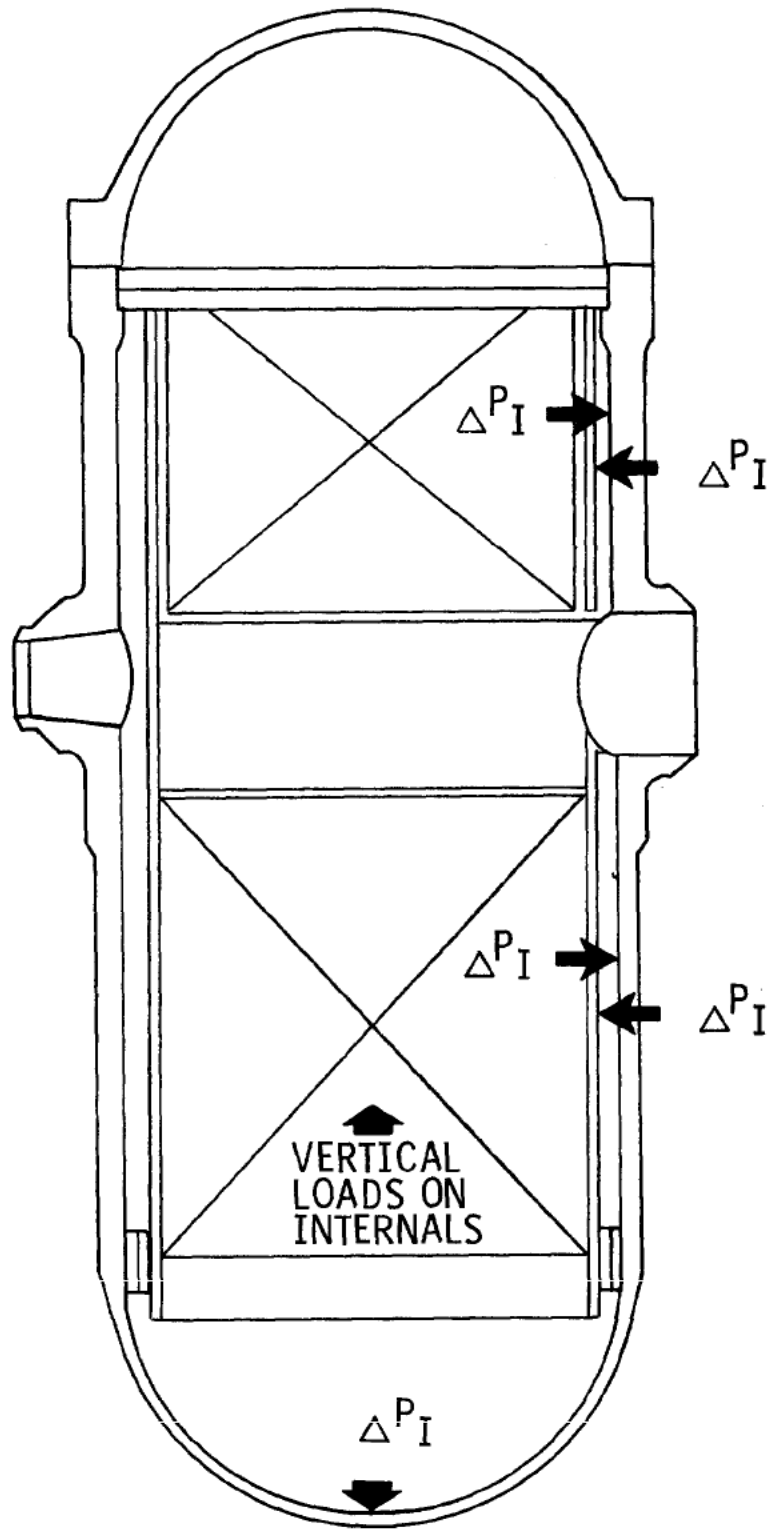
In addition to those mass degrees of freedom shown in Tables 5.4A-2 and 5.4A-4, the following were included for this break:

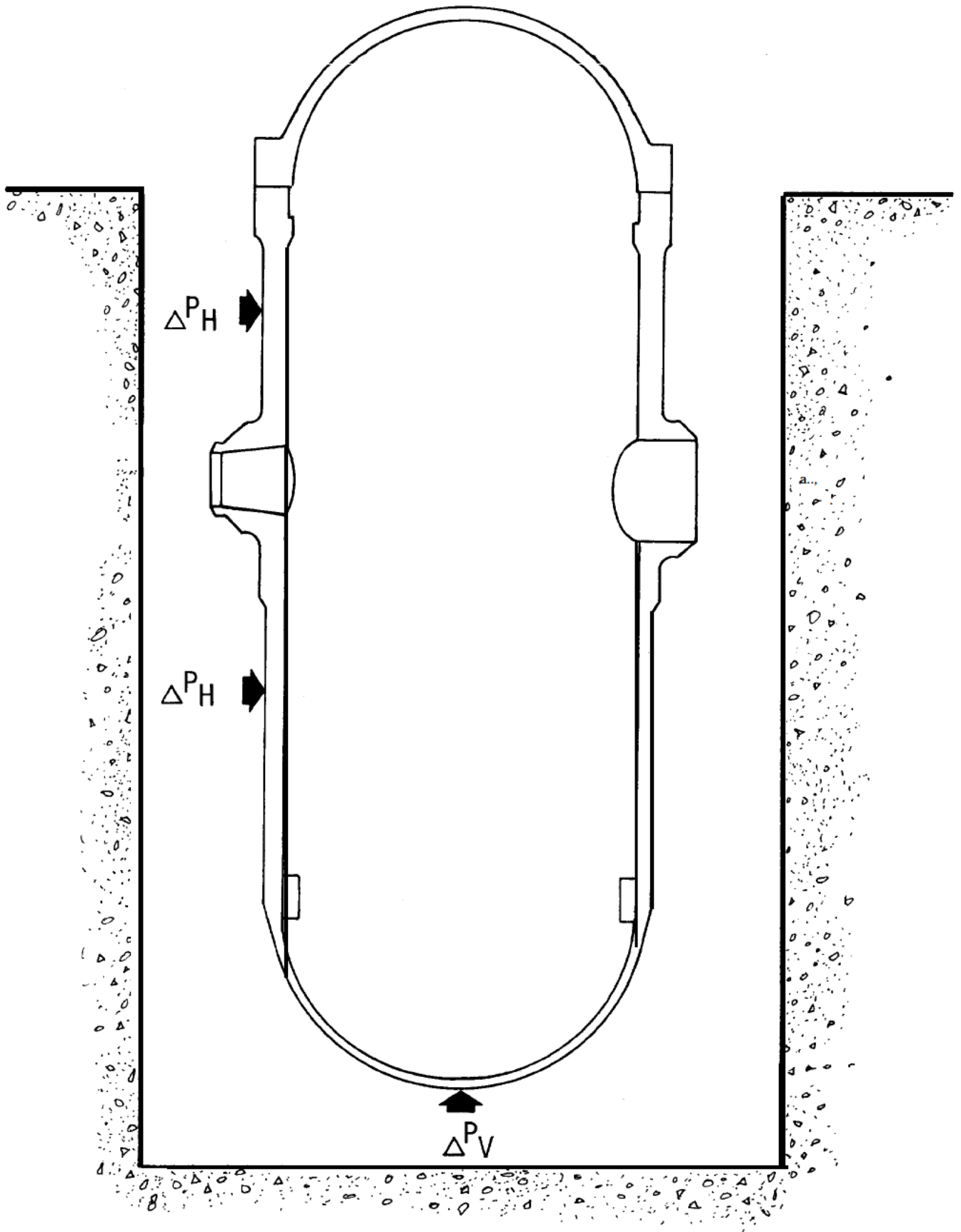
<u>Component</u>	<u>Mass Point Number</u>	<u>Degrees of Freedom</u>
Reconnected 2B R.C. Pump and Piping	4103	XYZ
	4101	XZ
	4760	XYZ
	4580	XYZ
Severed Hot Leg	3851	
	3850	XYZ
	38501	
	3550	XY
	3751	







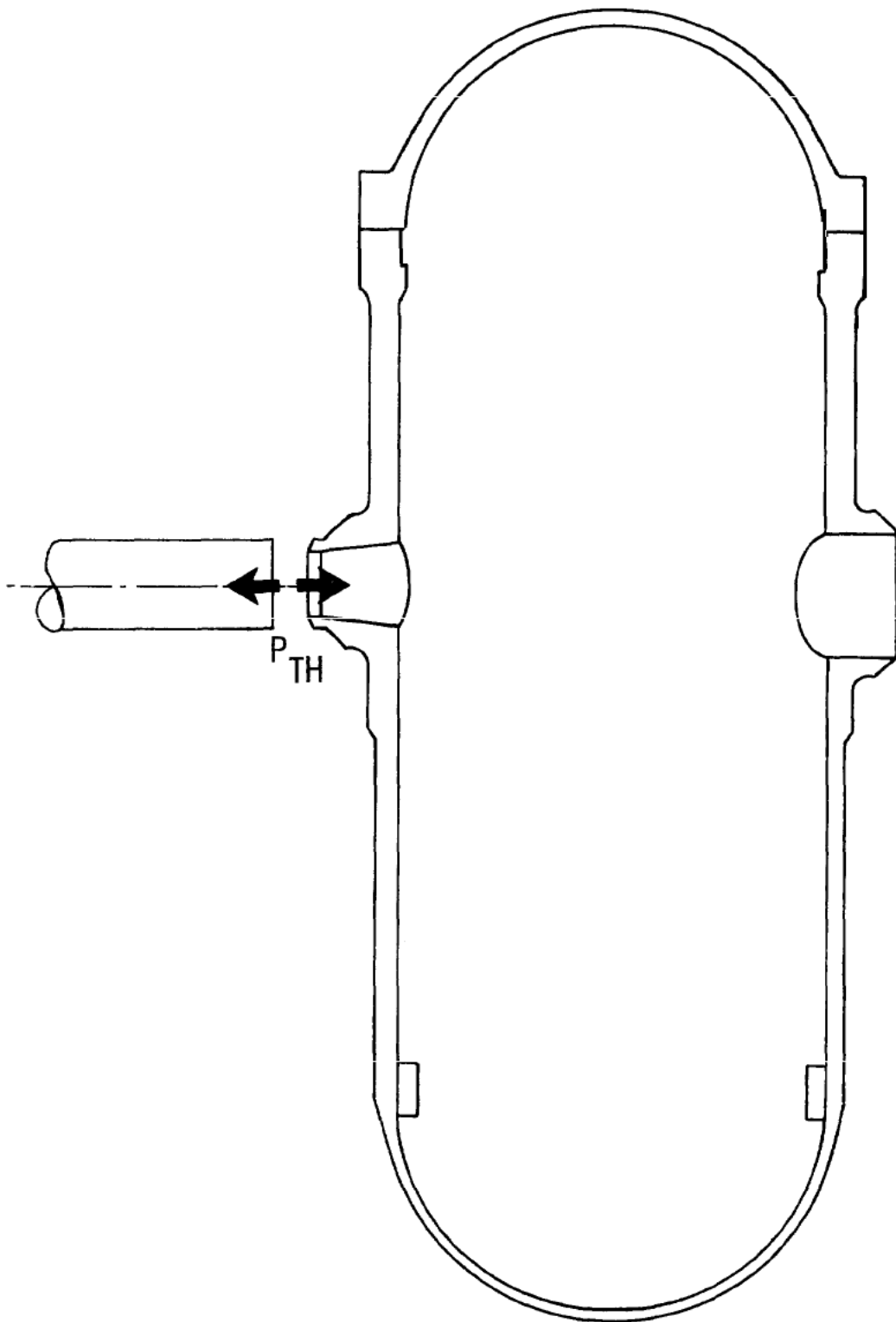




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RV ASYMMETRIC LOADS ANALYSIS  
REACTOR CAVITY PRESSURE LOADS

Figure  
5.4A-4



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RV ASYMMETRIC LOADS ANALYSIS  
PIPE THRUST LOAD

Figure  
5.4A-5

$P_I^V$  = VERTICAL HYDRAULIC FORCES ON INTERNALS

$P_R^H$  = HORIZONTAL HYDRAULIC FORCES ON VESSEL

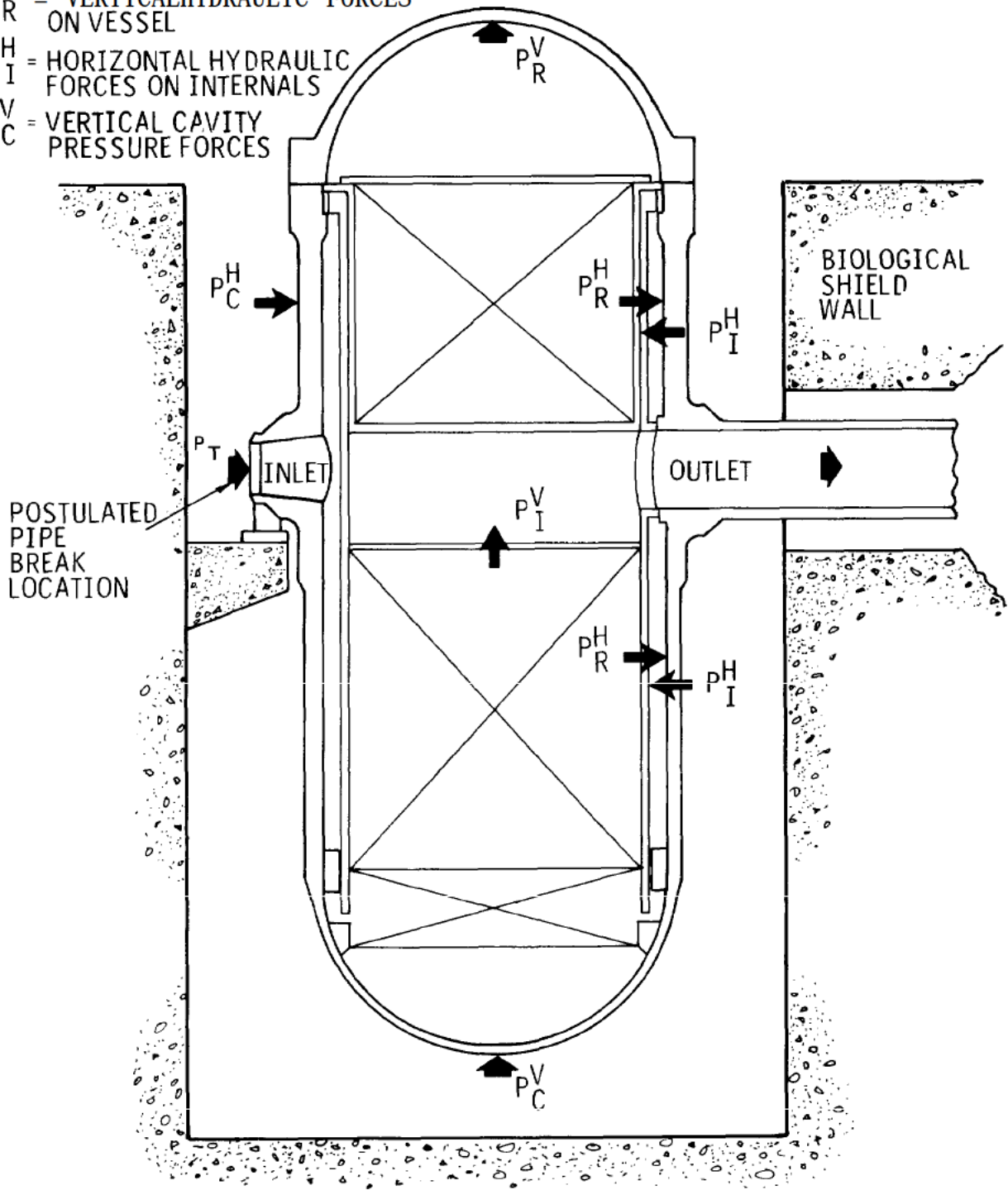
$P_R^V$  = VERTICAL HYDRAULIC FORCES ON VESSEL

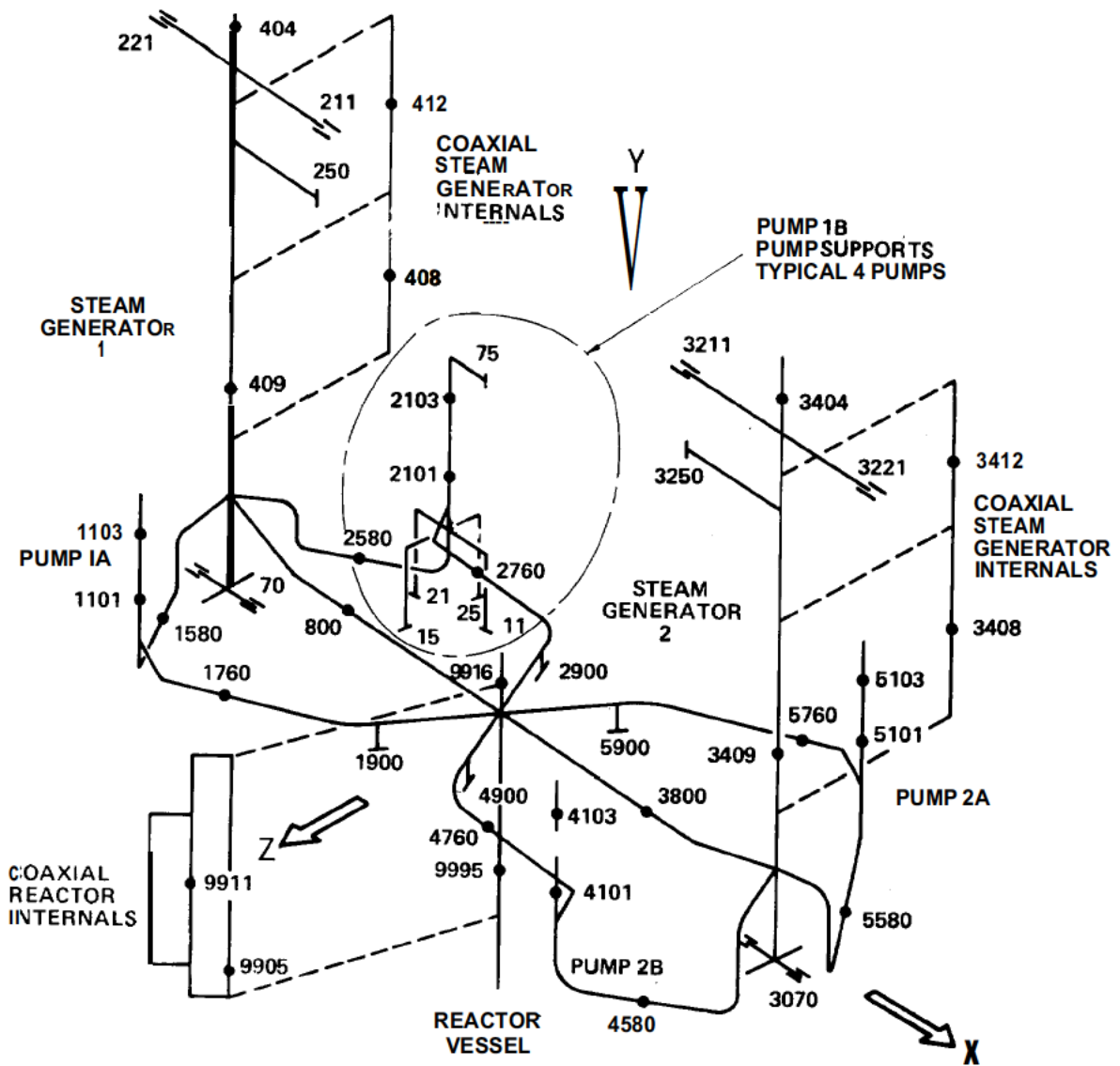
$P_I^H$  = HORIZONTAL HYDRAULIC FORCES ON INTERNALS

$P_C^V$  = VERTICAL CAVITY PRESSURE FORCES

PT = PIPE BREAKTHRUST

$P_C^H$  = HORIZONTAL CAVITY PRESSURE FORCES



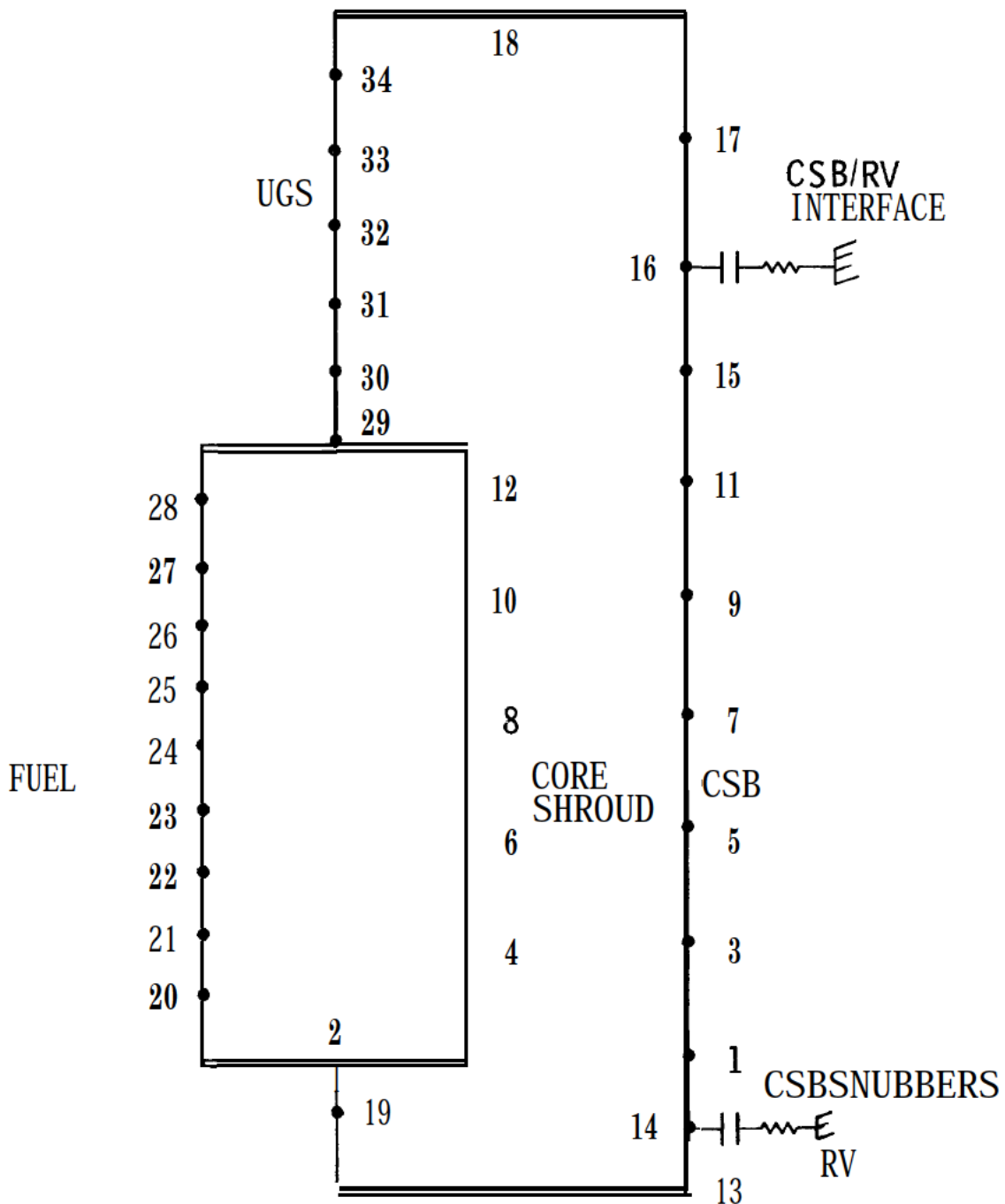


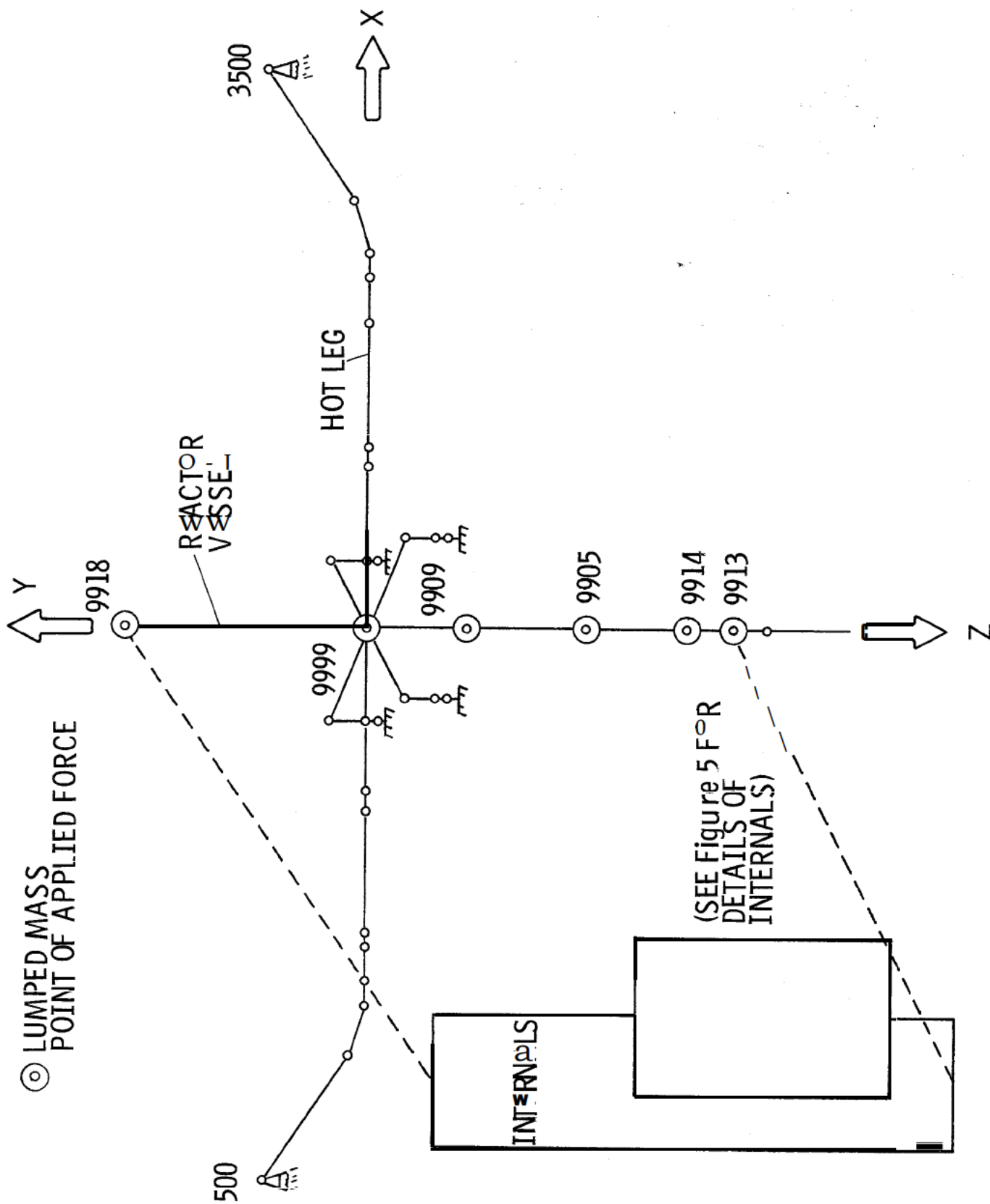
COMPONENT NAME	MASSPOINT NUMBER	DEGREES OF FREEDOM	COMPONENT NAME	SUPPORT POINT NUMBER	RESTRAIN1
REACTOR	9916, 9911, 9905 9995	X, Y, Z X, Z	REACTOR	1900, 2900, 4900, 5900	FX, FZ FIXED
STEAM GENERATORS	404, 3404, 412, 3412 409, 3409 408, 3406	X, Y, Z X, Z X, Z	STEAM. GENERATORS	70, 3070 250, 3250 211, 221 3211, 3221	FY, FZ FX FZ FZ
PUMPS	1103, 2103, 4103, 5103 1101, 2101, 4101, 5101	X, Y, Z X, Z	REACTOR COOLANT PUMPS (TYPICAL)	11, 15, 21, 25 75	FY FX, FZ FX, FZ
REACTOR COOLANT PIPING	800, 3800 1760, 2760, 4760, 5760 1580, 2580, 4580, 5580	Y, Z X, Y, Z X, Y, Z			

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Electric Station

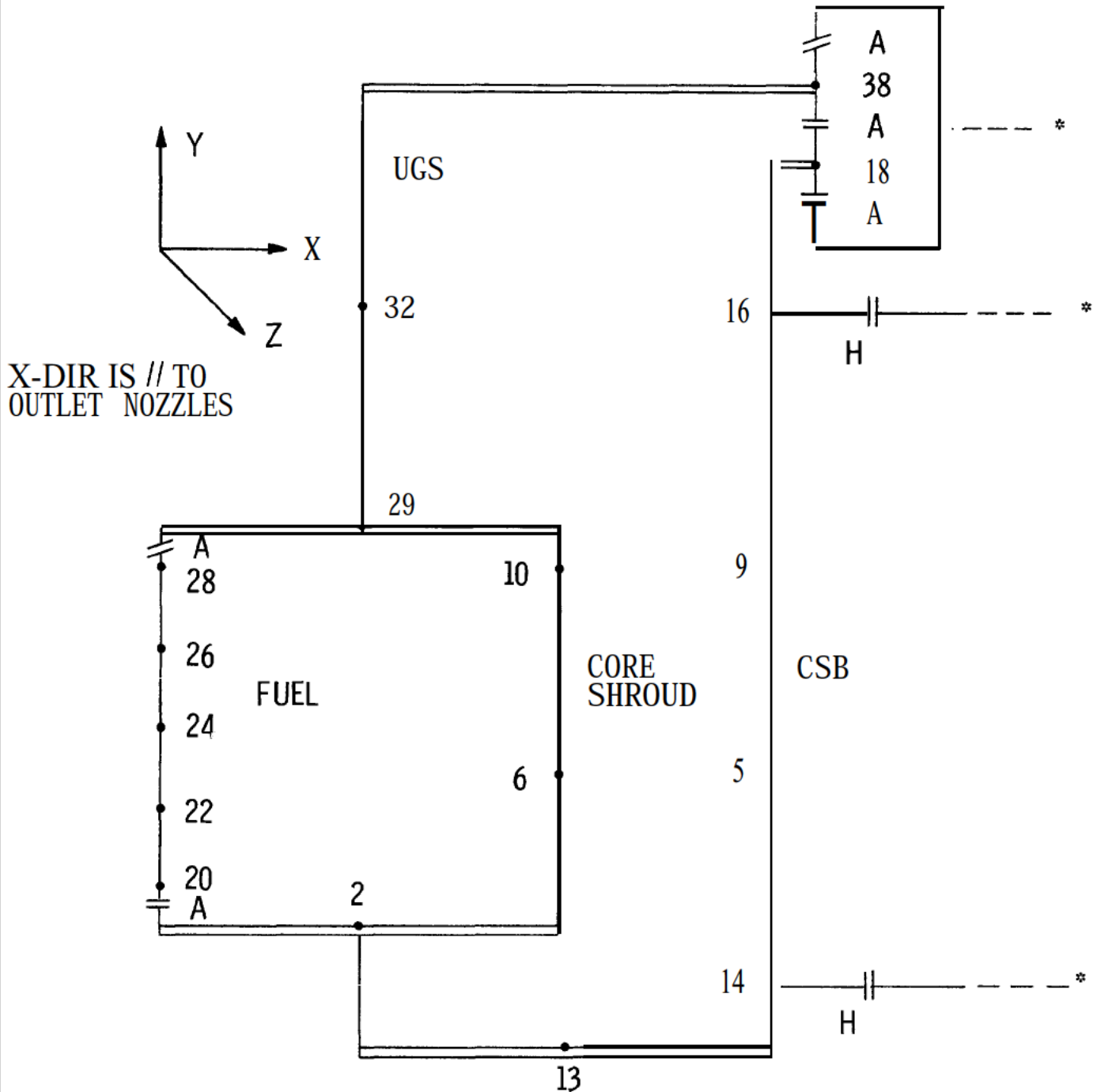
RV ASYMMETRIC LOADS ANALYSIS  
DETAILED REACTOR COOLANT SYSTEM MODEL

Figure  
5.4A-7





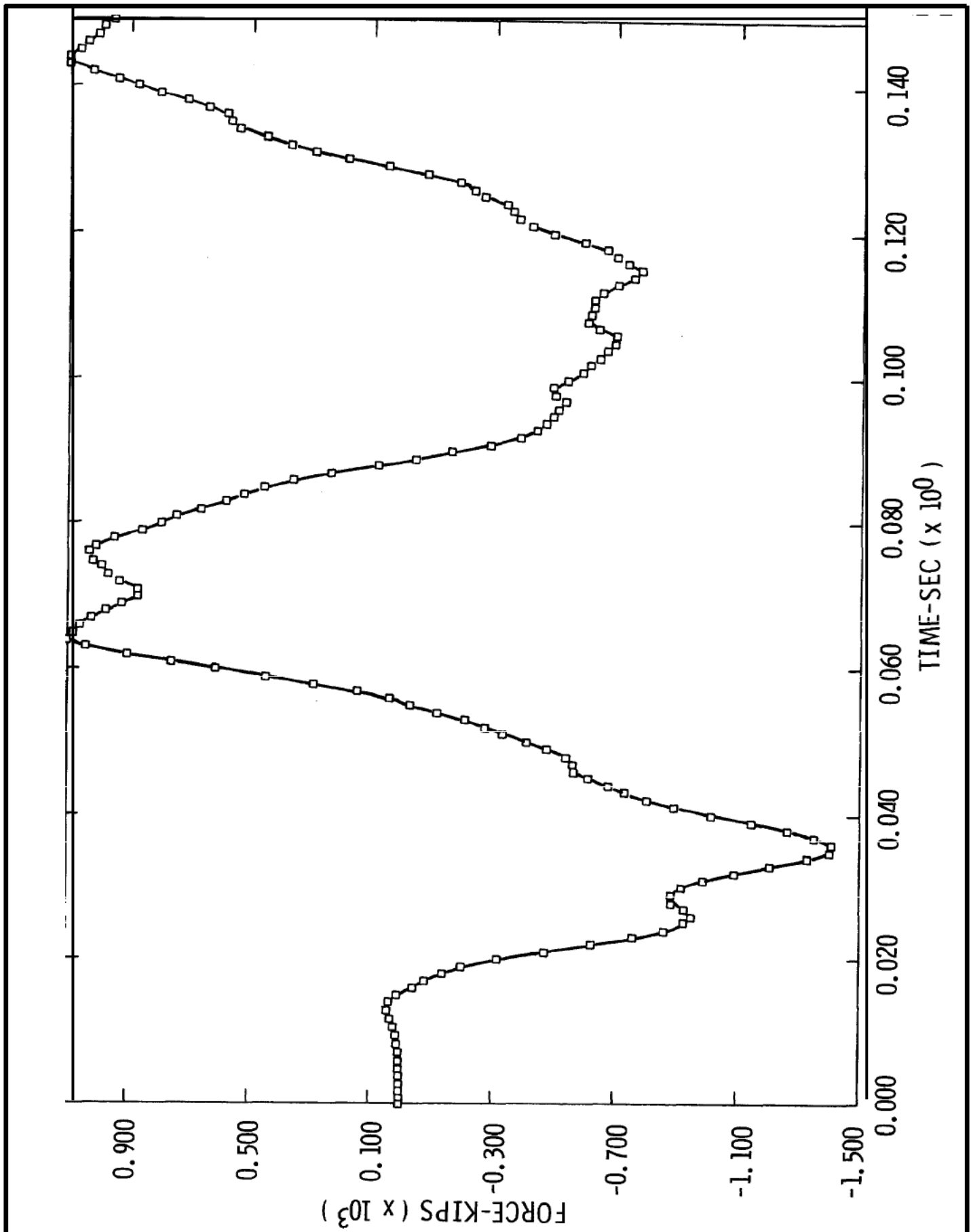




\* SEE Figure 9 FOR DETAILS OF REACTOR VESSEL AND PIPING

**LEGEND**

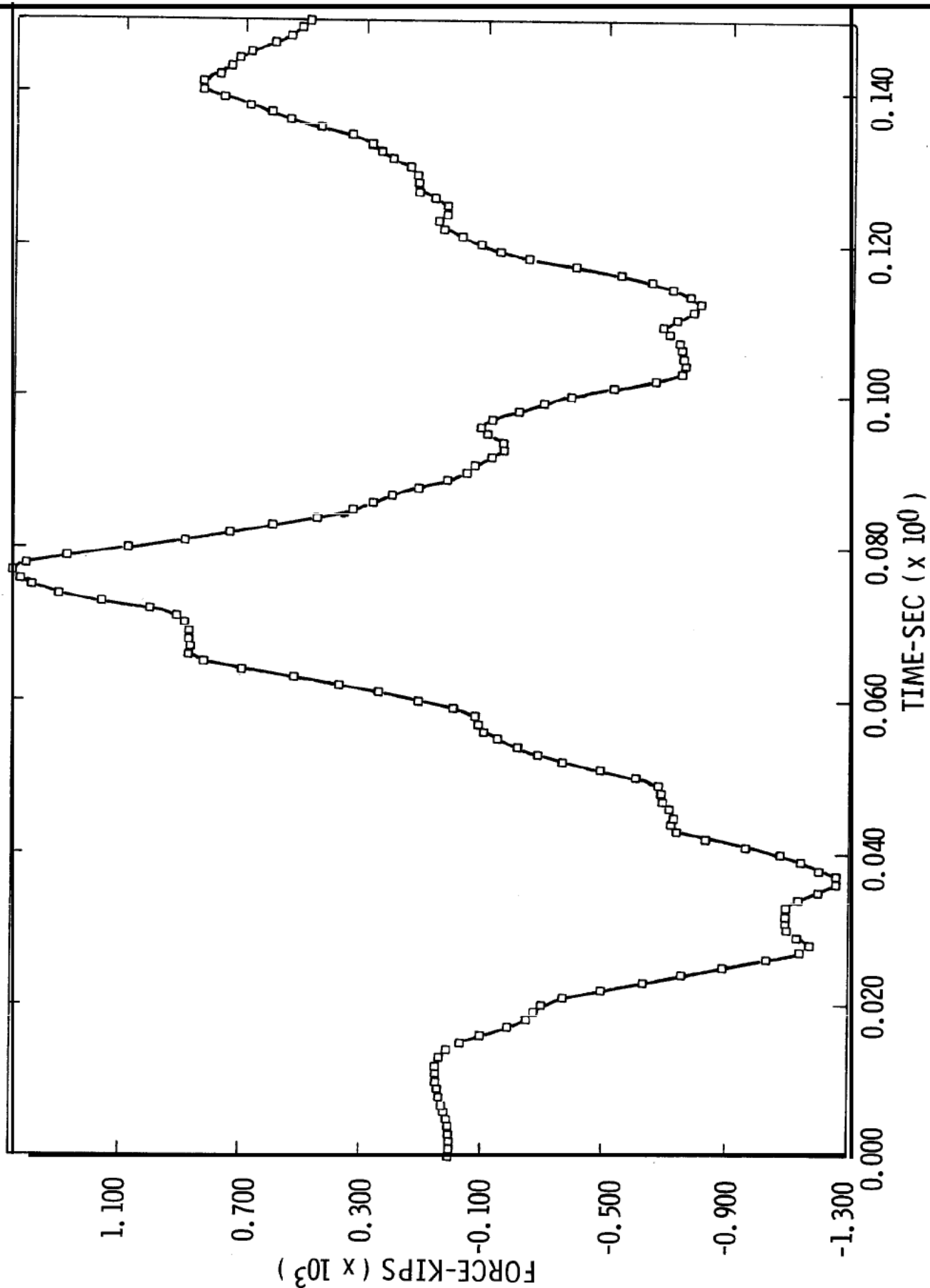
- A = AXIAL GAP
- H = HORIZONTAL GAP
- // = PRELOADED COUPLING
- | = GAP COUPLING
- == = COLINEAR CONNECTOR



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Electric Station

RV ASYMMETRIC LOADS ANALYSIS  
REACTOR VESSEL SUPPORT REACTIONS  
SIMPLIFIED INTERNALS AND VESSEL

Figure  
5.4A-11

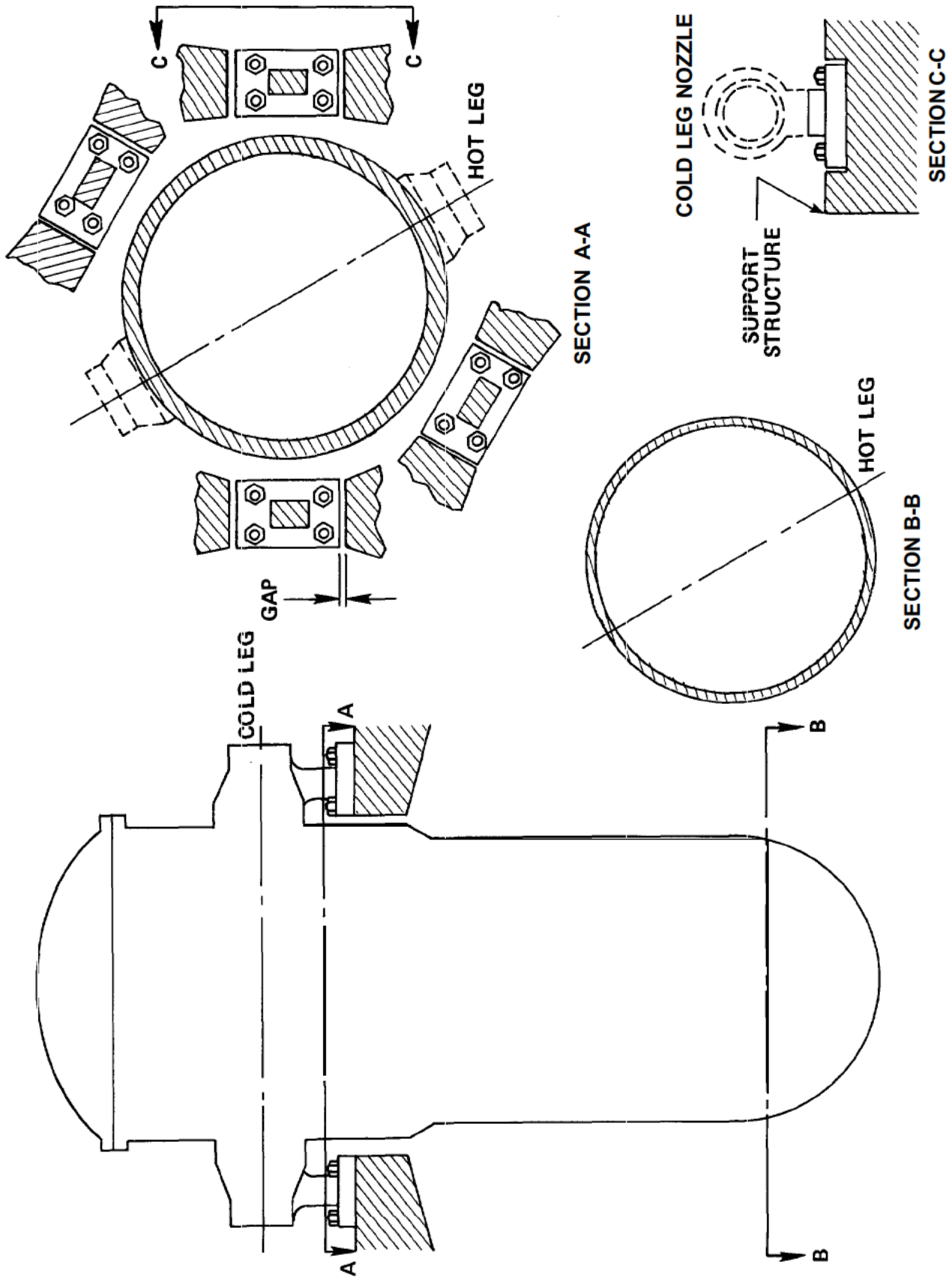


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Electric Station

RV ASYMMETRIC LOADS ANALYSIS  
REACTOR VESSEL SUPPORT REACTIONS  
DETAILED INTERNALS, SIMPLIFIED VESSEL

Figure

5.4A-12

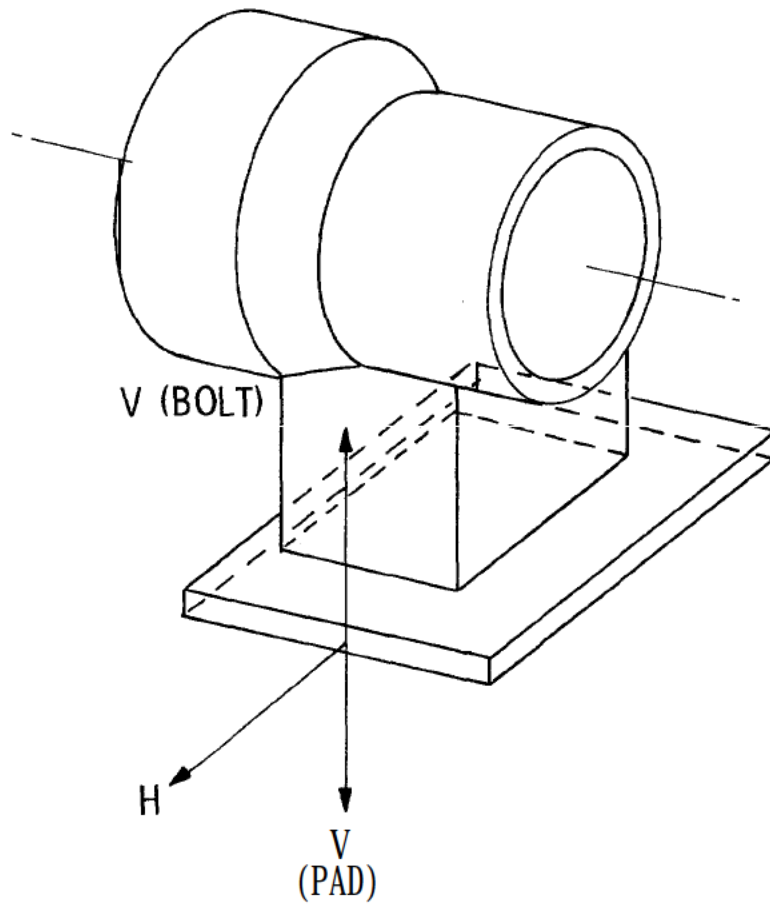


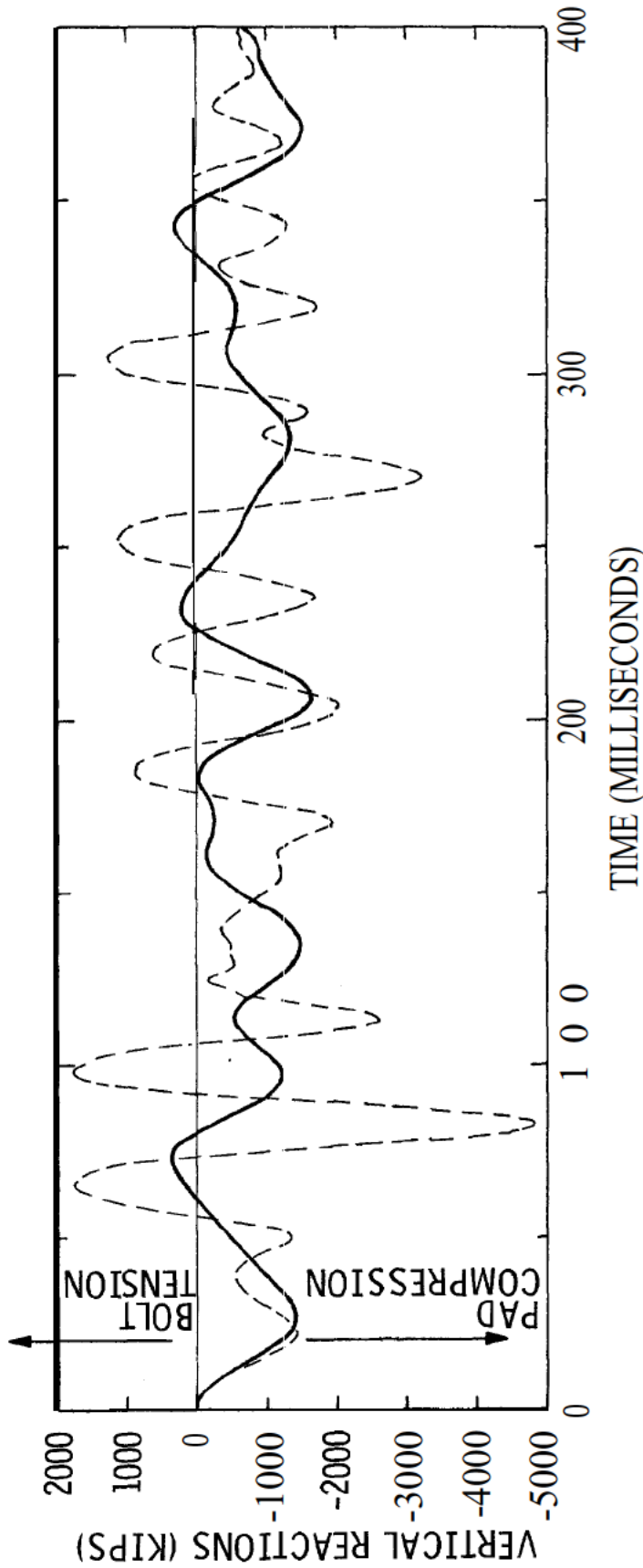
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Waterford Steam  
Electric Station

RV ASYMMETRIC LOADS ANALYSIS  
REACTOR VESSEL LOCA SUPPORTS

Figure  
5.4A-13

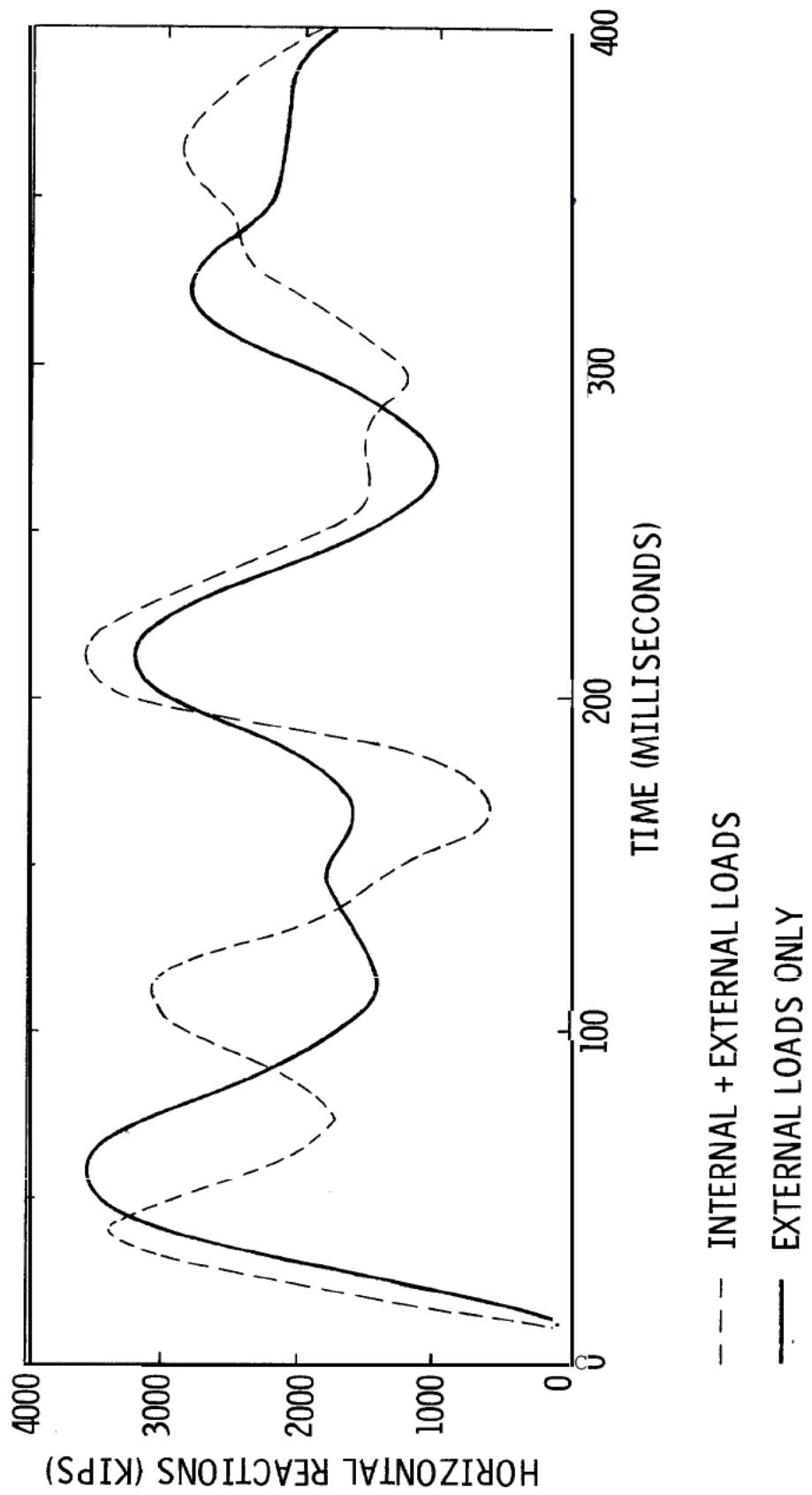
RV CENTERLINE





- - - INTERNAL + EXTERNAL LOADS    (+) TENSION, ALL BOLTS, PER 1 SUPPORT  
 ——— EXTERNAL LOADS ONLY        (-) COMPRESSION, 1 PAD

NOTE: PLOT IS FOR PIPE RUPTURE LOADS ONLY. FOR PIPE RUPTURE + NORMAL OP., SUBTRACT 880 KIPS FROM THE BOLT LOAD AND ADD IT TO THE PAD LOAD.



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RV ASYMMETRIC LOADS ANALYSIS  
350 SQ. IN. COLD LEG GUILLOTINE AT RV NOZZLE  
MAX. RV HORIZONTAL SUPPORT LOAD vs TIME

Figure  
5,4A-16